Special Symposium:
50 Years of
Nuclear Fission in Review

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50 Years of Nuclear Fission in Review

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Editor: Malcolm Harvey

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Preface

The discoveries of 1939, that a heavy nucleus can undergo fission and that the subsequent release of neutrons could induce a chain reaction, were events that have changed the course of human history. Suddenly mankind had the means to release an enormous amount of energy with which to wage horrific warfare or create a better quality of life for the ever-growing world population. It is no surprise that fifty years later a number of conferences should be held around the world to recognize the anniversary and to review the consequences.

Canada was not directly associated with the initial discovery of fission - this having been centered in Europe through the activities of Lise Meitner, Otto Frisch, Otto Hahn, Fritz Strassman, Irène Curie and Pavel Savic. Neither was Canada directly involved with the later demonstration of the first man-induced chain reaction - this honour going to Enrico Fermi and his Chicago group. But neither was Canada a neophyte in the field of nuclear physics. The early research of Ernest Rutherford, the father of nuclear physics, was done in McGill University at the beginning of the century and he was to play an influential role in the life of Otto Hahn - as Les Cook recounts in his talk. George Laurence, who went on to become one of Canada's most eminent scientists, was building a nuclear "pile" at the same time as Fermi. Partly through these influences, and because of its fortunate geographic location and the turmoil in Europe during the Second World War, Canada was able to join in with the research associated with fission at a very early stage and could quickly benefit from the stimulus that this gave to a wide range of scientific and industrial fields. It is most appropriate, therefore, that a Special Symposium recalling these events should form part of the Annual Conference of the Canadian Nuclear Association and the Canadian Nuclear Society in Ottawa, June 5, 1989.

The Special Symposium was designed to highlight how the technical information reached our shores and the effect that this discovery had in Canada in the fields of physics, chemistry, medicine and nuclear power. Six speakers were chosen who were pioneers of the period and who were each eminently qualified to review the separate fields.

As principal organizer of the Symposium, it is with deep gratitude that I acknowledge the cooperation from the distinguished speakers - Bertrand Goldschmidt, Les Cook, Geoff Hanna, John Foster, Alvin Weinberg and Sylvia Fedoruk - in firstly addressing the meeting and then in providing me with their manuscripts. The reader will find in this collection of talks a snapshot of the birth of nuclear technology in Canada, its subsequent growth and influence on the sciences. I can only hope that this small volume will stimulate a work of much greater dimension to record in detail the achievements of the past fifty years.

I would like to thank Robert Bothwell who, through his introduction, brought to the event the perspective of a non-technical person who is nevertheless an authority on the history of the nuclear industry in Canada.
I am very grateful to Ara Mooradian who had the unenviable task, as chairman, of limiting the speakers to their allotted time which was regretfully all too short for the wealth of information. Ara’s sage advice on the structure for the Symposium was much appreciated.

This Symposium would not have occurred without the help, advice and encouragement from my subcommittee of Gerald Dolling, Paul Fehrenbach and Alec Stewart, and from Ron Veilleux and Dave Cowper of the CNA/CNS organising committee.

Finally, I would like to thank those who have helped put this volume together: Monique Lapointe for the collation; Charles Gale and Martin Elliott for their technical advice and help in transferring files from an IBM computer onto a MacIntosh through which the final manuscript was prepared; Geoff Hanna, Varley Sears and Gerald Dolling for their help in proof reading; and last, but by no means least, Margaret Carey who retyped all manuscripts and patiently suffered the whims of the editor.

In preparing this volume I have kept the editing to a minimum in order to preserve the style of the various speakers. I apologize for any errors that might have crept in and hope that any blemishes do not detract from the enjoyment and value of the text.

Malcolm Harvey
Chalk River, Ontario
August, 1989
Chairman, Moderator, and Speakers

From left to right Ara Mooradian, Geoff Hanna, Bertrand Goldschmidt, Leslie Cook, Robert Bothwell, John Foster.

Sylvia Fedoruk.
Born in 1922 in Hamilton, Ontario, Ara Mooradian gained his early training as an engineer and scientist at the University of Saskatchewan and the University of Missouri. His career began at the nitrogen division in the Consolidated Mining & Smelting Company before joining the staff at the Chalk River Nuclear Laboratories. At Chalk River he has been Head of the Development Engineering and Fuel Development Branches. In 1964 Ara Mooradian became the Managing Director of the Whiteshell Nuclear Research Establishment of Atomic Energy of Canada Limited and later the Vice-President-in-Charge. He became Vice-President of the Chalk River Nuclear Laboratories in 1971 before taking up the position of Executive Vice-President for Research & Development at AECL (1977) and then Corporate Vice-President (1978). Ara Mooradian is noted for his contributions to the development of low cost fuel for CANDU nuclear power generating stations. His honours have included the Canada Medal, the W.B. Lewis Award and Fellowships of the Royal Society of Canada and the Chemical Institute of Canada. His extra-curricular activities have included being the first Mayor of the Town of Deep River. He is noted for his furniture design and construction, and his enthusiasm for the outdoors.
Born in Ottawa, in 1944, Professor Bothwell was educated at the University of Toronto and Harvard University. He returned to the University of Toronto in 1970 and is now Professor of History there. Professor Bothwell has been editor of the "Canadian Historical Review", and has received the Corey Prize of the Canadian and American Historical Associations. He is a member of the Canadian Committee for the History of the Second World War and has authored many books, articles and reviews both in the United States and Canada. Among those of significance to the developments of science since the discovery of fission are "Nucleus: A History of Atomic Energy of Canada Ltd.", "Eldorado: Canada's National Uranium Company" and, with co-author William Kilbourn, "C.D. Howe: A Biography".
Dr. Bertrand Goldschmidt, born in Paris in 1912, was the last personal assistant of Marie Curie who engaged him in 1933 (the year before her death) at the Radium Institute where he got his Ph.D. and worked until 1940.

During the war, he participated in the atomic projects in North America as a member of the British team to which he had been seconded by the Free French. He worked in Glenn Seaborg’s group on plutonium extraction at Chicago during the summer of 1942. Then he joined the Anglo-Canadian project, from its beginning at Montreal, as section leader of the Chemistry Division, which he directed in Chalk River before returning to France in 1946.

One of the founders of the French Atomic Energy Commission in 1946, he was in charge of its Chemistry Division till 1960 and of its International Relations Division till his retirement. Bertrand Goldschmidt is noted for his expertise in the chemistry of radioactive elements, the extraction of plutonium and the metallurgy of uranium. He is the author of a number of books on the history of the development of nuclear technology, the latest being “Pionniers de l'Atome”.

Co-Laureate of the “Atoms for Peace” Award in 1967, he was the French Governor on the Board of the International Atomic Energy Agency from 1958 to 1980 and Chairman of this Board in 1980.
Leslie G. Cook

Born in Paris, Ontario in 1914 and educated at the University of Toronto, the University of Berlin and Cambridge University, Leslie Cook has had a distinguished career as a chemist and energy analyst. He has held positions of Research Chemist and Manager of Physics Research at the Aluminium Research Laboratories of the Aluminium Company of Canada; Director of Research of the Chemistry & Metallurgy Division of Atomic Energy of Canada; Manager of Research Program Planning at the General Electric Research Laboratory in Schenectady; Vice-President for Research Programs at the National Research Council of Canada, and Manager of Research Planning for Exxon Corporate Research in Linden, New Jersey. Dr. Cook now owns the consulting firm of L.G. Cook Associates in New Jersey and continues his enjoyment for music and piano. He brings to the Special Symposium the unique perspective of one who was in Otto Hahn's laboratory in 1936-1938 during the time that fission was discovered.
Geoffrey C. Hanna

Born in Stretford, Lancashire, England in 1920, Geoff Hanna was educated at Cambridge University. He started his career as a physicist working on radar development during the Second World War with the British Ministry of Supply and came to Canada in 1945 as part of the United Kingdom mission first in Montreal and later at the Chalk River laboratories. He joined the staff at Chalk River in 1950. Geoff Hanna has had a distinguished career in nuclear physics where he is an authority on alpha radioactivity; ionization from the slowing down of particles ranging from electrons to fission fragments; nuclear fission; and on neutron cross section measurements. He is noted for his development of instrumentation in particular the multi-channel pulse height analyzer. Since 1962 he has shouldered an increasing administrative load as Head of Nuclear Physics, Director of Physics Division and Director of Research at the Chalk River Nuclear Laboratories. In addition to these duties Dr. Hanna has played an active role in the funding of physics in universities through Canadian funding agencies. Geoff Hanna is a fellow of the Royal Society of Canada, has an honorary D.Sc. from McGill University, and has been president of the Canadian Association of Physicists. Now officially retired, Geoff continues as an advisor for Canadian physics.
Alvin M. Weinberg joined the original group that developed the first chain reactors at the University of Chicago in 1941. He has since been a leading figure in the development of nuclear energy: among his accomplishments was the proposal to use pressurized water reactors for nuclear submarine propulsion. He served as Research Director and then Director of the Oak Ridge National Laboratory from 1948 to 1973. In 1974 he was Director of the Office of Energy Research & Development in the Federal Energy Administration. From 1975 to 1985 he was the Director of the Institute for Energy Analysis of the Oak Ridge Associated Universities. He is now a Distinguished Fellow at the Institute.

Alvin Weinberg, a native Chicagoan, received his B.S., M.S., and Ph.D. degrees in physics at the University of Chicago. He collaborated with Eugene Wigner to write the standard book on nuclear reactor theory, “The Physical Theory of Neutron Chain Reactors”. In addition to his strictly scientific and administrative work, he has been a prolific writer on the interaction between modern technology and society. Characteristic of these writings has been his coining of phrases that have become part of our everyday language: “Big Science”, “Technological Fix”, and “Faustian Bargain” are words that have flowed from his pen. Many of his early essays were published in 1967 in a collection entitled “Reflections on Big Science”.

For his contributions to the development of nuclear energy, Alvin Weinberg has received the Atoms for Peace Award, the Harvey Prize, the Heinrich Hertz Award, and the Fermi Award. He is a member of the National Academy of Sciences, the National Academy of Engineering, the American Academy of Arts & Sciences, the American Philosophical Society, and a Foreign Member of the Royal Netherlands Academy of Sciences.

Dr. Weinberg lives in Oak Ridge, Tennessee with his wife Gene. When not engaged in his scientific work, he can usually be found playing the piano or perfecting his backhand on the tennis court.
John S. Foster

John Foster was educated at Dalhousie University and the Technical University of Nova Scotia, graduating with degrees in Mechanical and Electrical Engineering. He served as an engineering officer in The Canadian Navy during World War II and was later with Montreal Engineering as a design and field engineer on thermal power plant projects in Canada and Latin America.

In 1954, John Foster became a member of the team that performed the conceptual studies for NPD, Canada's first nuclear power plant, and for the next 20 years headed engineering and project management for CANDU nuclear power plants in Canada and abroad. He was also responsible for the first phase of the Bruce Heavy Water plant and the Nelson River HVDC Transmission Facilities. He was President of Atomic Energy of Canada Limited from 1974 to 1977. He returned to Monenco in 1978 and was a Vice-President of Monenco Companies until 1983, when he retired.

Dr. Foster served on the Council of the Association of Professional Engineers of Ontario from 1970 to 1973. He was a member of the Executive Committee of the Canadian Nuclear Association for several years and Chairman in 1980.

Dr. Foster has long been active in the World Energy Conference. A member of the Executive Committee of the Canadian National Committee for many years, he was Chairman from 1979 to 1982. He was Vice-Chairman of the Conservation Commission and Chairman of the Program Committee before election this year to the office of Chairman of the International Executive Council, the first Canadian to be so honoured.

Dr. Foster has honorary degrees from the Technical University of Nova Scotia and Carleton University and is the recipient of the W.B. Lewis Medal from the Canadian Nuclear Association, the Gold Medal of the Association of Professional Engineers of Ontario and the Julian C. Smith Medal of the Engineering Institute of Canada. He is a Fellow of the Royal Society of Canada, the Canadian Academy of Engineers and the Engineering Institute of Canada.
Sylvia Olga Fedoruk

Born in Canora in 1927 and educated at the University of Saskatchewan, Sylvia Fedoruk has had a distinguished career in medical physics, specializing in the use of radiation in the diagnosis and treatment of cancer. For thirty-five years Dr. Fedoruk was associated with the Saskatoon Cancer Clinic, where she served as Chief Medical Physicist, and with the Saskatchewan Cancer Foundation, where she was Director of Physics Services. Dr. Fedoruk has also held the positions of Professor of Oncology and Associate Member in Physics at the University of Saskatchewan. At the end of 1986 she took early retirement from the university.

Sylvia Fedoruk was involved in the development of the world’s first Cobalt 60 unit and one of the first nuclear medicine scanning machines. She served for fifteen years as a member of the Atomic Energy Control Board of Canada and has also served as a consultant in nuclear medicine to the International Atomic Energy Agency in Vienna.

An avid curler, Sylvia Fedoruk was inducted into the Canadian Curling Hall of Fame in 1986. In the same year she was voted Y.W.C.A. Woman of the Year and made a member of the Saskatchewan Order of Merit - the province’s highest honour. In 1987 she was named an Officer of the Order of Canada and received an honorary doctorate of science from the University of Windsor. She was also Vice-Chairman of the National Forum on Post-Secondary Education held in Saskatoon that year.

Dr. Fedoruk has been Chancellor of the University of Saskatchewan since 1986. Until her vice-regal appointment as Lieutenant Governor of Saskatchewan, September 7, 1988 she was a member of the Saskatchewan Commission on Direction in Health Care.
Chairman's Opening Remarks

Ara Mooradian

I anticipate a delicious problem this morning. I know our moderator and all of our speakers who are now with me on the stage. Each alone could profitably occupy the entire session in conveying their insights and perspective of the events and characters which have made for so eventful and significant a half century. I myself am not noted for brevity. There is a risk that we shall be tempted to review the past 50 years in real time.

If we are to receive a balanced and comprehensive overview, all of our speakers must be heard. I therefore apologize to you in advance if I am driven to arbitrarily impose time constraints. I understand that the proceedings of the session are to be published in full and that the authors will have a second opportunity to touch any points which should have been heard at this session. [Editor’s Note: this has indeed been done.]

This Special Symposium is to commemorate the 50th anniversary of the discovery of fission and has been jointly sponsored and organized by the CNA, and the CNS with co-sponsorship from the Royal Society of Canada, Canadian Society for Chemistry, Canadian Association of Nuclear Medicine, Canadian Association of Physicists, Canadian Society for Chemical Engineering, and the Chemical Institute of Canada.

Their timing is impeccable. Half a century is no more than a sufficient perspective from which to confirm the significance of the discovery, nor is it so long as to have lost the integrity of the history.

Many, including governments, make the mistake of characterizing atomic energy as merely an option, an industry, a project, or even a business. Events have now confirmed that its significance is nothing less than that predicted by the most visionary of the early pioneers. What was launched 50 years ago was nothing short of an era, a watershed that changed forever the outlook for humanity.

Never before has man had in his hands the key to so much power: power to light the world, power to preserve his health and environment, power to extend his knowledge and power to destroy himself.

These are no longer the projections of a handful of early visionaries. The past five decades have actually witnessed the demonstration of all of these capabilities. It seems that, simultaneously, we have been given the tools to get on with the business of world civilization while being subjected to
the most unforgiving discipline against the failure to do so.

To a significant extent, the next half century and beyond will be characterized by how well we manage our nuclear affairs. If we are to chart a sensible course, it is important that we understand the roots and significance of the epoch in which we find ourselves.

We are indeed, this day, commemorating the launching of an era. How better to do this than to hear the story directly from those present at the launching party and those who have played an important role in confirming its significance. All of our speakers today have authored distinguished careers and remain very active. Together they constitute an historical resource exceeding 200 years of pertinent total recall.

Now Ladies and Gentlemen, the time has come to turn you over to our moderator, Dr. Robert Bothwell, Professor of History, University of Toronto, who will introduce our speakers and impose a coherent historical perspective throughout our session this morning. Professor Bothwell is uniquely qualified for the task. While he is more broadly noted as a specialist in the history of twentieth century Canada, he is better known to this audience as the author of two works bearing directly on the Canadian nuclear industry - the first a history of Eldorado Nuclear and the latest "Nucleus - the History of Atomic Energy of Canada Limited" published just last year.

In the eloquently simple statement of C.D. Howe who authorized the Canadian nuclear program back in 1942, "Okay" Dr. Bothwell, "let's go".
Moderator's Remarks

Robert Bothwell

The history of atomic energy in Canada, as is indicated in this symposium, proceeds from international roots to national application, and from pure science—or rather sciences, recognizing that medicine, chemistry and physics are all represented in this gathering—into their application in the engineering of the Canadian nuclear programme, and finally, in Alvin Weinberg's paper, to reflections on the consequences of atomic energy, and the responsibility that an atomic program confers on those who design, build and manage it.

Our first speaker takes us back to the prehistory of the Canadian atomic project, to its French origins and to its accidental transfer, as part of the British atomic enterprise, to Canada. The British, in the early 1940s, gave some consideration to involving Canada, with its uranium resources and its wide-open spaces, possibly suitable for testing, with their atomic research. To Canada, therefore, they transferred a research team in 1942-43. The scientists and engineers on this team were to act in cooperation with the much larger and richer American atomic project. Keeping them in Canada preserved an element of British control, and also reflected the reluctance of the Americans to import another draft of foreign scientists, and possibly foreign influence, to the United States.

Bertrand Goldschmidt was part of that team. He connects us with the pioneer of atomic research, Madame Curie, whose student he was. He is well-known as a distinguished pioneer of atomic research, as a radio-chemist, subsequently as an atomic diplomat, and, incidentally, one of the foremost historians of atomic energy—one of the few who has made the transition from science to history with aplomb, distinction, and success. But what Dr. Goldschmidt wants most to be remembered for, he tells me, is his successful campaign to abolish Rule 11 at Chalk River—the rule that kept men scientists out of the women's quarters, and vice-versa.

Next we have another member of Goldschmidt's laboratory, Les Cook, another chemist, who came to the atomic project from the University of Toronto, via Berlin and Cambridge. Les Cook represents the connection that formed between Canadian science and the Cavendish laboratories; his career in Canada and the United States exemplifies both the opportunities and the difficulties that confronted Canadian science, and Canadian scientists, during and after the Second World War. To historians, Les is notable for his ability to encapsulate history in a telling personal
anecdote. It is a rare gift, and one that Les develops to perfection.

With Geoff Hanna, we have before us someone whom Ara Mooradian properly describes as a "scientist's scientist", one who has enjoyed a distinguished career at Cambridge, at Malvern during the war, and subsequently at Chalk River, where he has successively headed up nuclear physics, then physics, and finally research. Geoff represents, as well, the British link with Chalk River, for as we know Chalk River was the fruit of a collaboration between Canada and Britain, a collaboration that, as the British atomic historian Margaret Gowing has argued, was rather thoughtlessly broken off in 1945-46. The result was a diminution of the contacts between Canadian and British atomic science and scientific development, probably to the detriment of both sides. It was not the first example of the influence of politics on Canada's atomic project — the project was after all the fruit of a political decision — but it was not one of the happiest occasions.

John Foster represents the transition from the relatively pure laboratory period of the late 1940s and early 1950s to the application of scientific knowledge to the practical construction of reactors. The decision to build a Canadian power reactor, the future CANDU, was made in roughly six months, between July 1954 and January 1955, and principally by two men, Bill Bennett, AECL's then president, and Dr. W.B. Lewis, the company's research director. At the same time, it was decided to build a Canadian reactor in conjunction with a major Canadian utility, which turned out to be Ontario Hydro. It was a fortunate decision, and spared Canada for many years from the utility wars that have elsewhere characterized the development of atomic energy. The speed, and the decisiveness, of these events should be underlined, as well as the availability of a talented team of engineers, who sat in W.B. Lewis's extended seminars at Chalk River, and who included John Foster. John later went on to Canadian General Electric, where he participated in the design of the NPD reactor, to Douglas Point, the first true CANDU, to head up the AECL nuclear power division, and finally to be president of AECL. John Foster, throughout his career, has been known as a mainstay of the nuclear industry, not least because of the personal qualities — sincerity and candour — he has brought to his work.

Finally, we have Alvin Weinberg's contribution. Dr. Weinberg's reputation needs no comment from me. He has been for many years one of the most prominent commentators, not only on science, but on the role of science in society. He is best known in atomic circles for the attention he has given to the responsibility of science for its creations, especially nuclear energy, and it is in that line that his paper proceeds.
How It All Began in Canada
The Role of the French Scientists

Bertrand Goldschmidt

For Canada, it really began in February 1940, a few months after the start of the Second World War, the day the French Prime Minister Edouard Daladier gave his consent to the dispatch to Norway of a French secret agent to acquire the worldwide stock of heavy water, a paramount decision in the race toward the chain reaction started a year earlier following the discovery of fission. This discovery had been itself the result of an extraordinary case of scientific collaboration and competition between the main European laboratories prying into the secret of the matter.

First took place two major discoveries: the neutron by James Chadwick in 1932 at the Cavendish Laboratory in Cambridge, then artificial radioactivity in 1934 by Frédéric Joliot and his wife Irène Curie (the daughter of the famous Curie couple) at the Radium Institute in Paris. Soon afterwards Enrico Fermi and his team in Rome found that neutrons were the most efficient particles to produce artificial radioisotopes, still more when slowed down, and they then started the study of the complex mixture of radioactive products obtained by neutron bombardment of uranium. From 1935 to 1938, Otto Hahn, Lise Meitner and Fritz Strassmann at the Kaiser Wilhelm Institute in Berlin investigated thoroughly the problem and were convinced that they had successfully identified the many supposed transuranium radioelements formed in the reaction. However in 1937 and 1938 Irène Curie and the Yugoslav scientist Pavel Savic, once again in Paris, questioned and disproved the Berlin team’s result without finding themselves the solution but putting at last the Germans on the right track. Thus at the end of December 1938 Hahn and Strassmann gave the chemical clue to the enigma leading a few days later to the physical explanation of fission by Lise Meitner, now a refugee in Sweden, and her nephew Otto Frisch who himself gave the crucial experimental proof in early January 1939 at the Niels Bohr Institute in Copenhagen.

The race for the chain reaction was definitely on when in March 1939, at one week’s distance, the two future competing teams in that race - Joliot and his collaborators Hans von Halban and Lew Kowarski at the Collège de France, and Enrico Fermi and Leo Szilard at Columbia University - showed that neutrons were liberated during the fission of uranium.

In April the French announced that
they evaluated the number of neutrons liberated by fission to be $3.5 \pm 0.7$ (it is in reality 2.5). By then Joliot, Halban and Kowarski, who had been joined by the theoretical physicist Francis Perrin, felt they had a rather clear picture of the outline of both a future energy producing machine as well as an explosive device, and decided to patent the features of such a machine and device if only to have a protection in France against patents taken in another country.

Three patents were thus registered in secret between May 1 and May 4 in the name of the Caisse Nationale de la Recherche Scientifique, the main financial backer of the Collège de France research. The four inventors intended to keep only 5% each of the benefits with the remaining 80% going to help scientific research. These patents were the first ever to be taken out on the chain reaction in uranium.

The first two patents concerned energy production and were entitled “Device for energy production” and “Method for stabilizing a device for energy production”. They roughly defined the principles of the main components of our present power reactors: moderator in heterogeneous or homogeneous arrangements, cooling fluid, control rods, protection shield. The third patent called “Method for perfecting explosive charges” was less brilliant from a foresight point of view though it proposed valid solutions for the trigger, the tamper and the rapid obtainment of the critical assembly of a possible explosive device. Finally, nearly a year later, after Alfred Nier’s experimental confirmation in March 1940 of Niels Bohr’s theoretical prediction that uranium 235, the rare isotope of the mixture in natural uranium, was responsible for fission by slow neutrons, the French took out in April 1940 an additional patent on the advantage of using enriched uranium for the chain reaction.

In May 1939, immediately after the taking out of the first patents Joliot decided to deal with the question of uranium procurement, and on May 8 met in Brussels with the leaders of the mining firm Union Minière du Haut-Katanga to explain to them the new importance of uranium, a mainly unused residue of the production of radium of which they were the main world supplier. Thanks to the richest known ore deposit in the Belgian Congo (65% in uranium oxide) the Belgian firm had enjoyed a world monopoly for the production of radium (one gram of which is present for every three tons of uranium) from the mid-twenties to the mid-thirties, and had shared it since the mid-thirties with the Canadian firm Eldorado Gold Mining which exploited an important but less rich deposit discovered in 1930 at Great Bear Lake. A 60%-40% cartel had even been formed in 1938 between the two firms to maintain the price of radium at $25,000 a gram, while the pound of uranium oxide was worth one dollar. Today, thanks to uranium fission, radium has been totally dethroned for medical uses by cobalt-60, which, in a form about 100 to 200 times more concentrated, is sold at a price of about a dollar for the amount equivalent to one gram of radium.

On May 10, Edgar Sengier the chairman of Union Minière, consulted in London with Henry Tizard, principal scientific adviser of the British Government, who was then rather doubtful about the practical future of uranium fission. On May 13 Sengier came to Paris and agreed to set up an exclusive joint venture between his firm and the CNRS, the rightful holder of the patents for the future exploitation of the patents. An agreement was drafted and initialed the same day, but was never concluded because of the advent of the war a few months later and also the delays linked to the complications in setting up a joint enterprise between a French Government
agency and a foreign industrial firm.

The agreement stipulated that a Franco-Belgian syndicate would be responsible for the world exploitation of the French patents and would be set up after the successful outcome of two experiments. The first, with five tons of uranium oxide, the second with fifty tons, were to be supplied by Union Minière. A first consignment of five tons was sent to Paris immediately in June 1939 and later, in April 1940, three more tons were sent before the German invasion. This early procurement by Joliot of eight tons of uranium shows clearly the advance taken at that stage by the Collège de France team on the Columbia University one, as Fermi and Szilard were not able to get hold of their first few tons of uranium oxide before the second half of 1941. These eight tons of uranium oxide were hidden in Morocco during the five years of the German occupation of France and turned out to be an indispensable asset for the start up of the French Commissariat à l'Energie Atomique for fueling its first two low power experimental piles at a time when all the uranium resources available in the Western world had been cornered by the Anglo-American supply agency created following the 1943 Quebec Summit Conference and agreement for atomic collaboration.

The importance of the Franco-Belgian discussions was acknowledged in 1957 in a letter from Sengier to Halban: "Needless to say, these conversations impressed me very much, and drew my most serious attention to the importance of uranium as a potential material for bombs, and the danger of uranium ores falling into the hands of a possible enemy.... That is the reason why I shipped from Africa to America, a stock of rich ore and placed it at the disposal of our Allies".

During the summer and fall of 1939, while the war started in Europe, the French team gave its attention to the choice of a moderator. Their experimental results led them to the conclusion that it was almost impossible to obtain a sustained chain reaction with hydrogen and natural uranium and doubtful that such a reaction could take place in a carbon moderated system. Deuterium in heavy water seemed by far the best moderator.

At the beginning of the war Joliot had obtained full support from Raoul Dautry, the Minister of Armaments, responsible also for the CNRS, for the production of an energy generator as a first step towards a possible submarine engine. As he had convinced the leaders of Union Minière six months earlier, he had easily convinced his minister who did not feel it necessary to seek the opinion of other scientists or, as was the case in England and the United States, to create an official committee for evaluating and supervising the uranium project.

In a report submitted to Dautry at the end of 1939, Joliot specified: "a properly constituted mixture of uranium and heavy water, at this present stage of our knowledge, has all the conditions favorable for the realization of a chain reaction, and consequently for a massive release of energy".

At that time heavy water, discovered in 1932 by the American chemist Harold Urey, worth about half a dollar a gram, had no other uses than research. Nevertheless a Norwegian enterprise, Norsk Hydro, whose majority stock was in the hands of French holders, had begun a small scale production by fractional electrolysis profiting from the low cost of electricity. By 1940, a total of about 200 kilograms had been separated. Joliot suggested to Dautry to try to get a loan of this entire stock valued at $120,000,
specifying in his report that: “if our experiments are successful, that is, if we achieve a massive release of energy, even though our materials are likely to be destroyed, the loss would be negligible compared with the industrial consequences of such a success. If we fail then all the materials will be completely recoverable”.

News from Norway, that the German firm I G Farben was also interested in buying this stock and all future increased production, hastened the departure of a French secret mission under the direction of Jacques Allier, an engineer and banker who was already in contact with Norsk Hydro. His mission, approved by Edouard Daladier, the prime minister, and helped by French military intelligence agents was a complete success. The whole available stock of 185 kilograms was loaned to France for the duration of the hostilities. In addition Norsk Hydro undertook to accelerate future production, all of which would be reserved for France. The mission was back in Paris, with 26 precious cans, on March 9, 1940, exactly one month before the German invasion of Norway.

During this vital mission, French military intelligence, insisting on absolute secrecy, had expressed concern about the original nationality, Austrian and Russian, of Joliot’s two main coworkers Halban and Kowarski who were only naturalized French in 1939. To guarantee, in case there was a security leak, that neither could be suspected, Joliot asked each of them to spend the duration of the mission under surveillance. Their compulsory holidays in two islands, one in the Mediterranean, the other in Brittany, ended with the arrival of the heavy water in the cellars of the Collège de France. From being security risks, Halban and Kowarski became once again indispensable scientists.

The two following months were devoted to setting up the decisive experiment but, in spite of strenuous efforts, this was still not ready when the German advance started to engulf the country. When all hope of stopping this advance was lost, Dautry ordered Joliot, Halban and Kowarski to go to Bordeaux and then to England. Joliot however decided not to abandon his laboratory, where the first European cyclotron was under construction, but on June 18th, Halban, Kowarski and the 26 cans of heavy water boarded a coal-carrying vessel for England. Their mission order read: “They are required to continue in England, in absolute secrecy, the research begun at the Collège de France”. So three months after being under a form of house arrest during the heavy water purchase in Norway, Halban and Kowarski found themselves charged with the great responsibility of trying to preserve France’s position, initially so promising, in the atomic race.

Halban and Kowarski, their heavy water and their project were well received in England by their scientific colleagues - in particular by James Chadwick and John Cockcroft - with an interest that was all the greater because so far the British had turned their attention towards the uranium-235 bomb, following a March 1940 memorandum by Otto Frisch and Rudolph Peierls, and had given little thought to the recovery of useful energy.

The coordinating British committee, code-named the Maud Committee, invited the two French refugee scientists to stay in England and continue their research at the Cavendish Laboratory in Cambridge. In mid-December they were ready to proceed with the crucial experiment planned in Paris. It consisted of the measurement of the neutron density distribution curves inside and outside a 60 cm diameter aluminium sphere rotating at 20 revolutions per min-
Some of the initial research staff of the Montreal Laboratory, 1943 with the Canadians having (C) after their name.
Seated: W.J. Knowles (C), P. Demers (C), J.R. Leicester, H. Seligman, E.D. Courant, E.P. Hincks (C), F.W. Fenning, G.C. Lawrence (C), B. Pontecorvo, G.M. Volkoff (C), A. Weinberg (U.S. Liaison Officer), G. Placzek.

A young Bertrand Goldschmidt.

Lew Kowalski (left) with Frederik Joliot and Hans von Halban.
ute so as to maintain in homogeneous suspension fine uranium oxide powder in the heavy water. A neutron source of 1 gram of radium-beryllium was situated in the center of the sphere which was placed in a tank full of mineral oil so as to recover the heavy water in case of accident. The neutron flux emitted from the sphere was more intense in the presence of uranium oxide in the sphere, and the increase of flux was sufficiently large for the experimenters to be able to affirm, with reasonable certainty, that with much more heavy water and uranium a chain reaction was indeed possible.

Exactly two years after the discovery of Hahn and Strassmann, and six months after leaving France, Halban and Kowarski were the first to demonstrate that it would be possible to produce atomic energy starting with natural uranium. As the report of the Maud Committee of June 1941 stated: “Their results showed a definite indication of a divergent chain process; for each initial neutron, $1.06 \pm 0.02$ were produced in one set of experiments, $1.05 \pm 0.015$ in another set. The system used was relatively small owing to the fact that the amount of heavy water at their disposal was only 180 kilograms and a large loss of neutrons from the surface prevented a divergent chain from developing. They estimate that the critical size of the system which would liberate large amounts of energy would require 3 to 6 tons of heavy water ...”.

The world supply of heavy water had kept its promise and yielded its secret. The full-scale experiment would require fifteen to thirty times more and this was proved exact by the two first natural uranium heavy water piles built and which diverged in the USA at Argonne in May 1944 and in Canada at Chalk River in September 1945.

“Drs. Halban and Kowarski have done all they can with the supplies which they brought to this country”, and advised “that they should be allowed to work in the U.S.” where steps were being taken to produce heavy water industrially mainly on the basis of the favorable results of their experiments. Furthermore Glenn Seaborg, working at the University of Berkeley and utilizing its cyclotron, found that uranium 238 gives, by neutron bombardment a long lived isotope of element 94. This plutonium 239 has the same fissile properties as that of uranium 235. It would be produced in a “machine of the Halban type” which could therefore be used for military purposes.

During the fall of 1941, the whole atomic effort in Great Britain was reorganized under the Department of Scientific and Industrial Research (DSIR) and given the code name of Tube Alloys. Wallace Akers, the Research Director of Imperial Chemical Industries (ICI) was to become its director and the Lord President, Sir John Anderson, the minister in charge. Halban was to be responsible for the work on the chain reaction by slow neutrons.

In early 1942, Akers led a mission to the USA of the main leaders of Tube Alloys. Halban was hoping to convince the Americans to host his Cambridge team as an independent British unit at or near the University of Chicago where all the American work had just been concentrated under the code name of Metallurgical Laboratory and under the leadership of Arthur Compton. This project, soon to comprise more than a hundred scientists, was going ahead at full speed in two main directions: the graphite uranium system developed by Fermi and his team, and the plutonium isolation and extraction problem by Seaborg’s team.

The Maud Committee concluded that

At that time the Cambridge group
comprised only ten researchers: five of German and Austrian origin, three Frenchmen (Jules Guéron, a physical chemist had recently joined Halban and Kowarski) and only two English scientists. Washington, absolutely opposed to their settling in the U.S. as an independent British unit, offered Halban a position in the Chicago group as responsible for the heavy water work, assisted perhaps by either Kowarski or Guéron but only if an equivalent scientist could not be found in the U.S.

Halban refused to sever his British links. He wanted to lead his own team and still had hopes to be the first to achieve a sustained chain reaction. He convinced the British authorities that their position was strong thanks to the Cambridge work and future rights on the French patents. In exchange for the help received in England, he had undertaken to obtain from the French after the war a transfer to the British government of the rights, outside France and the French empire, of the initial French patents considered by him as master patents for any future industrial development.

He then proposed, with the approval of Washington, that his team should be transferred to Canada as the nucleus of a much larger Anglo-Canadian outfit. Churchill agreed and the final decision to launch such an enterprise was taken in London on October 12, 1942, at a meeting between Anderson and Clarence Decatur Howe the Canadian Minister of Munitions and Supply. C.D. Howe and Dean Jack Mackenzie, acting President of the National Research Council (NRC), were to be during the war the two Canadians responsible for this first scientific multinational project ever created. Its international character was reinforced by the French nationality of Halban, its first director, and the European origin of several of its division and section leaders. Furthermore, it was to be depend-}

ent on the U.S. for technical help and basic materials such as heavy water and pure uranium metal.

Final arrangements, and the decision to locate the laboratory in Montreal, were the results of a meeting on October 30 in Ottawa where I had just been summoned by Halban and where I visited with great interest the graphite uranium experiment undertaken at the NRC by George Laurence. I was a specialist in radio chemistry, having been recruited by Marie Curie at the Radium Institute in 1933, the year before her death. Having joined the Free French Forces in the States in early 1942, I had been seconded to the DSIR which had sent me to Chicago during the summer of 1942 where I participated, under Seaborg, in the identification of some of the main long-lived fission products as well as to the isolation of the first quarter of a milligram of plutonium. I was now to be a section leader with the Montreal project in the chemistry division under Fritz Paneth, a renowned Austrian radiochemist.

Thereafter several weeks were spent in Montreal looking in vain for a suitable location for the laboratory, until my chance encounter with a French refugee biologist, professor at the Université de Montréal, Henri Laugier, the former director of the CNRS, who as such had been involved in the taking out of the French patents. He suggested to me the unoccupied wings of the University which, because of lack of funds, had not yet been equipped for the medical school. This was to be an unexpected French contribution to the new project.

Agreement by Halban, Ottawa and the University followed and the plans for the new laboratory were expeditiously prepared by Ernest Cormier, the architect of the University. The premises were ready
for occupancy in March 1943 when the Cambridge team and many newly recruited British technicians, scientists and engineers reached the calm side of the Atlantic.

Kowarski was not to be among the pioneers of the Montreal laboratory. His personal relations with Halban had deteriorated with the passing of time and the politicization of the venture, and another French refugee physicist of worldwide repute, Pierre Auger, had accepted the direction of the physics division of the project.

A far greater turbulence was going to rock the Anglo-Canadian outfit before it had taken off. On December 2nd, 1942, the first self-sustaining chain reaction had taken place in the pile of uranium and graphite Fermi built in Chicago. Three weeks later, Roosevelt, briefed on the great technical advance of the American project, recently put under army responsibility and control, decided to limit drastically the exchanges with the British, a first American move on the road to non-proliferation. The allocation of the first tons of heavy water produced in the U.S. and, with complete American funding, at Trail, B.C. was to be decided at a later date, meanwhile the Montreal laboratory would have to limit itself to fundamental research on the utilization of heavy water. In a way, the future Anglo-Canadian team was being asked to furnish its gray matter to the American scientists and industry in case they should decide to build a first heavy water pile.

The breakdown in relations became complete in March, but not before a last minute successful "spying" mission to Chicago by Auger and myself, in early February, where we obtained from our American colleagues the most recent technical information on Fermi’s pile and on plutonium extraction, as well as four micro-

grams of plutonium and a sample of long-lived fission products from the material I had dealt with in Seaborg’s laboratory.

For the team just settled in Montreal, and expecting to profit from being at an easy distance from Chicago’s Metallurgical Laboratory and many of the American resources, Roosevelt’s decision was a terrible blow. It was further amplified when it was learned that, unknown to the British, the Canadians had sold out in advance to the American project all their uranium production until 1946. The roughly one hundred technicians and scientists of the Anglo-Canadian team were thus denied access to uranium, heavy water, plutonium and American help and so found themselves condemned to inaction almost before they had begun to work. Their demoralization was to be further increased by the difficult character, the authoritarian manners and the poor managerial abilities of Halban, their leader.

Fortunately Churchill, who did not know that Roosevelt himself had been responsible for the break, fought back and, at each summit meeting with him in 1943 (Casablanca in February, Washington in May, and Quebec in August) insisted that collaboration should be complete in this field as in all others independently of the ratio of the contributions of the two Allies. Roosevelt agreed each time but without giving instructions to lower levels. He could not keep up this double attitude for long however and agreed at Quebec that collaboration should be resumed. However it was specified in the Quebec Agreement that, in the industrial field, exchange of information should be limited to what was necessary for the pursuit of the war. Churchill agreed on this clause to counter the American accusations that the British were overly interested in patent rights and postwar commercial benefits since they had
given ICI an excessive part in running Tube Alloys.

The renewal of this somewhat limited collaboration was helped by the British decision to send Chadwick to Washington to maintain liaison with the American project and its head General Leslie Groves. Early in January 1944, a meeting presided over by these two leaders took place in Chicago to re-establish the links between the Metallurgical and Montreal Laboratories. The meeting was inconclusive and marked by dissension as Groves insisted that plutonium chemistry and extraction technology were to be excluded from a renewed collaboration.

Furthermore it was revealed that a first heavy water pile was under construction at Argonne, Illinois and divergence due to be achieved in a few months (it diverged in May 1944). This task had been given by Compton to the Chicago pioneers of the first graphite pile when they had been deprived of the responsibility of planning and building the large graphite-producing piles which had been transferred, in mid-1943, to the private industrial firm DuPont. So the "raison d'être" of the Anglo-Canadian project had been, unknown to Montreal, entrusted to the Chicago scientists like a toy to soothe their frustration. It was only then that Halban realized that, contrary to his hopes since 1940, he would not be the first to obtain the heavy water chain reaction. He was to be replaced in May 1944 by the British physicist John Cockcroft as head of the Laboratory and was to leave atomic energy one year later after a political drama caused by an untimely visit by him to Joliot in Paris after the Liberation to discuss the patent rights problems. This visit had been authorized by Anderson but vetoed by Washington.

After the January 1944 Chicago meeting, Chadwick continued to try to reverse the American opposition to the building of a large heavy water reactor in Canada. In February, Groves, Mackenzie and he entrusted to Major Arthur Peterson, Groves' representative in Chicago, a study on the question. It came again to a negative conclusion because a Canadian pile would be of no help in winning the war. However this report emphasized the qualities of heavy water piles: "the advantage of heavy water piles may be so marked, and their post-war applications may be so important and far reaching, that their development cannot be wholly neglected".

Chadwick was in Los Alamos, the secret bomb centre in New Mexico, when he got the report. Groves joined him on March 27th still opposed to the Canadian pile. At the end of the day Chadwick succeeded in getting him to change his mind, abandoning the idea of a pile whose plutonium production could have had a real military importance, and accepting instead a ten thousand kilowatt one, thirty times greater than the first one which was about to be completed in Chicago.

It was on that evening that Halban and Kowarski's mission finally took shape. After the long and disappointing delays, the Montreal Group was to build an industrial-size pile, the future NRX, a genuine step toward atomic energy. Halban's main objective was achieved but by an irony of fate, he was not going to bring it to fruition. Rather it was to be Chadwick and Cockcroft, the very scientists who had received him and Kowarski so cordially upon their arrival in England nearly three years earlier.

However, in July 1944, Cockcroft took the wise decision to tackle first a much smaller unit of near zero power and entrusted its construction to Kowarski whom he had called back from Cambridge. It was
to be ZEEP which diverged at Chalk River on September 5, 1945. Justice had been done. One of the members of the French nuclear “tandem” had the honour of building, if not the first heavy water pile in the world, at least the first atomic pile outside the United States.

Chadwick succeeded also in partially solving the plutonium problem. Renouncing “until further notice” any American know-how on the new element, he obtained, at a meeting in Chicago on June 8, 1944 (two days after the Normandy landings) with Groves, Cockcroft and Mackenzie, the American promise of a few irradiated uranium rods from the second American pile (the Oak Ridge one). These contained a few milligrams of plutonium and enabled the Montreal group to work out independently extraction and chemical properties of the new element.

Taking advantage of my experience with Seaborg’s group in 1942, I was able, with a small team of Canadian chemists, to establish the outline of the first solvent extraction process for plutonium in 1945, thus demonstrating for the first time the relative ineffectiveness of the policy of secrecy in such a specifically sensitive field as the reprocessing of irradiated fuels and paradoxically between close allies during the war. The so-called Trigly process led to the production, in Chalk River in the late forties and early fifties, of about fifteen kilograms of plutonium which would probably have allowed Canada, had it not been the first country to freely renounce the bomb, to be the third military nuclear nation in addition to becoming the world champion of heavy water power reactors.
Personal Reminiscences
Leslie Cook

Figure 1 reminds us that the McGill physics building was the original home of modern nuclear physics and of radiochemistry. It was Rutherford's world leadership reputation that brought Otto Hahn to McGill in 1905. This slide shows Rutherford, Eve, Hahn, Boyle, and other associates on the front steps.

Hahn, in his autobiography, speaks glowingly of Rutherford as a man and as a scientist - clearly a man to emulate in his own career. Hahn and Rutherford were both eventually awarded Nobel prizes in chemistry - the first alumni of any Canadian university to be so honored. But when they both left in the next two years, radiochemistry left McGill with them, not to come back again for 45 years until Leo Yaffe (with us today) brought it back again from Chalk River. The Chemistry and Physics Departments had always got along very well at McGill, according to Winkler, by having nothing to do with each other. Leo was to be a bridge between them - which both saw as a potential disaster. The bridge was put in place only by the determined insistence of Steacie, who had come to understand its importance as a result of his association with the Chalk River project. And it has been an enormous success.

Meanwhile Hahn developed his own school of radiochemistry in Berlin and for eight years continued to develop his painstaking and perceptive elucidation of the natural radioactive chains, and of the chemical techniques for minute trace materials.

Kaiser Wilhelm Institute, Berlin 1937/38

But Hahn's personal world was soon to be bruised by World War II. In 1912 the Kaiser came to open the first of his new research institutes in which Hahn was to work (Fig. 2) with its ominous military helmet tower (Fig. 3). Within two years Hahn was cheerfully off to war games with his guitar and machine gun (Fig. 4). But things soon turned serious and ugly. Hahn found himself in the front line trenches at Ypres introducing poison gas warfare to the world (Fig. 5).

By 1933 Hahn was lecturing at Cornell and Toronto when such disturbing news reached him that he rushed home. He was devastated to find that the sword of Nazism was already cutting the scientific community into bitter factions; that his long time physics partner, Lisa Meitner, had been dismissed from the university; that Fritz Haber, director of the adjacent institute, had been replaced by a Nazi incumbent; that he had a rival in his own institute who was openly talking of his expectation of
Fig. 1 On the steps of McGill University in 1905. Showing Rutherford (front right), Eve (second row, third from right), Hahn (second row, second from right) and Boyle (back row, right).

Fig. 2 Kaiser Wilhelm visiting the institute that bore his name in 1912.

Fig. 3 The Kaiser Wilhelm Institute, 1912.

Fig. 4 Otto Hahn off to war games in 1914 with guitar.
soon taking over from Hahn as director; that several of his own physicists and chemists were active Nazi party members, complete with a Gauleiter and brown uniforms, reporting on anything and everything.

It was a frightening situation which got rapidly worse as he defied the Nazi machinery. Fear of retribution on his associates, his family and himself began to dominate his mind as he and Meitner struggled to continue their research. In his autobiography Hahn writes that “he was under extremestress”, “nervous to the point of exhaustion”, and “subject to sudden and embarrassing weeping spells in conversation”. No wonder that he and Meitner began to slip in their judgement of critical technical clues.

In retrospect Hahn was to write of their work at this time, “all our ideas turned out to be wrong”, “our reasoning was admittedly faulty”, and “how could we have remained in the dark so long”. The political and personal pressures had taken their toll on performance, as they always do.

I wish there was time today to tell the full story in all its fascinating technical and personal detail - but I can only touch on a few high spots.

Contrary to what seems to be science history dogma, Hahn and Strassmann never mentioned fission, nor ever claimed to have discovered it. What they actually did was confirm a unique and new discovery made by Irène Curie and Pavel Savic in Paris, that the irradiation of uranium with neutrons yielded a radioactive substance that chemically resembled lanthanum more than any other known element. Even so, Hahn was very dubious that these substances were actually lanthanum and barium as any reader can discover by reading his published paper.

Hahn had kept all these strange results secret from his colleagues in the institute, trusting them only by mail to Meitner who was in refuge in Sweden. Thus it was Meitner, after sharing them with her nephew Frisch, who guessed that the results had to be interpreted as fission. They quickly sought the recoil pulses, found them, and announced their discovery to the world - much to the anger of Hahn’s colleague physicists in the institute who felt they had been unjustly cut out of sharing in this great discovery.

Even so, this interpretation was not original with Meitner and Frisch. In 1933 Fermi reported finding five radioactive substances on irradiating uranium with neutrons, and with the support of some primitive chemistry suggested they might be transuranics - which of course was what he expected them to be. Very quickly a German woman chemist published a detailed critique of this and suggested fission instead as a conceivable alternative explanation. For anyone with the equipment (which Ida Noddack did not have), it would have taken half an hour’s work to check this idea out. But Hahn and Meitner rejected it out of hand as “sheer fantasy”, coming from someone not even in the nuclear field, and totally ignorant of nuclear physics. Moreover a proper appreciation of Noddack’s technical comments was swamped by bitterness over some “reprehensible behaviour” made use of in a quite unpleasant manner clearly connected with Nazi pressures. ("The Griffin", Arnold Kramish, Houghton Mifflin publishers - 1986, p.236 - Walter Noddack, Ida Tacke). Whatever this was, it left a deep legacy of bitterness in
Hahn. When her suggestion turned out four and a half years later to be correct it was a bitter pill that Hahn never would swallow by admitting that her suggestion had played any role in their final thinking. A careful reading of the Hahn-Meitner correspondence however leaves little room for doubt that it did.

Hahn himself, later on, publicly and persistently belittled Curie and Savic's contribution, and dismissed Noddack's totally on the superficial grounds that she did not do anything about it experimentally. These were both most un-Rutherford-like actions. It is a bitter object lesson in the interaction of politics and science.

Some of us were greatly privileged, a month ago in Washington, to hear Pavel Savic tell his story for the first time ever before an English speaking audience. He is the only living survivor of the seven people whose experiments and thinking led directly to the discovery.

I cannot help observing also that the three women scientists - Curie, Noddack and Meitner - were the contributors of all of the intuitive, original and correct ideas that repeatedly guided the experiment back onto the right track. Only they apparently were able to think big enough to encompass the situation, and this despite the fact that they were anything but friends.

Fig. 6 shows this unhappy institute family going through the motions of their annual Christmas party in 1936.

The Nuclear Project in Canada, 1944-46

And now we return to Montreal six years and World War II later. The British were about to cancel the Montreal project, when Chadwick, at the last moment, thought up the idea of NRX and sold it to a reluctant General Groves, who because he controlled all uranium and heavy water supplies - even in Canada - and all information flow, held the trump hand. If anyone must be spoken of as the Father of Chalk River, that honour belongs to Chadwick.

Actually NRX was the pilot of a second generation of plutonium production reactors, and DuPont engineers came to Chalk River to learn all they could before designing the Savannah River reactors, now 30 years old and much in the news. As we all know, NRX was a tremendous success, and its key conceptual designers, Newell and Ginns, deserved the highest recognition from the Canadian engineering profession - which they never got as far as I know.

As part of this NRX deal Chadwick had insisted on some agreement about the touchy subject of plutonium processing. Groves finally agreed to provide a few kilograms of weakly irradiated uranium - nothing more. But Bertrand Goldschmidt had worked with Seaborg and knew what to do. Soon, as a result of our work in the Université de Montréal wing, Bertrand Goldschmidt recommended two processes. The one was ideally suited to large-scale continuous processing, and the other uniquely suited only to small-scale intermittent batch processing. Steacie, now part-time associate director of the project, opted decisively for the small-scale batch process, reflecting the stirrings of ambiguity and indecision in domestic post war nuclear politics. The British immediately asked to send over a separate group of their own to develop the large-scale process for their own use at Windscale, and Bob Spence came out with that group and did it.

Meanwhile the notorious Trigly process was installed at Chalk River, over my
Fig. 5 Otto Hahn in the trenches of Ypres, First World War.

Fig. 6 A not too cheerful Christmas party at the Kaiser Wilhelm Institute, 1936. Showing from the right Fritz Strassmann, Les Cook and Erbacker (the Institute's Nazi gauleiter).
objections as to what would happen when the storage tank got filled with the messy waste. Steacie said that would be someone else’s problem - not his. And indeed that is exactly what happened as some of you well know.

Anyway the Trigly plant supplied plutonium for NRX fuel for years. When the U.S. agreed to supply enriched fuel, the plant was shut down and the technology forgotten. However I see that France is now recovering and re-using plutonium from its nuclear power plants. Perhaps the time is not that far off when Canada will follow this lead and use its CANDU plutonium.

The plutonium nitrate from the Trigly plant had to be converted into metal fuel elements for use in NRX. Since we had no metallurgy branch we had to take it under our wing in chemistry. The success of the effort was entirely due to the good fortune of finding just the right young engineer newly minted at the University of Toronto to do it. John Runnalls is with us here today as President of the CNA. He discovered that molten aluminum would directly reduce at least two plutonium compounds, and thus enable him to make the plutonium-aluminum alloy in one step. He then designed a building, and all the necessary equipment to execute this very dangerous operation. The operation was run successfully for many years, supplying NRX with the essential enrichment to enable fuel testing for future reactors to be done. All this before he moved on to several new careers in Ottawa and the University of Toronto.

NRX was not only ideal for plutonium production, but also for a variety of other radioactive materials for research such as carbon-14, iodine, phosphorus, sulphur, tritium, etc. Bill Stevens, also with us today, wanted to use carbon-14 in his own work and soon found he could make it with very high specific activity, and with minimal effort due, of course, to the high flux of NRX. Soon we had the idea of getting into the supply business and augmenting the AECL budget by selling radioactive materials all over the world—for no one could compete with us. But we had to work fast because the U.S. would clearly not let Canada have the world champion flux for many years. The money was soon budgeted and Bill designed a new building and equipped it for the job in double-quick time. I let W.B. Lewis know we were about to start advertising and shipment. Within days C.J. McKenzie arrived in my office alone and asked to see the lab. All unsuspecting we gave him the grand tour. I wondered at his silent demeanor on the road back until he suddenly turned to me and said, "You can’t do this". Dumbfounded, I asked why? "Because the U.S. are not exporting and until they do, we can’t." It was a bitter lesson in politics. And the ban continued until it was broken by Henry Seligman’s "fait accompli" at Harwell. A professor from Paris wrote to him asking for a few thousand counts of phosphorus or sulphur — I forget which now. Henry wrapped them up in a small match box and dropped them in the mail. Shortly, he received a phone call from some nameless bureaucrat in London telling him he couldn’t do things like that. Henry told him that he, Henry, did not work for him but for Sir John Cockcroft, and to direct complaints to him. He slammed down the phone, he told me, for good measure. Nothing more was heard of it — except that the first isotopes catalogue from the U.S. soon appeared. But the golden opportunity for us in Canada was gone. The high flux MTR at Idaho was now in operation.

Few people today remember that the AECL Board actually committed itself to the construction of two Douglas Point size reactors for Ontario Hydro in 1954 — for
1957 operation. This was evidently because they realized that Rickover's Shippingport reactor was close to operation and both the first BWR and the first PWR of Douglas Point size were close to commitment. I wish Bill Bennett could have been here today because this occurred during his Presidency and he could have told us more about it. I always assumed that the Board realized that the marketing horse race was already beginning and they felt they had to be a part of it then and there. When W.B. Lewis surprised me with this I was stunned. My question was whether the government was deadly serious in this. He assured me that it was and asked for help in how we could plan to meet a PERT diagram that he produced.

To make a long story short, within weeks we had inaugurated a long overdue physical metallurgy effort, reorganized our chemical forces, and were off and running. After a few months I began to wonder why I was not being pestered for progress reports. Lewis was completely silent. Now really worried, I cornered Bill Bennett to ask whether these two reactors would be built or not. "I am afraid not" he said "someone forgot to tell C.D. Howe".

The result was predictable. Douglas Point came on line 13 years later instead of three or four. By this time both Dresden I and Yankee Rowe had over six years of operating experience and two dozen full size follow-on units were under construction. As the Board undoubtedly feared, ten crucial years had just been lost at the very beginning of the race. With due apology to Shakespeare, there are tides in the affairs of men which must be taken at the flood if they are to lead on to fortune.

During this period, when NRX was at its peak as a world-leading technology, Admiral Rickover installed equipment in NRX to test fuel elements for his submarine engines. When the accident in December 1952 put NRX out of commission, Rickover was quick to offer help. This arrived shortly as a group of naval ratings under the command of a young officer, one Lieut. Jimmy Carter! I have often wondered if his allergy to nuclear power stemmed from his experience in the basement of NRX.

Speaking of Presidents, no doubt you have all noted that President Bush has appointed a new science advisor, Allan Bromley of Westmeath in the Ottawa Valley, Queen's University and, of course, AECL at Chalk River. Whatever AECL's status may be here in Ottawa, it now has a representative right at the top in Washington.

I want to say now a few words about the impact that the Chalk River project has had both on Canada and on the world, and I can think of no better way to do this than to remind you briefly of the outstanding contributions that have been made in the careers of Chalk River chemistry alumni.

No less than six Chalk River chemistry alumni have taken their places in Canadian universities. John Runnalls, in addition to his post at the University of Toronto, is known round the world as Mr. Uranium of Canada and also has served on the Board of Ontario Hydro. Leo Yaffe, as I mentioned before, founded his own school of radiochemistry at McGill. He successfully bridged the chemistry and physics departments and his reward was chairmanship of the chemistry department and vice principal of the university — a worthy successor to Rutherford and Hahn even if 45 years late!

Lou Siminovitch, whom many of you know as a world renowned biochemist, began his career at Chalk River before going
to the Pasteur Institute in Paris. He later became professor at Toronto, Department Chairman, and is now Director of Research at the new research institute at Mt. Sinai Hospital. Bob Betts left his position at Chalk River as chemistry research head to become head of the department of chemistry at the University of Manitoba. Bob Jervis went to the University of Toronto to bring the techniques of radiochemical analysis to forensic chemistry, founded his own radiochemistry school there, and became Dean of Research. Maurice Lister also became professor of chemistry at Toronto. The late Henry Heal became professor of chemistry successively in the West Indies, at the University of British Columbia, and finally at Queen’s University Belfast. John Spinks returned to the University of Saskatchewan and founded his own radiochemical school there. And of course Harry Thode, supported from Chalk River, built his own radiochemistry school around his own reactor at McMaster.

The late Mac Lounsbury designed, built and operated successfully the very first mass spectrometer outside of the United States that could analyze the isotopic composition of transuranic elements, an accomplishment that gained an accolade from Chadwick himself. Bernard Harvey published from Chalk River the very first scientific paper on the chemistry of plutonium and then moved on to a continuing career in the chemistry of other transuranic elements at the Radiation Laboratory in Berkeley — elements which we could not make at Chalk River because, at that time, we did not have the necessary accelerators and had no prospects or plans to build any.

Then we have a long list of Bldg. 107 chemistry alumni from Chalk River who have made their mark in the U.K. Bob Spence and Lewis Roberts both succeeded Cockcroft as directors of Harwell. Frank Morgan became Vice Director of the Aldermaston weapons establishment. Nick Miller joined the staff at Edinburgh University. And Henry Seligman of course became director of isotope manufacture and distribution at Wantage.

And then there is a long list of people who have played key roles at Chalk River itself. Eric Perryman had rapidly built up a physical metallurgy effort from zero to meet the demands of the aborted attempt to build two reactors for 1957. The effort played an important role later anyway; Bill Campbell, also with us today, headed our chemical engineering group for many years and eventually directed the chemistry division. Bob Robertson who supervised the water chemistry in Rickover’s test units in NRX went on to become a water chemistry expert for reactors. Alex Eastwood headed chemistry research for many years; John Davies became internationally known for his work on solid state channeling. I could go on and on.

All the above are worthy successors in Canada to Rutherford and Hahn, a whole generation later. And not a bad record for good old Building 107.

Bob Bothwell has given us a penetrating history in “Nucleus - The History of Atomic Energy of Canada Limited” the evolution of the Chalk River project at the policy, strategy and political level. But no one has yet written the history of accomplishment of the scientists at the working level. This, after all, was where all the real action and real accomplishments took place. I hope someone will be motivated to do this soon — before it is too late.
I have been asked to talk about the effects of the discovery of fission on the development of physics in Canada. This involves Chalk River and the universities, and three government agencies that have played a key role in shaping university physics: AECB, the Atomic Energy Control Board; NRC, the National Research Council; and NSERC, the Natural Sciences & Engineering Research Council.

Initially the discovery of fission had little impact in Canada. Nuclear physics research was at a low ebb after the Great Depression. There were no particle accelerators; McGill and Queen's were ready to proceed with accelerator projects just as war broke out, but both had to be shelved.

The first notable fission-related experiment (Fig. 1) was done at NRC by George Laurence, who was head of the Radiology Section. He did the experiment in great secrecy, in his spare time, and assisted only by Bern Sargent from Queen's during summer vacations.

It was the first attempt anywhere to see whether a chain reaction would go with natural uranium using carbon as a moderator. Progress was slow, however, and the result inconclusive - it was clear that purer materials would be required. By the time the experiment was set aside, in the summer of 1942, Fermi was on his ninth uranium-graphite lattice, which gave a k-infinity of 1.007; the Stagg Field pile went critical on December 2, 1942.

Laurence's initiative was not without value. Contacts had been established with classified research in both the US and UK, and when Laurence and Sargent moved to the Montreal Laboratory, they brought valuable experience with them.

The Montreal Lab and the early days at Chalk River have been covered by the previous speakers. I want to draw your attention to the Atomic Energy Control Act which established a five-member Board with wide-ranging powers including authority to make grants in aid of research. The NRC was also a granting agency until 1978 when this function was taken over by NSERC, and it operated many research labs of its own, including the Chalk River project until the foundation of AECL.

In 1948 C.J. Mackenzie became President of the AECB while remaining President of NRC. He was a man deeply convinced of the value of basic science and he wanted Canadian universities to contribute in the field of nuclear research. Accordingly he encouraged universities to apply
to the AECB for grants very much larger than those NRC was in the habit of giving, in order to set up major nuclear facilities. Fig. 2 shows the result of this policy over the next decade. In fact it was NRC that contributed to the capital funding for the McMaster reactor, but AECB funded its operation.

Meanwhile, at Chalk River, MacKenzie's original commitment to basic research had been reinforced by Cockcroft and Lewis. Both NRX and NRU had excellent research facilities; indeed NRX was, in its day, the best research reactor in the world. The Van de Graaff was planned in the very early days as part of the commitment to a broad range of nuclear science. It was the ancestor of a series of accelerators that have taken over an increasing fraction of the nuclear physics research at Chalk River. Indeed, today, the principal physics work at NRU is the study of condensed matter by slow neutron scattering.

Figure 3 shows NRX in its heyday with a mass of experiments at the end of the beam tubes. One of the most celebrated (Fig. 4) was John Robson's study of the radioactive decay of the free neutron. It was published in 1950 and is described in one authoritative text as "an experiment of exemplary care and ingenuity". Fig. 5 shows another important Chalk River contribution, a triple-axis neutron spectrometer at NRU in 1958, and its inventor Bert Brockhouse; it is now the standard technique used worldwide for the study of condensed matter by neutron inelastic scattering.

Figure 6 shows the world's first Tandem Accelerator installed at Chalk River in 1959 and (Fig. 7) three distinguished alumni - Harry Gove, Ted Litherland and Al Bromley, the latter having recently been appointed U.S. Presidential Science Adviser.

This Tandem and its successors would provide for an evolutionary nuclear physics program. But what would the next step be beyond NRU? Dr. Lewis's answer was the Intense Neutron Generator, ING, designed to out-perform NRU by almost two orders of magnitude.

Figure 8 shows what ING might have looked like. A mile-long linac would have accelerated protons to an energy of 1000 MeV and delivered them to a liquid lead-bismuth target surrounded by a heavy water moderator. To get the required thermal neutron flux of $10^{16} \text{n cm}^{-2} \text{s}^{-1}$ a proton current of 65 mA would have been needed and achieving this would have been a considerable challenge even today.

ING would also have been a test bed for Dr. Lewis' concept of electronuclear breeding — the injection of accelerator-produced neutrons into a Th-232/U-233 thermal reactor breeding cycle. In this context it is interesting that the decision to go for ING was taken, in 1964, after reviewing a number of alternative programs; two of the unsuccessful proposals were fusion and fast reactors.

No doubt there were many reasons why, in September 1968, the government instructed AECL to terminate the ING project, but one of Chalk River's serious mistakes was its failure to bring the outside community, particularly the universities, into the planning process. The lesson was learnt too late for ING, but ever since then serious attempts have been made to enlist university cooperation, with some successes which I will come to.

Perhaps Chalk River had not appreciated the growth in university-based nuclear physics. Starting in 1961 NRC had followed AECB's lead and funded the con-
Fig. 1 Cutaway section of the Laurence "pile", 1961-62

![Diagram of the Laurence "pile"](image)

<table>
<thead>
<tr>
<th>University</th>
<th>Facility</th>
<th>Operated</th>
</tr>
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<tbody>
<tr>
<td>McGill</td>
<td>Cyclotron</td>
<td>1949</td>
</tr>
<tr>
<td>Queen's</td>
<td>Synchrotron</td>
<td>1950</td>
</tr>
<tr>
<td>UBC</td>
<td>Van de Graaff</td>
<td>1951</td>
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<tr>
<td>Saskatchewan</td>
<td>Betatron</td>
<td>1952</td>
</tr>
<tr>
<td>McMaster</td>
<td>Swimming Pool</td>
<td>1959</td>
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</tbody>
</table>

Fig. 2 Growth of nuclear physics facilities 1949-59.

![Image of the NRX reactor at Chalk River](image)

Fig. 3 Early view of the NRX reactor at Chalk River (1950).
Fig. 4 John Robson with equipment used in the first precise measurement of the radioactive decay of the neutron, 1950.

Fig. 5 The early triple-axis spectrometer installed at NRU in Chalk River, 1958 and its inventor Bert Brockhouse.

Fig. 6 The world's first tandem accelerator installed at Chalk River in 1959.
struction of seven major nuclear installations in a five year period.

Figure 9 lists the twelve university installations operating at the end of the sixties and the dates they had started to operate. By 1969 they were consuming 4 M$ a year in operating funds, equal to all the rest of the university physics research. Of this 4 M$, 60% came from AECA and 40% from NRC. Obviously this was too large an investment in low energy nuclear physics and already in 1966 NRC had decided to call a moratorium on the construction of such facilities.

This did not apply to TRIUMF, in Vancouver, which was funded through the AECA in 1968. It was enthusiastically supported by George Laurence, by then President of AECA, who convinced Treasury Board. And the decision not to fund ING freed up 1.5 M$ to help get things going.

TRIUMF was a joint initiative by three universities, UBC, Simon Fraser and Victoria, who were soon joined by the University of Alberta. TRIUMF was the third Meson Factory to be built in the world and was much larger than the existing Canadian accelerators, as can be seen from Fig. 10. Figure 11 shows the extensive facilities now in place. As well as pure science experiments in the proton and meson halls, a pion beam is available for cancer therapy, and radio-isotopes are produced on a large scale, mostly in a small dedicated cyclotron.

Figure 12 compares TRIUMF and ING. A small fraction of the ING beam would have powered a handsome meson factory, and the ING plans included such a facility. By the time it was fully operational in 1977 TRIUMF had cost about 40 million (1967) dollars which is equivalent to 160 million in today's dollars. Its current budget is about 30 M$ a year, which is nearly as much as Canada is spending in all other areas of basic physics research.

Regarding other current activities, I will come back to NRU at the end of my talk. The nuclear physics program at Chalk River is now based on the Tandem Accelerator Superconducting Cyclotron, TASSC (Fig. 13). The Superconducting Cyclotron — a new idea for a compact and powerful accelerator — was designed and built by the Chalk River Accelerator Physics Branch. The branch was born during the ING days and has undertaken a variety of accelerator projects since then, the cyclotron being its main contribution to physics research.

TASSC can accelerate most elements in the periodic table to an energy of 10 MeV per nucleon or more. It is the largest nuclear physics facility in Canada, after TRIUMF. Much of the research involves people from other institutions, especially from Canadian universities, and one of the principal instruments (Fig. 14) was funded jointly by NSERC and Chalk River at a cost of some 4 M$. It is called the 8π spectrometer and is used for sorting out the complex gamma ray cascades that follow heavy ion reactions.

Of the twelve university installations I showed on an earlier slide, only one has survived as a pure nuclear physics lab, the Saskatchewan electron linac. It has recently been upgraded with a storage ring which converts what was originally a pulsed output into a continuous current of electrons with energies up to 300 MeV. This CW feature is very valuable — it makes coincidence experiments possible — and the Saskatchewan Accelerator is currently unique and much in demand by scientists from inside Canada and abroad.

It was a very economical upgrade. The ring was squeezed into the existing building by the ingenious expedient of
Fig. 7 Three distinguished alumni of the tandem accelerator at the controls. From the right Ted Litherland, Al Bromley and Harry Gove.

Fig. 8 The proposed intense neutron generator facility, ING (1967).

Fig. 9 Major physics facilities at Canadian universities at the end of the sixties, with the dates they had started to operate.

<table>
<thead>
<tr>
<th>Facility</th>
<th>Year 1</th>
<th>Year 2</th>
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<tbody>
<tr>
<td>McGill Cyclotron</td>
<td>1949</td>
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<tr>
<td>Saskatchewan Linac</td>
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<td>1965</td>
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<td>UBC Van de Graaff</td>
<td>1951</td>
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<td>Queen’s Van de Graaff</td>
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<td>1966</td>
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<tr>
<td>McMaster Reactor</td>
<td>1959</td>
<td></td>
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<tr>
<td>Toronto Linac</td>
<td></td>
<td>1966</td>
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<tr>
<td>Alberta Van de Graaff</td>
<td>1962</td>
<td></td>
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<tr>
<td>Ottawa-Carleton Dynamitron</td>
<td></td>
<td>1967</td>
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<tr>
<td>Laval Van de Graaff</td>
<td>1963</td>
<td></td>
</tr>
<tr>
<td>Montreal Tandem</td>
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<td>1967</td>
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<tr>
<td>Manitoba Cyclotron</td>
<td>1965</td>
<td></td>
</tr>
<tr>
<td>McMaster Tandem</td>
<td></td>
<td>1969</td>
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Fig. 10 The lower face of the TRIUMF magnet during construction, 1971.

Fig. 11 The layout of the TRIUMF meson facility as it exists today.

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**TRIUMF** | **ING**
---|---
Proton Energy | 500 MeV | 1000 MeV
Current | 100 $\mu$A | 65 mA
Estimated Cost (1967) | 23 M$ | 128 M$

Fig. 12 Comparison of the relevant parameters of the TRIUMF and ING accelerators.
hanging it from the ceiling. Figure 15 shows the layout — the original linac, the 180-degree injection line into the racetrack-shaped ring, and the extraction line to the target areas. Figure 16 shows the low-energy end of the linac on the right and the ring above it on the left. Figure 17 shows the 180-degree injection line and the ring in the background.

What of the future? There are at present two major physics projects before the Canadian government and both of them are nuclear (subatomic) - the KAON Factory and the Sudbury Neutrino Observatory.

The KAON Factory (Fig. 18) would use the present TRIUMF cyclotron to inject 100 µA of 500 MeV protons into a complex accelerator system that would increase their energy to 30 GeV. In the context of modern high-energy accelerators this is not a high energy, but the current is very high. As expressed by its proponent, Erich Vogt, it would be at the intensity frontier — a factory making intense beams of secondary particles, notably kaons, antiprotons and neutrinos. Kaons belong to the next generation of mesons after the pions that TRIUMF now produces and they are the first “strange” particles. They are of fundamental interest in their own right — their decay has always been a puzzle — and because of the interesting effects they can produce e.g. when introduced into ordinary nuclei.

Figure 19 shows the TRIUMF site and indicates roughly where the synchrotron tunnel would be.

The KAON Factory would cost very roughly half a billion dollars, and last July the Federal and British Columbia Governments each provided five and a half million dollars for a Project Definition Study. The study, expected to take 15 months, will finalize cost estimates and evaluate the scientific and economic benefits. There is much international interest in this project and if Canada decides to go ahead it can expect contributions from abroad amounting to perhaps a quarter of the total cost.

The Sudbury Neutrino Observatory (Fig. 20) is already an international proposal involving the US, the UK and Canada. The uniquely Canadian contributions are the Creighton mine and one thousand tonnes of heavy water. Its prime purpose is to resolve the Solar Neutrino Problem, which is that the measured neutrino flux is a factor two to three smaller than predicted from models of solar energy generation. Neutrino interactions are very improbable (cross sections are about 10^{-19} barns) so that a large detector is needed to give a reasonable count rate. The rate is still very low so that it is necessary to work at great depth to attenuate cosmic rays, to choose a rock formation of very low radioactivity, and to use ultra pure materials in the detector (Fig. 21). Heavy water is a very advantageous detector — three different neutrino interactions can be studied which will be helpful in resolving the problem. The interactions produce Cerenkov light which is detected in some 2000 50-cm diameter photomultiplier tubes. The capital cost is about 50 M$, and I understand that prospects are good for funding.

Both these projects owe their existence to Canada’s response to the discovery of fission. It would be stretching things to claim that SNO is a lineal descendant of Pontecorvo’s pioneering work at Chalk River in the late forties, but the thousand tonnes of heavy water has a clear enough origin. KAON would be the son of TRIUMF, which was itself delivered by George Lawrence following the AECB tradition, established by C.J. Mackenzie, of encouraging the setting up of large nuclear physics fa-
Fig. 13 The layout of the TASCC tandem accelerator, superconducting cyclotron facility at Chalk River.

Fig. 14 The $8\pi$ spectrometer at the TASCC facility at Chalk River.

Fig. 15 The layout of the upgraded electron accelerator at the University of Saskatchewan.
Fig. 16 Beam lines of the electron linac on the right and the storage ring strung from the ceiling on the left, at the University of Saskatchewan.

Fig. 17 The 180-degree injection line and the ring in the background of the electron linac at the University of Saskatchewan.

Fig. 18 Layout of the proposed KAON Factory showing the existing TRIUMF cyclotron near the centre of the 30 GeV Synchrotron ring.
Fig. 19 A photograph of the TRIUMF site with an indication of where the main 30 GeV synchrotron ring would be placed.

Fig. 20 A schematic cross section of the Sudbury Neutrino Observatory (SNO) to be placed in the Creighton mine of INCO.

Fig. 21 Schematic of the SNO detector.
Fig. 22 Model of DUALSPEC, an instrument being installed at the NRU reactor at Chalk River to utilize two of the beam tubes.

Fig. 23 The high-throughput diffractometer that forms the lower part of DUALSPEC.

Fig. 24 The triple axis spectrometer which forms the upper part of DUALSPEC and which can use a polarized beam and has a polarizing analyzer.
This talk has stressed nuclear physics. Let me restore some balance by concluding with a few remarks on neutron scattering in its application to condensed matter physics. Thanks to the NRX and NRU reactors Canada was a pioneer in this business, and there is a strong continuing basic research effort at NRU, and an increasing amount of applied work using this technique which is carried out for industry on a commercial basis. Of the basic research, more than half involves researchers outside the Chalk River Nuclear Laboratories, many from Canadian universities. A new experimental facility is to be installed at NRU next year called DUALSPEC (Fig. 22). It is jointly funded by Chalk River and NSERC, on behalf of the university users, at a total cost of about three and a half million dollars.

DUALSPEC will use two beams from NRU. The lower one (Fig. 23) is for a high-throughput diffractometer for condensed-matter structure studies. It provides a high-intensity monochromatic beam and a large position-sensitive detector for rapid data collection.

The upper instrument (Fig. 24) is a triple-axis spectrometer with a polarized beam and a polarizing analyzer. These features will permit the study of spin-dependent scattering, which will be particularly useful in the investigation of magnetic materials.

With DUALSPEC operational NRU will be saturated. Canada needs a new research reactor. The one at McMaster is obsolescent and a proposal has been advanced to have it upgraded by installing an AECL Maple Core. A first attempt to secure funding for this was not successful, but, in my view, it would be a very economical way of meeting a national requirement, and at a university where the nuclear science program goes back to the early days of fission, indeed to the establishment of Harry Thode's laboratory as a division of the Montreal project in 1943.
The Development of Nuclear Power

John Foster

It is a great pleasure to be here this morning and to share the platform with distinguished old friends and colleagues in recalling some of the early experiences and impact of the discovery of nuclear fission. When I was invited to participate in this session it was suggested I might speak on the effects on industry and regulatory agencies in Canada of the development of the CANDU line of reactors. When I set about composing what I would say, however, that scope seemed rather too narrow, too parochial, and perhaps even too esoteric for this occasion. Furthermore, having read and enjoyed Bertrand Goldschmidt’s “Pionniers de l’Atome” and heard Alvin Weinberg ruminate in his usual interesting way on energy and its uses at the American National Academy of Engineering last year, and being very aware of the majestic sweep of views we could expect from Les Cook and of Geoff Hanna’s broad knowledge of modern physics, the scope seemed rather too focused. Finally, the notion didn’t accord with my idea of the fitness of things. It didn’t seem appropriate, assembled here as we are, to celebrate the fiftieth anniversary of the greatest thing in the field of energy since cooked meat, to devote a lot of time to the contemplation of heliarc welding and the Atomic Energy Control Board. So I have chosen a rather broader subject - “The Development of Nuclear Power”.

I was thinking of calling it “The Second Stage”. As Bertrand Goldschmidt has told us, the potential to produce useful energy was recognized from the very outset. As we’ve heard, however, the Second World War intervened and the fission reaction was first applied in the manufacture of bombs. This was the First Stage, the stage of development of the use of nuclear fission which uncovered many of the fundamentals relative to the process and demonstrated the feasibility of building and operating nuclear fission reactions. To develop any use of fission it was necessary to have comprehensive knowledge of the fission products and their properties in order for a chain reaction to be sustainable.

Not long after the war was over people turned their attention back to the constructive use of the energy available through the fission process. For the past thirty-five or forty years we have been creating a nuclear power industry, its related infrastructure and the building of the first generation of nuclear power plants. This I am regarding as the Second Stage — the development and widespread application of the first, basically simple, power reactors.

These first generation plants employ a simple fuel cycle, that is, their fuel is made
from fresh uranium and the used fuel goes to storage. There are, admittedly, fuel reprocessing plants, notably in Europe, that separate the unused uranium and plutonium from the waste products in the used fuel and return the recovered plutonium to the power reactors. This recycled fuel, however, contributes very little additional energy in these first generation reactors, does not materially affect their economics, and its use is not impelled by any world wide scarcity of fresh uranium. There are other motives for recycling fuel in current power reactors, not the least of which is preparation for the future. The time will come when it will be economic, even to the point of necessity, to extract much more of the energy available in uranium than we are doing today.

One of the nice things about nuclear fission — and there are many — is that the energy available in the fuel is not lost through inefficient initial use. When coal was used at 2 or 3 percent efficiency in the early steam engines, all of the useful constituents went up in smoke or out in the ashes and were gone forever. Not so with uranium. The part we don't use in today’s comparatively inefficient reactors remains available for use in more efficient reactors in the future. In my chronology, the employment of advanced fuel cycles is the Third Stage in the development of nuclear fission reactors. Although experimentation with such reactors and fuel cycles began in the very early days, and has led to the construction and operation of large prototype fast breeder power reactors in France and the USSR, widespread application — what I think of as the Third Stage — is decades away. Although the high efficiency reactors and associated fuel cycles have been significant, even important parts of many countries' nuclear programs, they are not part of my subject today. I am going to confine myself to what I called the Second Stage — the development and general application of the essentially simple fuel cycle power reactors that provide virtually all of our nuclear power today.

I am going to do this under the following topics:

The Technical Choices,
The Organizational Arrangements,
The Nature of the Task,
The Results,
Observations on the Present Situation.

THE TECHNICAL CHOICES

When countries turned their attention back to using nuclear fission to produce energy for constructive uses the first question was what kind of reactor to build. The potential scope for choice was very great indeed. The reactor might be moderated or not and, if moderated, graphite, beryllium, water, heavy water, or an organic liquid might be the moderating material. The fuel might be uranium metal or an alloy, oxide or carbide of uranium. Heat might be removed from the reactor by water, heavy water, carbon dioxide, helium, organic liquids, molten salts, or liquid metals. If water, it might be boiling or not. The fuel might be sheathed in stainless steel, zirconium alloys, graphite or other ceramics. Fortunately, not all the permutations and combinations are possible. Nevertheless, there are probably a few hundred conceivably feasible combinations and a surprising number were attempted. Many more were proposed. I am sure that a hundred years from now, when nuclear power is as normal as apple pie was before Meryl Streep, enterprising young engineers will find that their latest bright ideas were anticipated by someone in the 1950's.
The USA

The situation was not the same in every country. Uncle Sam, with the Manhattan Project behind him, a large weapons program under way, and a nuclear marine propulsion program started, was in a class by himself. Aware that the water-cooled graphite, heavy water production and light water propulsion reactors were chosen for reasons specific to their military roles based on the state of the art at the time of choice and with a large R&D, and that operational communities tend to propose better mouse-traps, the government sought proposals from industry/electric utility teams. Many of these proposals, including some from its own laboratories, were supported to the operating prototype stage. Others were dropped by the wayside. Debate swirled around whether the power program should be carried out by the government or by private enterprise; whether to concentrate on a single type or develop several; whether the first power reactor should be single-purpose (that is power only) or dual-purpose (power and plutonium); about the value of plutonium in power reactor economics; about the impediments of secrecy. Scientists became politicians and economists overnight. Walter Zinn described nuclear power economics as a complicated relationship among neutrons, dollars, and nitric acid.

Britain, France, and the USSR

For another group of countries the situation was quite different. These were America's major wartime allies - Britain, France and the USSR. (Canada might be regarded as a member of this group but I prefer to discuss this country separately.) The three countries I mentioned were not inhibited politically in proceeding with a nuclear weapons program, had all gained varying degrees of understanding of the business from nationals or friends who had been engaged in the Manhattan Project or the Montreal Laboratory, and had proceeded to build plutonium production reactors. All were using graphite-moderated reactors as the Americans had done, although the British and French were using gas rather than water for cooling. These countries did not have the experience or the resources of the United States. Nor did they, at that stage, have the same concern as the Americans about the appropriate roles of public and private enterprise, nor any compunction about confusing the economics by using a reactor to produce both plutonium and electricity. Their choice was very pragmatic. In the Joule Memorial Lecture of 1951 Sir John Cockcroft, after explaining that the gas-cooled graphite reactor technology was manageable, said: "we do not expect to produce a cheaper source of power than that derived from coal — it is likely, in fact, to be somewhat more expensive. What we are aiming at is to increase the total power available". He concluded by saying: "The essential thing is now to get on and build some power reactors". In 1952 Britain proceeded with the dual-purpose Calder Hall reactors and, about the same time, the French proceeded with the G series of similar reactors at Marcoule. The USSR had begun construction of a 5 MW prototype water-cooled graphite power reactor which went into operation in 1954.

PWR

Meanwhile, back at the ranch — the U.S. ranch — debate still raged and then one day there was stunned silence. The U.S. Atomic Energy Commission had announced that it was going to build a 60 MW nuclear power plant employing an enriched uranium, pressurized light water reactor: PWR. To that point this type of reactor had received hardly any serious consideration
for electrical utility use. It was not among those favoured by the study teams that began work in 1950. In fact, in a Nucleonics editorial, its editor, Jerome Luntz, said: "the Joint Congressional Committee on Atomic Energy has labelled the PWR least likely to succeed in the achievement of economically competitive nuclear power". On the other hand, as he also reported: "supporters of the project optimistically stick their necks out and say that a 1964 version of the PWR, having benefited from operating experience and advances in the reactor art, will produce competitive power in the United States". If not a front runner, how did the PWR get chosen as the first to build?

After the war, goaded by then Captain Rickover, the US Navy, in conjunction with the USAEC, commenced development of nuclear power reactors for naval ship propulsion. For this application, economics was not a prime consideration: compactness, endurance, reliability were. The pressurized light water reactor was a very wise choice. By 1952 the first land-based prototype for submarine propulsion was nearing completion in Idaho. In the middle of that year a 75,000 hp unit for aircraft-carrier propulsion, called CVR, was authorized. In May 1953 this was canceled. Admiral Rickover, as he then was, was not the kind of person to be deterred from developing a reactor for a carrier by a little thing like the lack of a carrier. Congress voted the USAEC $7,000,000 for power reactor development and in the middle of the year (1953) the USAEC authorized the construction of the 60 MW(e) PWR, the project to be run by Admiral Rickover and the nuclear plant to be built by the former CVR contractor, Westinghouse. If I had to pick the single decision that had the most far-reaching effects on the course of the development to date of nuclear power, a leading candidate would be the decision by the US Congress, in 1953, not to proceed with a nuclear aircraft-carrier.

Canada

Bertrand Goldschmidt has told us about the early history of nuclear fission leading up to the installation of a heavy-water moderated reactor, NRX, at Chalk River. Conceived in wartime to produce plutonium, it began operation in 1947 and, although it was used to produce plutonium, its main role has been as an experimentation and test reactor. In 1952 Atomic Energy of Canada Limited was created by the Canadian Government and the Chalk River Nuclear Laboratories, started during the war, were transferred to this new company. A main objective was the development of nuclear power. Although there was a ferment of discussion about possible reactors for this purpose and a good deal of uncertainty about the approach to be taken, there was general agreement at senior technical levels that the heavy-water reactor, with which they had experience, was the proper choice. In 1954 a small team of engineers from electrical utilities and industry across Canada was assembled to work with AECL staff in investigating the feasibility of utilizing a natural-uranium fueled, heavy-water moderated reactor for electrical power production. In 1955 it was decided to proceed with this type of reactor. Like Uncle Sam and his other main wartime allies, Canada chose to develop for electrical power production a type of reactor first built in the country to serve a military purpose.

Other Countries

The third class of countries to enter the field were the smaller allies, neutrals, and the vanquished in World War II. Without pretensions to nuclear arms, the nuclear agencies they created were relatively free spirits without either the benefits or
constraints of a military program. Because of its high moderating efficiency, heavy water had an intrinsic appeal to the physicists who played important roles in these agencies and, more importantly, offered the possibility of economic nuclear power without dependence on others for fuel enrichment or fuel reprocessing services. West Germany, Sweden, Switzerland, Norway and Holland together, and Czechoslovakia all built prototype heavy water power reactors. (Incidentally so did Britain, France, the United States, and later Japan). Some ran into problems but, in any event, the utilities in these and other countries could not wait for their domestic nuclear agencies to develop systems when types proven elsewhere were becoming available. Italy and Japan each bought a gas-cooled graphite unit from Britain who was first off the mark with a substantial nuclear power program and had something operating to show potential buyers. France shared a similar unit of its own manufacture with Spain. By the beginning of the sixties, however, the American program was coming up to speed. General Electric was in the field with the BWR, a boiling light-water reactor and, with Westinghouse, these two companies, directly or through licences, dominated the world market. BWRs were installed, under licence, by Ansaldo in Italy; by AEG in Germany and Switzerland; by General Electric itself and by Hitachi and Toshiba in Japan; and in Spain, India and Taiwan. Westinghouse PWRs were built everywhere throughout Western Europe. Even France built a PWR station in conjunction with Belgium. ASEA in Sweden had developed its own BWR but that country also built Westinghouse PWRs. Westinghouse and Mitsubishi, under licence, built PWRs in Japan. Later, Westinghouse built units in Korea. The USSR, the second power to build nuclear-powered submarines, developed its own version of the PWR. Besides employing this type to supply half its domestic needs, the USSR exported reactors of this type to all the countries of Eastern Europe except Romania.

There was continuing interest in the heavy-water reactor because of its simple fuel and the independence this offered with respect to fuel supply. During the period when the American and Soviet light-water reactors were flooding the world, or at least that part of it that could use nuclear power, Canada sold CANDU stations to India and Pakistan.

By the late sixties the American domestic nuclear power program was burgeoning. The country's two major steam generating equipment suppliers had entered the field as suppliers of large components, notably the reactor vessels, to GE and Westinghouse, and as suppliers of PWR nuclear steam supply systems in their own right. With the great success of the light-water reactors at home and abroad, the variety of other types that Uncle Sam had experimented with gradually passed out of the picture.

LWR's, Graphite Reactors, and HWR's

Throughout the world the field was left to the light-water reactors, the gas- and water-cooled graphite reactors and the heavy-water reactor, all reactors chosen in the First Stage of the development of fission for military purposes. The world had beaten swords into ploughshares and spears into pruning hooks.

For someone unfamiliar with the times this might seem simply a matter of industrial momentum or expediency. Certainly there was everywhere an urgency to get on with the peaceful application of this new, wonderful resource; but there was real conviction in the choices.
For the British and the French the high temperature available from a gas-cooled reactor was key. I remember this coming up when I was participating in a small round table session in the Hochschule in Lausanne after the 1964 World Energy Conference Sectional Meeting. It was raised by Sir Christopher Hinton, former head of the UKAEA’s engineering establishment at Risley, which designed the first production and power reactors, and in 1964 Chairman of the CEGB. He asked: “We thought high temperature was extremely important. Were we wrong?” I think it was Ken Davis, previously with the USAEC, then with Bechtel, and who was later to be Assistant Secretary of Energy in the first Reagan Administration, who said: “No. You weren’t wrong, but there is a lot more than thermal efficiency that goes into the success of a nuclear power system.”

In this country, the attractions of a reactor system that could be built largely with our own resources, did not require enrichment for its fuel cycle nor reprocessing to make it be economic (as was then thought to be necessary for light water reactors) were the main determinants in our choice. They weighed every bit as heavily as did the experience with the heavy-water reactor.

The simplest nuclear steam supply system is that employing a light-water reactor. The moderating properties of light water are such that there is no room in the reactor for a separate coolant. Fortunately light water itself is an excellent coolant. The result is a common moderator and coolant system using the heat transfer and transport medium we know the best — good old hot water. Drawbacks are the need for large pressure vessels and enriched fuel. In the United States, with manufacturing plants capable of producing all the components required for light-water reactors, including the large thick-walled vessels, and with uranium enrichment facilities in place, a consensus rapidly grew that the light-water reactor was the logical choice for power generation.

Demise of the Graphite Reactors

By the mid-sixties, with the success of the light-water reactor in world markets, doubts began to grow in Britain and France, who now also had enrichment and fuel reprocessing facilities, about the wisdom of the choice of the gas-cooled, graphite reactors. Important constituencies in the national electric utilities and the major electric generating equipment manufacturers wanted to be in on what they called the mainstream-building and using light-water reactors. What happened next is a very complicated history involving operating and schedule problems in the gas-graphite programs, rivalries between the national nuclear agencies and the utilities, attempts by the nuclear agencies to develop heavy-water power reactors to provide an alternative, reviews within and between the electrical and nuclear organizations and by the governments, and so on. By 1970 France had opted for light-water reactors. The conversion in Britain was much more protracted because it coincided with a period when the country had ample generating capacity and low load growth. In fact there were so many reviews of so many different kinds at so many levels it is difficult to pick a date when the effective decision was actually taken. It is safe to say, though, that the decision to adopt light-water reactors was taken in Britain by 1985.

In 1986 the No. 2 reactor at Chernobyl, one of the line of boiling-water cooled, graphite reactors in the USSR, suffered its disastrous accident. The USSR has decided to phase out that line and to rely entirely on their PWR line for electric power produc-
The heavy-water reactor of the type developed in Canada now remains the only active alternative to the light-water reactor. There are three reasons for this: it uses uranium oxide, water and zirconium alloys like the light-water reactors; it is the variant of water reactor which offers simple, economic fueling; and we did a good job of it.

Besides meeting its own domestic requirements Canada has provided units and related technology to India, Pakistan, Argentina, Korea, and Romania. It is fifth in the world, only slightly behind France and Germany, in power reactor exports. Romania is in the process of building five of these units. India has built four with her own resources and has six more under construction.

I was in India recently and visited some of their facilities. The first unit in their fourth nuclear power plant at Narora on the Ganges, a 235 MW unit of the CANDU type, was just starting up. There is another two-unit plant of this kind in an advanced stage of construction and three more beginning construction. I saw the excavation for the pair that will be located beside the Rajasthan Atomic Power Station which we helped build. Besides these 235 MW units, India is commencing to build several plants with 500 MW units of the same type, to their own design. It seems strange, in walking around their new units which look so familiar, to see important equipment, such as the primary heat transport pumps from West Germany, that has come from third countries. Altogether the country plans to have 10,000 MW installed, in 200 and 500 MW units, by the year 2000. The attraction for countries with these plants is the simple fuel cycle which they can manage themselves. For India there is the added attraction that the heavy-water power reactor promises to be the right vehicle for the thorium/U-233 fuel cycle in the future, because India has large thorium deposits but apparently limited uranium resources.

THE ORGANIZATIONAL ARRANGEMENTS

The USA

The United States has a long tradition of turning to private enterprise to provide its goods and most of its services other than the military, postal, police, and essential municipal services. It was natural, therefore, in the Manhattan Project, to turn to large companies like DuPont, Union Carbide, Phillips Petroleum, and General Electric to operate laboratories and production centers. The Navy turned to its long time suppliers of boilers and propulsion machinery — Babcock-Wilcox, Combustion Engineering, General Electric and Westinghouse — for the equipment and systems needed for the nuclear ships. As a result all of these companies and many more had relevant experience when the country turned its attention to nuclear power. Many of them became partners in the industry/utility teams that responded to the Government's call for proposals for prototype nuclear power plants. More importantly, it was the most natural thing in the world, once the light-water reactor had been settled upon, for the utilities to turn to the navy's nuclear plant suppliers. The only novelty was that it was Westinghouse and GE, presumably because they had greater muscle, that had first been chosen by the Navy to provide the marine nuclear steam supply systems. Steam generation equipment was more naturally the province of the boiler manufacturers and, in due course, Babcock-Wilcox and Combustion Engineering entered the competition for utility NSSS (Nuclear Steam Supply System) business. As the other types of reactors faded from the American scene the suppliers associ-
ated with them either withdrew or became sub-suppliers to the main NSSS contractors. For several years the American market was large enough to accommodate the two boiler and two electrical manufacturers. When that market dried up GE withdrew from the market-place. It has always been the company position that they are manufacturers and that they do only such engineering as is necessary to sell their products. NSSS supply involved an unusually high degree of engineering and subcontracting. It is not surprising that as the market withered, even if it is only for the short term, and particularly because of some of the troubles they had with their boiling-water version - the BWR - and a decline in interest in this type, that they should decide to pull in their horns. Nevertheless the NSSS market in the United States has operated as and remains a typical free economy market with the established major steam generating and electrical equipment manufacturers prominent in it.

Britain and France

In Britain and France the situation was somewhat different. Perhaps because of the tradition of royal arsenals or because it was felt that there was not the necessary industrial capability or for security or for all of these and other reasons, the governments of these two countries elected to create nuclear agencies that would be operating organizations as opposed to the USAEC which was primarily administrative. Consequently these agencies comprised major engineering components that were responsible for the building and operation of production and prototype power reactors.

Britain

When the power program proper began in Great Britain, the Central Electric-

ity Generating Board placed turnkey contracts with consortia of manufacturers and civil contractors. These depended on the UKAEA engineering office at Risley for the basic plant designs, at first for those using the Magnox (natural-uranium metal-fueled) reactors and later for the AGRs (reactors using enriched-oxide fuel). At first there were only two consortia and the manufacturers were boiler and electrical generating equipment manufacturers. The program expanded very rapidly, however, and soon there were five consortia which even included aircraft manufacturers in their ranks. The market could not support this number as electrical energy demand stagnated and so the number sank just as rapidly back to two and eventually to one through withdrawals and amalgamations. The process was further accelerated by the general process of rationalization that was going on at the same time in the British electrical equipment supply industry. Eventually, to preserve even one supplier of nuclear power plants, it was necessary for the government, through some of its companies, to take a shareholding in the remaining company.

France

The situation in France was different but the result was much the same. During the sixties when France was building gas-graphite reactors, there was strong input from the Commissariat de l'Énergie Atomique in the nuclear system engineering and EDF employed a variety of contractors for the nuclear part of the plants. With the decision to switch to light-water reactors, EDF turned to the Westinghouse licensee, Framatome, and the GE licensee, CEG (Compagnie Electrique Generale)-Babcock/Atlantique, for the nuclear steam supply systems. Despite the disadvantage of being part of the Belgian Baron Empain's empire and with a 45% Westinghouse sharehold-
ing, Framatome won the first four orders in 1970-71 on price. By 1975 Framatome had been designated sole supplier to EDF with a responsibility to develop exports. 30% of the 45% Westinghouse shareholding was transferred to the CEA and the CEA mounted a major PWR research program in conjunction with Westinghouse as an extension to its own submarine propulsion program. The CEA acquired the remaining 15% Westinghouse share when the license agreement lapsed in 1982. Like Britain, France now has a single industry/government company for NSSS supply. As their nuclear agencies withdrew from direct participation in the power reactor program they switched their focus to the fuel cycle and today British Nuclear Fuels and Co-gema in France operate major fuel manufacturing, reprocessing, and associated facilities.

Germany

In the Federal Republic of Germany, without a military nuclear program, the young national atomic energy agency did not have the power nor the experience of building and operating production reactors. The utilities and industry were freer to follow their own inclinations from the outset. Siemens did develop and build a prototype heavy-water power reactor and subsequently sold two to Argentina, but with the success of the light-water reactors in the USA and their sale to Italy and Japan, Germany opted to install plants of this type. The country's two main electrical equipment suppliers, Siemens and AEG, supplied the NSSS's under licence from longtime associates Westinghouse and GE respectively. Two other major equipment suppliers to the electrical utilities, Brown-Boveri and Babcock got into the market belatedly as BBR but the first round was over before they got properly established. To economize on reactor and turbo-generator research and development and on marketing and, incidentally, to reduce competition, Siemens and AEG created a joint company, Kraftwerke Union (KWU), in 1969. Westinghouse canceled its licence with Siemens. Although the AEG side sold more units in the beginning, problems forced them to withdraw in 1976. Today Siemens is the sole NSSS supplier in the Federal Republic of Germany. And now another stage of rationalization is incipient. Perhaps in preparation for the increased integration that is coming in Europe in 1992, but more in response to the poor state of the business, Siemens and Framatome have entered into a joint marketing agreement.

Japan

After the purchase of a single gas-graphite unit from Britain, Japan opted for light-water reactors, at first buying plants directly from GE and Westinghouse and subsequently building them under licence. The situation in Japan is rather similar to that in Germany with the exception that Toshiba and Hitachi, the General Electric licensees, and Mitsubishi, the Westinghouse licensee, have found a way to coexist.

Canada

In Canada the situation was different from that in any of these other countries. The nuclear power program began in 1955 with the prototype NPD station. AECL, after visits by directors to the United States and Britain and with the agreement of the government, adopted a course somewhat similar to that being followed in those two countries: that is, a manufacturer was selected to provide the NSSS. The arrangement was special in that the manufacturer, Canadian General Electric, was a partner with AECL and the utility in the project, Ontario Hydro, contributing $2 million worth of services. Although there was a
superficial similarity to the situation elsewhere, there was an underlying basic difference. Ontario Hydro was not prepared to be dependent on a manufacturer for the engineering for a type of plant that it was convinced was going to be very important to it in the future, especially when most of the special knowledge was being developed at public expense. It would have preferred to do the system engineering itself. The utility, however, was not in a position to do this. The outcome was the creation of a nuclear steam supply system engineering organization within AECL. This organization also took on project management responsibilities where AECL had a major financial stake, such as in prototype plants in Canada and in export projects.

This arrangement was well suited to the circumstances in Canada. Canadian companies were only a fraction the size of their counterparts in the United States and Europe and the main suppliers of boilers and electrical equipment were subsidiaries of foreign companies, mainly, at that time, American and British. By putting the NSSS engineering with the utilities, it assured the ability to maintain a Canadian program. It also meant that the relatively small Canadian manufacturers were being asked to supply components, such as steam generators, reactor vessels, pumps, etc., against well-developed specifications rather than a whole nuclear steam supply system against a necessarily much vaguer specification. In time, Ontario Hydro has assumed more of the NSSS engineering for its own power plants. This is consistent with its original preference but creates a dilemma as to how best to serve other customers for the CANDU system.

The Seat of Control

The essence of these organizational arrangements is who has control of the NSSS engineering, for that is the seat of a large degree of control of the technical and economic aspects of the application of nuclear power, and the checks and balances in the arrangement to prevent this power being abused.

In the United States, with the largest electrical power market in the world, a large number of electric utilities, and a strong belief in free enterprise, there are four large, competing NSSS suppliers and the nuclear steam supply system engineering is performed by the suppliers. In Japan, with the third largest electrical market in the world, several privately owned utilities and with the same belief in private enterprise, the situation is similar but with only two or three suppliers. In Germany, too, the electricity supply environment is similar to that in the United States but, with an even smaller market, there is now only one supplier. However with none of the utilities shopping for nuclear power plants right now, other possible suppliers in the wings, and pan-Europe of 1992, things will probably change. In short, in all these countries, NSSS engineering rests with the suppliers, and the buyers rely on commercial competition to keep things in hand.

Britain and France have, as yet, smaller domestic electricity markets and these are served by publicly-owned national utilities. (This is being changed in Britain but nuclear electricity will remain a national undertaking.) Like Germany, they each now have a single NSSS supplier. The system engineering rests with these suppliers. However, the respective governments have taken major positions in these companies so that control of system engineering cannot be said to rest in the suppliers to the extent that it does in the first three countries but is divided, through interlocking ownership, between buyer and supplier.
Canada has the fourth largest domestic electricity market, between those of Japan and the Federal Republic of Germany, and is the world's largest exporter of electricity, exporting about 10% as much as it uses at home. Most of this is supplied by publicly-owned provincial utilities. Because of hydro and coal resources in other parts of the country the market for nuclear electricity is almost entirely in Ontario at the present time. Unlike the other countries, Canada has no NSSS supplier for its domestic market. (AECL acts in this capacity for export projects.) Instead, the NSSS engineering is performed by an engineering organization (AECL's CANDU Operations) engaged by the buyer, the utility. Control of system engineering rests with the buyer.

So we have the whole spectrum of system engineering control in the organizational arrangements in the various countries, from equipment-supplier control, through a sort of hybrid, to buyer-control.

THE NATURE OF THE TASK

The organizational arrangements are extremely important in determining how much voice which entities have in shaping the nuclear power program, the framework within which the work will be done, the objectives to be met in doing the work and the philosophies that will be brought to bear. Once this was settled, however, the task for those doing the job was much the same everywhere. Basically, it was to bring new knowledge and old experience together to produce successfully operating power plants on utility systems with all the necessary supporting facilities and infrastructure. The design of the plants, fuel, and equipment and of the necessary facilities to manufacture these and special materials; construction, commissioning and operation of them, together with the research and development to support all phases; the training; the establishment of regulatory organizations and regimes; the economic assessments and accounting; dealings with governments, politicians, special interest groups and the public; and so on involved most professional disciplines and trades and was, in fact, the establishment of a whole new business, I'm tempted to say economic sector. Even within a single unit, the adoption of nuclear power had far-reaching ramifications. A utility who thought they were just buying another way to boil water, and some did, were quickly disabused of this notion or suffered some hard knocks.

Clearly, in this paper I am not going to deal with this whole ball of wax. It is not a simple matter to talk about any one aspect, say plant design, without invoking many others for they are all inextricably bound together, except at the very technical level. I will therefore simply say a few things based on experience and personal observations to give some of the flavour of the task of developing nuclear power.

The Teams

Given the job of creating a nuclear power plant, the first thing was the selection of the team or rather teams to do it. Because of the need for new knowledge and experience what we needed was new graduates with 20 years experience. What we did, of course, was to hire a mixture of youth with the up-to-date knowledge and old hands with relevant experience. Extremely few of the old hands, however, had 20 years experience. When I was put in charge of the design of Canada's first nuclear power plant I had 12 years experience. I had 15 years of experience when I was put in charge of AECL's Nuclear Power Plant Division to engineer and manage the project for our second plant. The great majority of the
team in both instances were about the same age. Our homologues in other countries were also of about the same vintage. Although the average age was somewhat greater than in the First Stage (the wartime developments) we were still in an era when considerable responsibility devolved, of necessity, upon relatively young people. There was another novelty for Canadians. In established businesses such as the design of coal-fired power plants, when the utility went to a new size of unit, for instance, it was customary for management to go to someone who had been there before, usually in the U.S. or perhaps from the U.K., to hold the engineer's hand. With the first nuclear power plants, no one had been there before.

There was another key characteristic of the engineering organizations. The initial professional staff was an amalgam of scientists and specialist engineers, such as reactor, process and control engineers, drawn from the laboratories and other establishments already engaged in the development of nuclear energy, and of equipment and power plant design engineers drawn from the owner's organization or assembled by him. From that point on the organization had a life of its own and acquired whatever personnel it needed for completeness and expansion. Although the number of scientists and specialists was small in relation to the number of engineers in the more common disciplines, it was a new experience for most of the engineers to be working in an environment where the application of fresh scientific knowledge was an integral part of the design process. For the scientists, too, it was a new experience working in an engineering milieu although most of them who came to the Canadian organizations at least tended to be those with an interest in being closer to practical applications. One of the incidental consequences of the introduction of scientists into an engineering office was a heightened skepticism of the basis for designs of products available on the market.

The initial nuclear power plant operating staffs were built the same way, with a mixture of those who had research or production reactor operating experience and those who had power plant operating experience.

Innovation

The task was not only to do the job to build a power plant but, at the same time, to create an organization. In design there was a third problem. This was, of course, to create a new design for at that time we did not even know what materials we should use nor what basic methods of construction we should adopt, let alone the many other details that had to be resolved.

For instance, the Americans, who had started with the submarine program and were a few years ahead of us in their nuclear power program, were using stainless steel in the primary heat transport and other reactor water systems. Purity of water is essential to avoid fouling of fuel, other heat transfer surfaces and to prevent the buildup and circulation of corrosion products which would become radioactive and hamper maintenance. Our metallurgist at CGE on the NPD project considered that this did not prevent the use of carbon steel in these circuits and that the risk of stress corrosion cracking with stainless steel was of greater concern. He discussed his opinion with his peers at Chalk River who agreed with him. None of us to whom they reported had any reason not to accept the recommendation and carbon steel was adopted. It was successful and has been used in CANDU systems and their foreign variants ever since. BWR's did run into some stress corrosion problems in particular locations in their
stainless circuits but these have long since been resolved. Today the choice of material for these circuits can be almost automatic but, in the beginning, it was a major decision which, if wrong, could, as we were all very aware, jeopardize the whole project and maybe the program.

Let me give you an example concerning a construction method. For strength, corrosion resistance and, above all, neutron economy the pressure tubes in CANDU reactors are made of zirconium alloys. There are special fittings at each end, made of a high alloy steel, which connect to the external circuits and through which the fuel enters and leaves the reactor. The requirement was to effect a satisfactory joint between the zirconium alloy tube and the steel end fitting. To give some idea of the possible scope, a few years later, for the EL-4 gas-cooled heavy water reactor in France, a compound joint consisting of a screwed joint, a brazed joint, and an electron beam weldment was adopted for a similar purpose. The higher gas temperatures were a factor but it serves to illustrate the scope available at that time. For NPD, the head of reactor design proposed using a rolled joint. Rolled joints, in which the ends of tubes are expanded into other pieces with a mandril and rollers, had been used for a hundred years or more to connect boiler tubes to boiler drums and the tubes of other heat exchangers to their tube plates, although by the 1950’s the method was being superseded by welding and other processes. When it was proposed to me it seemed a bit old-fashioned but it was simple, it seemed that it might work, and there were no competing suggestions with any more promise. It was adopted. Of course it was necessary to mount a considerable development program to prove it and there were lessons concerning the detail that had to be learnt down the years in operation. A few years after we had adopted the rolled joint, when we were engaged in a mutual development program with the Americans, Dale Babcock of DuPont who had a responsible role in the engineering of the first production reactors at Hanford asked me how we had come to choose it. I said: “lack of imagination” and his comment was that in engineering, choice of the simplest was very often the right choice.

Quality

For the equipment manufacturer it meant working with new materials selected for radiation resistance, corrosion resistance, and suitability to forming and fabricating into the required shapes to new levels of precision, cleanliness and quality control. New codes were developed to cover many of these requirements. A new quality-monitoring process was introduced, unfortunately called quality assurance. Until the middle of this century customer inspection tended to be of two general classes:

1. a rather superficial in-process inspection carried out by customer’s inspectors on periodic visits plus a final acceptance test or inspection. This was usual for utility purchases.
2. continual in-process inspection carried out by the customer’s resident inspection staff. This was the practice for many military purchases such as the purchase of aircraft.

For nuclear power it was felt that more of the first and less of the second was required to obtain the necessary quality. A system was instituted which entailed inspecting and approving the supplier’s quality control organization, staff qualifications, material control, inspection procedures and facilities. This was followed by periodic audits of all of these and of their performance, plus periodic customer inspections of the work in process and of course a final inspection. In Canada we
called this system quality surveillance. In time this term was superseded by the term quality assurance which the Americans applied to a similar system and which has now been universally adopted.

In-core components were a whole new ball game since neutron capture characteristics and behaviour in strong neutron, gamma and other radiation fields were of paramount importance. The fuel required a whole new science and laboratories, like those at Chalk River, devoting a major part of their programs to understanding its behavior. For the manufacturers it was a brand new business whose character was entirely different from that of the other product-lines for nuclear power. Being a consumable it required mass production, with very high quality, on a considerable scale. The demand for it, unlike that for the other products was not related to the rate of installation of nuclear power plants, but integrated with the number installed. I remember remarking to colleagues in CGE when, very early in the NPD project, it had been suggested that fuel was not within CGE's scope of supply, a suggestion that came to nought, that making nuclear fuel is tantamount to having a royalty on electricity.

Everywhere new ground was being broken

In nuclear power plant operations it was necessary to develop new operating procedures with all the characteristics of other thermal generating stations plus radiation and a new class of control. New levels of staff selection and training were introduced, along with new organizations. In Canada, for instance, the traditional trades employed on station maintenance were replaced by new categories of employees, maintainers with multi-trade capability to avoid the unnecessary radiation exposure required by the traditional organizations.

Because of the hazard from the radioactivity produced, safety with respect to this new risk was a serious concern of all those engaged in any phase of these blossoming nuclear power programs. In addition, in most countries right at the outset, and in all countries eventually, separate national regulatory bodies were established to oversee the conduct of all these phases. Here, also, new policies and practices had to be devised. Different tacks were taken in different places. In the United States the Nuclear Regulatory Commission produced mandatory detailed guidelines and these were adopted by some of the countries that acquired American reactors. In other countries only mandatory principles were established and the practices and results of the operating organization, whether it was a mine, engineering or plant operating organization, were reviewed to see that they complied with the guidelines and met the safety objectives. The distinction in effect between the two approaches appears to be diminishing with the passage of time. There is, however, a more fundamental distinction between American practice and that in other countries. Approval of plans is, in the United States, very much more subject to a process of public argument and judgment than is the case elsewhere, where more reliance is placed on the professional judgment of a regulatory authority. Again, this distinction may tend to get blurred as environmental assessments, based on the adversary process, become more common.

From before the first atomic bomb was dropped there was always the concern of how to extend the benefits of nuclear fission without spreading the capability to produce bombs. Early reactors given or sold to foreign countries, such as the CIR research reactor given to India by Canada,
were exported under agreements stipulating that they were for peaceful purposes only. After the International Atomic Energy Agency established a system of international safeguards, it was mandatory that international trade in nuclear materials and facilities invoke these safeguards. In 1968 the Non-Proliferation Treaty received sufficient national ratifications to go into effect. This was an agreement whereby the signatory countries, acknowledged to have weapons, undertook to assist the other countries even with peaceful explosions if these seemed efficacious; and the non-nuclear weapon states foreswore acquiring such weapons so long as they remained signatories. Today Canada requires that before non-nuclear weapons states can receive nuclear materials and equipment they be a signatory of NPT, accept IAEA safeguards and enter into a bilateral agreement containing further restrictions, particularly on retransfer of material and technology. Many recipient countries find these restrictions denigrating so that export negotiations are greatly complicated by this factor. This is further exacerbated when the international safeguards regimes are evolving, as they were in the sixties and seventies, and when national requirements are also changing. Safeguards against diversion of nuclear items to military purposes is an extremely important factor in extending nuclear power from one country to another, and important regimes, national and international, have been established to control it.

In the early stages of development of every phase of this new energy sector, creation of new organizations coupled with innovation in design, manufacturing, construction, operation, R&D, and regulation were associated with the central task of getting the job done. It would be natural to think that once the organizations and the innovations were in place that the later jobs would get done much more expeditiously and efficiently. This didn't happen. Many later projects have taken up to twice as long to complete as the pioneer projects. There are several reasons for this, some of them internal to the sector itself, others social in origin.

In the established operating (as distinct from regulatory) organizations there are many more knowledgeable people who need to be consulted and there is immeasurably more information, much of it derived from experience with the earlier projects, that needs to be taken into account. In the beginning, although there was a more collegial atmosphere with broader discussions, there were perhaps only a dozen people who had an important part in all the most important decisions. The regulatory organizations, too, and their practices have been enormously elaborated since the early days and the interplay between them and the operating organizations is much more time-consuming.

But the external factors are at least as great as the internal. In the fifties and sixties the world was rebounding from the depression and World War II. An inadequate, outdated world infrastructure was being replaced. Housing was being built at an unprecedented rate, the automobile population was expanding rapidly everywhere and our modern highway systems were being built; jet fleets were replacing the old propeller-driven aircraft and modern airports were built to handle the exploding traffic and most of the baggage; communications were being transformed by solid-state physics and satellites; health care was improving dramatically and becoming more universal; and all these things required more energy. Not only was there the crying need for all these facilities after the Depression and the War, there was a wealth of pent up scientific knowledge waiting to be exploited.
Fig. 1 Number of nuclear power units entering service worldwide since 1954.

Fig. 2 The net orders for nuclear power units worldwide.

Fig. 3 Predicted cumulative thermal-electric generation additions in North America from 1988 to 1996.
and, above all, a fertile social climate. After the constraints of the Depression and the War there was an enthusiasm for seeing things happen. Politicians gained points by announcing new projects and eagerly sought opportunities to do so. A residue of the same experiences was a considerable acceptance of authority and a general respect for science. There was a third unrecognized attitude that distinguished those times from these: for a generation that had seen mighty armies assembled, equipped, and established overseas, navies built, air armadas launched, major cities in several parts of the world destroyed and rebuilt all in the space of a dozen years, it was unimaginable that it could take long to build a power plant.

The dynamic building era petered out in the seventies. The new infrastructure was largely in place, in the OECD countries at least. An increasing proportion of human activity swung away from building capital facilities to trade in consumables and wealth, and with a new toy, the computer, to expand the game. The demand for energy leveled off. The balance of social attitude changed from enthusiasm for what had been termed progress to growing concern for society and the environment, expressed most strongly in negative reactions to large projects. The leveling off of demand for electricity found several power plants in the pipeline that now would not be needed for a long time. Some of these were canceled; some were deliberately delayed. The reduced requirement for new plants allowed authorities to be extremely responsive to the reaction against them and even the utilities to be more tolerant of delay so that many project schedules were considerably protracted. The first wave in nuclear power plant construction subsided, but great things had been achieved.

THE RESULTS

The Power Plants

Figure 1 shows the number of nuclear units entering service worldwide since the first unit in 1954. There was an initial peak in plant commissioning in the mid-seventies but the crest of the first wave of nuclear power plants wasn't reached until about 1984, 30 years after the first unit. The decline after the crest is not as dramatic as suggested by the chart because there are still about 100 new units in the pipeline so that we can expect at least 10 a year to enter service for a few years yet. Nevertheless 1984 marks a crest that is quite different in character from the earlier peak.

As can be seen from Fig. 2 the net orders for nuclear units worldwide reached a plateau at about 500 units in 1979, about 400 of which are now in service. There will be relatively slow growth in nuclear power during the nineties. But, in case it should be thought that this is peculiar to nuclear power, Fig. 3 shows cumulative thermal-electric generation additions in North America from 1988 to 1996. As can be seen the expected additions of nuclear and fossil fueled units are the same over that period in the United States although, admittedly the trend favors the latter at the end of the period. In Canada, the four large units at Darlington considerably exceed other thermal generating capacity additions during the period.

Nuclear power plants are now producing about a sixth of the world's electricity, more than the world's total electricity supply when the programs began. Nineteen industrialized countries have nuclear power plants and seven developing countries, although 80% of the capacity in the latter group is in three countries - Taiwan,
Korea, and India.

In addition to providing a large amount of electricity, nuclear power has done this safely. Were it not for the Chernobyl accident I should have said remarkably safely. As a direct result of that accident, however, 31 people were killed and more than 200 hospitalized. On the unproven assumption that fatalities are directly proportional to exposure at any level, several thousand more people will have their lives shortened. Nuclear power has had such a good record, however, that even this accident does not remove it from the category of safe industries. A good measure of the social value of a thing is the amount people pay for it. Even taking the estimated delayed fatalities from the Chernobyl accident into account, nuclear power is as safe, on a comparable value basis, as manufacturing or the service industries, 10 times as safe as primary industries. On the same basis, it is as safe as air travel, 10 times as safe as the automobile or life around the home.

Fuel Cycle and other Facilities

In the early fifties there was concern about the availability of enough uranium for the military programs. The United States offered attractive contracts to those who could find and open new mines. The requirements were very soon met. Today, uranium mining is on a very strong footing in several countries, including Canada which is the world's largest producer. It is now estimated [OECD(NEA)/IAEA International Uranium Resources Evaluation Project, 1977 updated 1983] that total world uranium resources, recoverable at up to $260/kg (equivalent to coal at about $4 a ton), are 20 to 28 million tones. If used in reactors of current design, this is enough to provide electricity, at the rate the world is using it today, for 50 years. If used in breeder reactors at 50% utilization, it could provide all the primary energy, at the rate the world is using it today, for a millennium.

As a matter of interest, the amount of energy extractable in existing power plants from Canada's annual uranium exports is twice as great as the energy available in all our coal, oil, gas, and electricity exports combined, and more than three times as great as our net exports in these commodities. Another interesting relationship is that, in the 80,000 tonnes already exported, Canada has transferred resources that have more potential energy, at 50% utilization possible in future systems, than all the energy thought to be available from all the coal, oil, gas, and even the tar sands in this country.

At this time, too, adequate enrichment facilities exist in the United States, the USSR, France, Britain, and in a joint British-German-Dutch plant to supply world needs for some time to come. South Africa has also built an enrichment plant. The large fuel reprocessing plants in Britain and France are serving the international market as well as meeting domestic requirements. Canada is self-sufficient in heavy water, India has several plants in operation and is well on the way to self-sufficiency, and Argentina and Romania have some capacity to serve their domestic needs.

Over the past 35 years a whole new economic sector has been created based on the understanding of nuclear fission. The most obvious results of this are the physical manifestations-the power plants, the mines, the R&D establishments, the fuel cycle facilities. These, however, have a relatively short life and their benefit to mankind limited and transitory. The great result of the past 35 years of worldwide endeavor, by a million or more people in all walks of life, is that the way to put to use one of the world's
greatest energy resources is now firmly within human ken.

CLOSING OBSERVATIONS

We are just over the crest of the first undulation in the application of nuclear power. This was only a foothill. Periods of much greater expansion in the use of nuclear power from fission lie ahead. The world has a great and enduring need for energy. Uranium and thorium are the largest store of potential energy we have. The process for realizing that energy has been demonstrated on a large scale to be safe, economic and clean. Nuclear power plant wastes, curiously treated as a great problem, are one of the great merits of nuclear power. Compact and confined, their safe management is easier than for most other wastes of a modern society.

The burning question is when do we pass through the swamps and reach the ranges ahead. We are fast approaching the time when the industrialized world will need new power plants. In some areas, however, painful experiences in obtaining permission to operate nuclear power plants are current or still fresh in people’s minds. So is the economic consequence of Three Mile Island. Real interest rates are, in many areas today, twice what they were when we began to climb that first range of hills. Things are cyclical, however, and human attitudes, which are at the root of these things, can change surprisingly quickly.

Although I can’t say just when we can expect to reach the next high ground, I think we can look forward to another cycle of what I referred to in the beginning as the Second Stage - the development and widespread application of basically simple power reactors. There are ample uranium resources for a big program with simple fuel cycles, and the more efficient reactors, the breeders, are not yet ready for general application.
The Second Fifty Years of Nuclear Fission

Alvin Weinberg

Early 1944 was when the Chicago Metallurgical Project re-established contact with our colleagues from Canada. Since my job, as Eugene Wigner's assistant, was to keep track of the multiplication constant, I quickly became good friends with George Volkoff and George Placzek, who were responsible for the theoretical work at Montreal. I recall how impressed we at Chicago were with the elegance of mathematical techniques that the Montreal group had developed to solve the transport equation. By contrast, our Chicago group used simple recipes — "Fist-Formulas" as George Placzek called them — to estimate the chain-reacting properties of natural uranium "piles" moderated by graphite, heavy water, and light water. I actually spent a few weeks in Montreal about this time helping Placzek, Volkoff and their team use these formulas in the design of the lattice for the experimental heavy water reactors that were forerunners of the wonderfully successful CANDU reactors. But at that time the Montreal project was focused on plutonium, not on power.

The Chicago project by this time had pretty well completed its design of the Hanford graphite-moderated reactors, and so much of our attention was turned to future applications of nuclear fission. We organized a New Piles Committee which met weekly for three months during the spring of 1944. Here the senior luminaries, such as Fermi, Wigner, Szilard, and Franck, together with a few younger assistants like myself, discussed various ideas for reactors: for power, for submarines, for production of plutonium, even for inducing endothermic chemical reactions. Our imaginations ranged widely as we considered various moderators, coolants, and configurations. Inventing a new reactor was an everyday occurrence, simply because no one else had thought about these matters. At that time we were under the impression that uranium would always be scarce. Our thinking was therefore directed mostly to breeders: unless breeders were successful, nuclear energy would not survive very long. Of course, this all happened at least ten years before Bennett Lewis thundered in his famous paper, Breeders are not necessary, that there are ways of skinning the cat of uranium scarcity short of full-fledged breeders. This question, in 1989, still remains moot!

† [Editor's Note: Regretfully, Dr. Weinberg was prevented, at the last moment, from attending the conference. His paper was read by Malcolm Harvey.]
Although at least 20 reactor concepts have received serious consideration since 1944, only five types have become commercial power plants: light water (in both pressurized and boiling versions); heavy water (CANDU); graphite-moderated, steam-cooled (RBMK) like Chernobyl; gas-cooled graphite; and liquid-metal cooled fast breeders. The world's total nuclear electrical capacity in operation and under construction is now some 427 GW(e), or about 10% of the world's total installed electrical capacity. The 1650 terrawatt-hours of electrical energy produced last year by these reactors is more than 16% of the world's electricity, and about 6% of the world's primary energy. This widespread use of fission as a source for 16% of the world's electrical energy must be regarded as an extraordinary achievement.

Of the world's 500-odd commercial reactors, about 85% are moderated and cooled by light water. As one who was involved in the original decision to power NAUTILUS with a light water reactor, I have never outgrown my astonishment that LWR became the dominant reactor. After all, light water was chosen originally for the submarine because such reactors are compact, and at least in principle, relatively simple. They were not chosen because light water recommended itself as the best choice for generating electricity cheaply. To achieve compactness, we had to use highly enriched uranium, and at the time enriched uranium was very rare and expensive. Moreover, to retain the simplicity of its core design, the original NAUTILUS prototype simply burned U-235, without generating any new fissile material. Given our impression at the time that enriched uranium was very scarce, we could not visualize a light water reactor ever producing electricity at a competitive cost (see Table 1).

Two developments changed our perception. The first, I think largely attributable to Karl Cohen, was that gaseous diffusion, when fully rationalized, would produce enriched uranium at costs sufficiently low to allow its use in LWRs. Cohen occupied an all-but-unique position in those early days since he was probably the only person at the time who possessed an intimate knowledge of both gaseous diffusion and nuclear reactors. He had participated in the development of the original theory of the diffusion cascade at Columbia in 1942 and, when Harold Urey was banished to Chicago by General Groves in 1943, Cohen also came to Chicago to work with Wigner on the design of heavy water reactors. Cohen therefore was the first to command a detailed understanding of both reactors and diffusion plants. As matters turned out, slightly enriched uranium was eventually produced at a price that could be afforded in civilian reactors.

The second change was in the outlook for uranium ore. At the New Piles Committee, we spoke of 20,000 tons of uranium worldwide; by the late 1940's we realized that uranium was much more plentiful — as it turns out, at least 1000 times more plentiful — than we had estimated. Thus the incentive for breeding was very much diminished — a situation that persists to this day.

Was pressurized water for commercial power chosen because it was obviously the best choice? Not as I remember the matter. It was chosen for Shippingport after President Eisenhower had vetoed the Navy's proposal to build a nuclear aircraft carrier powered by a larger version of the NAUTILUS power plant. A demonstration of a power plant that would operate as part of an electrical utility was being urged by the Atomic Energy Commission. The only reactor that was on hand was the one designed for the canceled aircraft carrier —
what was more natural than to rescue Rickover’s carrier reactor by putting it on land, and operating it as part of the Duquesne Company’s grid?

Given that LWR was chosen largely as a matter of expediency, was this a bad choice? At the time, Calder Hall, a Magnox, gas-cooled power reactor was operating; the CANDU was in final design; the Soviets had operated a small version of their RBMK; and two small prototypes aimed at breeding, Zinn’s EBR-I, and Oak Ridge’s Aqueous Homogeneous Reactor, had operated. The Atomic Energy Commission hardly had a serious alternative to choose from, given its desire to demonstrate a peaceful use of nuclear power. To have diverted effort from the main light water line would have involved risks — and the arguments favouring the alternatives were not really compelling. After all, they too were expedient, not “best” choices: Calder Hall was an outgrowth of the Windscale plutonium plants, just as the Russian Obninsk was a variant of their plutonium producers, and CANDU, based on natural uranium, was an outgrowth of Canada’s NRX experimental reactor.

The two primary aims of nuclear power — inexhaustible energy, i.e. breeding — and economically competitive electricity — have both been demonstrated. The breeding ratio in France’s sodium-cooled PHENIX has been shown to be 1.13, and Admiral Rickover’s thorium-based U-233 seed-blanket light water breeder has been shown to have a breeding ratio of around 1.01. These demonstrations of actual breeding have passed rather unnoticed. I regard them as extremely important since we now can say with certainty that nuclear fission, based on breeders that burn very low grade ores, represents an all but inexhaustible source of energy. And, as I have already described, the goal of economically competitive nuclear electricity has been demonstrated in many places. What has not yet been demonstrated is electricity in a large scale breeder that is cost-competitive today. The largest breeder, the 1200 MWe Super-Phenix, is too expensive, and the same can probably be said of the light-water breeder. On the other hand, if successors to Super-Phenix could be built more cheaply — a very likely prospect — or if the life of a breeder could be extended to say 100 or 150 years (instead of the planned 30-40 years), the cost of the electricity over the entire lifetime could become competitive.

Super-Phenix has placed a cap on the cost of electricity for many future generations, if not for millenia, and this cap is surely less than twice current costs of electricity. Whether fusion or solar electricity will match this, and will eventually displace the breeder, remains to be seen. I therefore regard nuclear energy today as having all but achieved the primary goal we set for it at the New Piles Committee some 45 years ago — an inexhaustible energy source at a cost that ought not to escalate even as high grade ore is exhausted.

The Need for a Second Nuclear Era

As one of the surviving participants of the New Piles Committee, I should feel very good: our main aim, expressed 45 years ago — power breeding has been achieved! And competitive electricity from non-breeding light water reactors, a goal that we hardly considered important at the time, is a reality in many places. But, despite these extraordinary successes, nuclear energy hangs in the balance, and is all but dead in many countries. What went wrong, and how can we set it right?

Let me return to the New Piles Committee. Fermi even then seemed to sense
that nuclear energy might encounter public opposition. I quote from Ohlinger's report of the April 26, 1944 meeting of the New Piles Committee, at which Fermi outlined his ideas for a fast breeder that fed its fissile plutonium to small satellite power plants: “There may be nontechnical objections to this arrangement, for example, the shipment of Pu-239 to smaller consuming plants offers the serious hazard of its falling into the wrong hands.” And I can remember as if it were yesterday, though I cannot document it, Fermi's pronouncement at one of these meetings, that for the first time mankind would be confronted with enormous amounts of radioactivity; we must not assume that this will be accepted easily by society, he warned.

Fermi's warning is proving to be closer to the mark than the optimistic predictions of early enthusiasts such as myself. One explanation for this turn of events is that the dawning of the Nuclear Age has coincided with the dawning of what I call the "Age of Anxiety". Although life expectancy in most of the developed world has increased enormously since the turn of the century, we are much more anxious about survival — as individuals, as a society, and even as an inhabitable earth — than ever before. We worry about low levels of chemical insult, about the possibility of nuclear obliteration, about irreversible pollution of the planet. All of this is greatly exacerbated by television, and by new trends toward participatory democracy. Some of our worries are justified: nuclear war, possibly the greenhouse effect, possibility of a catastrophic accident like Chernobyl or Bhopal. Some are without scientific justification — like exaggerated claims that radiation (or many other toxins) at levels much lower than background pose any hazard to health. But in the Age of Anxiety the public does not distinguish between those worries that are real and those that are unjustified.

In the nuclear case, we must distinguish among the public's primary concerns. First, there is proliferation of nuclear weapons. I would judge this to be more of a worry for the professional arms control expert than for the public at large. I would classify this worry as justified, but as not evoking great public concern.

The second worry is radioactive waste disposal. We must remember that James Conant, President Roosevelt's personal monitor of the Manhattan Project, predicted in 1953 that nuclear power would not be worth the candle because waste disposal would prove to be intractable. The public is by and large convinced that Conant was right: wastes are the public's main concern about nuclear energy. Yet I would insist that wastes fall largely into the category of unjustified concerns. The reason is that, once the spent fuel has been removed from the reactor, and has been dispersed in a cooling pond or in a disposal cask, there is no longer enough energy being generated to cause widespread dispersal of large amounts of radioactivity. No analysis of a high level depository, including the accompanying transport system, has identified a credible accident that disperses really large amounts of radioactivity over really large amounts of land. An accident that imposes non-stochastic doses to a few people in the immediate vicinity of a depository — yes, though with small probability — or an accident that imposes very low doses on large numbers of people, these are also possible. But an accident, such as Chernobyl, that imposed large doses on large groups of people — no. And if, as I believe, the weight of evidence supports the view that doses of X-rays at less than background are hardly harmful, and may even be beneficial, I would insist that the public's concern over wastes is much exaggerated — it is a manifestation of our fearfulness in this Age of Anxiety.
Not so with the possibility of a reactor accident. As Chernobyl showed, and as Rasmussen had estimated in 1975, an un-contained meltdown of a large nuclear reactor could be a catastrophe of immense proportion. Unlike waste disposal, it can impose large doses on large numbers of people as well as contaminating large tracts of land. The nuclear community has tried to deal with this reality by invoking probabilistic arguments — the a priori probability is very small, say less than $10^{-6}$ per reactor year, or even lower for accidents as large as Chernobyl. But I don’t think the public accepts such probabilistic arguments, even when the numbers show the absolute dangers are comparable to the dangers accepted in other, more familiar technologies, such as hydro-electric dams. What the public seems to want is a return to the situation when an engineer would say “This is safe-period”, not “This is relatively safe”, or the “Probability of an accident that causes unacceptable consequences is a small number, like $10^{-4}$ per reactor-year”.

Can we develop nuclear reactors whose safety is deterministic, not probabilistic, and which, if developed, would meet the public’s yearning for assurance of safety, not simply assurance of the probability of safety? This is the task that has engaged many nuclear engineers in a search, now ten years old, for an inherently safe reactor. Now, in some sense, a device that produces 200 megawatts of afterheat and is immune from meltdown under every circumstance, conceivable and inconceivable, is a contradiction in terms. But a device whose safety depends on the working of immutable laws of nature, with a minimum of interventions either mechanical, electrical, or human — a device in short whose safety is so transparent that the skeptical elite, as well as the informed public, will regard it as safe — this I regard as eminently possible and worthwhile. There are now at least two ideas for reactors that are largely inherently safe — the SECURE-P (also known as PJUS) of the ASEA-ABB company of Switzerland and Sweden; and the Modular HTGRs of General Atomic Company in the US and of the KWU Company in Germany. Argonne is developing a metallic-fueled small breeder that embodies almost as many passively safe features as do these two. The advanced developments sponsored by Westinghouse and General Electric aim at producing relatively small reactors that are incremental improvements over existing LWRs, yet provide substantially more passive safety than do existing reactors. And here in Canada, SLOWPOKE and a power version of SLOWPOKE, provide important elements of passive safety.

I cannot say where the quest for inherently safe, or at least transparently and passively safe, reactors will end. Will it lead to a second nuclear era, risen from the ashes of Chernobyl and Three Mile Island, in which the public accepts reactors as safe — not because quantitative arguments predict a very low probability of an accident with unacceptable consequences — but because the safety of these reactors is understandable and plausible? Or will the second nuclear era dwindle in a chaos of retribution and protest because the public simply cannot be convinced, even by reactors that embody passive and inherent safety?

What the public’s reaction will be surely depends on the alternatives to, and upon the incentives for, nuclear power in the next 50 years. As for incentives, greenhouse may be the key. I have estimated that to make a dent on greenhouse, the world might have to build approximately 5000 very large reactors, producing about 300 quads of primary energy. Such a nuclearized world is impossible if the core melt probability is as high as $10^{-4}$ or even $10^{-5}$ per
reactor-year since this leads to a core melt every two to 20 years. And as Chernobyl, even TMI-2, showed a single core melt of the magnitude of Chernobyl puts an end to nuclear power in many countries. The public may regard greenhouse as worse than Chernobyl, and may not demand inherent safety as the price of a second nuclear era. I would not count on this though. I believe the nuclear community, descendants in a way of Fermi and the New Piles Committee, have no choice but to pursue these ideas for passively safe reactors.

For, in a way, there are alternatives to nuclear. There was no heavenly-ordained requirement that the age of fossil fuel be replaced by the age of fission, nor a cosmically-ordained anthropic principle that required nu to exceed 1 so that a chain reaction is possible, or exceed 2 so that breeding is possible. Had fission not been available we would, willynilly, be conserving energy at a much faster rate than now, we would be pushing fusion even harder — and as a last resort, we would turn to solar energy. A solar world in which primary energy is three times as expensive as it is today is hardly an impossible world, especially since in such a world, energy would be much more strongly conserved than it is now. We would not be relegated to a Malthusian poverty if the only reactor upon which man depended for energy was located 150 million kilometers away from the earth.

But Hahn and Strassmann’s discovery of 50 years ago, and God’s providence in adjusting the nuclear constants so as to make a power breeder practical, have given us another option. We nuclear engineers of the first nuclear era have had success, yes, with our 500 commercial reactors, and our practical breeders. But the job is only half-finished. The generation that follows us must resolve the profound technical and social questions that are convulsing nuclear energy. The challenge is clear, even the technical paths to meet the challenge are clear. All of us old-timers wish that we shall be here to see how these challenges are met, but even if we are not, we wish you well in fashioning an acceptable Second Nuclear Era!

Table 1. World Nuclear Power Plants by Type, 1988

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<th>Construction</th>
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The Growth of Nuclear Medicine

Sylvia Fedoruk

I am delighted to have been asked to participate in the 1989 Annual Conference of the Canadian Nuclear Association and the Canadian Nuclear Society and to help commemorate the discovery of fission fifty years ago. It is because of this discovery that I have had the opportunity of being associated with a part of the medical industry that has utilized fission products for the benefit of man.

Every year nearly one million nuclear medicine procedures are carried out in Canada alone, while around the world, over a half a million cancer patients are treated annually with cobalt therapy units designed and developed by AECL. Canada has played and continues to play an important role in nuclear medicine and radiation therapy. I am proud of the fact that these two disciplines occupied my full attention throughout most of my years as a student and then as medical physicist with the Saskatchewan Cancer Foundation and the University of Saskatchewan.

It is my intention today to briefly describe two early developments in nuclear medicine and then conclude with my recollection of the early days of the first two cobalt units. Time does not permit an overview of all aspects of the clinical uses of radionuclides but I do know that Dr. Wilkinson will undoubtedly do so in his paper at this conference.

I am finding the career change from medical physicist to Lieutenant Governor of Saskatchewan to be both interesting and exciting. One of the perks of office is an invitation to visit with the Queen at Buckingham Palace. Often while in Europe I have stood outside the palace gates, like every tourist, and watched cars come and go and wondered about the occupants who were lucky enough to visit the palace. Well, my turn is coming - I will be in the car and tourists will stare at me and wonder, "Now who can she be?"

If I could answer, I would tell them that I am a girl from Saskatchewan who considers herself a lucky person to have been able to study and work as a physicist.

† [Editors Note: This address by Her Honour Sylvia Fedoruk, although part of the Special Symposium, was given after lunch on June 6th. Dr. Fedoruk was introduced by Dr. Alexander Wilkinson, President, Canadian Association of Nuclear Medicine.]
during the pioneering days of the use of high energy radiation in the treatment of cancer and the use of radionuclides in the diagnosis of disease.

I was a graduate student of Dr. Harold Johns, and following graduation from the University of Saskatchewan, I became his assistant at the Saskatoon Cancer Clinic in 1951. The following year, Dr. Johns arranged for me to tour several radiological centres in the United States. During the two month tour, which incidentally cost less than a thousand dollars, I became aware of the growing interest in the use of radionuclides to diagnose disease. It was an exciting time in the world of medical physics: high energy accelerators such as the betatron were being used to treat cancer; cobalt units were just beginning to be tested clinically; scintillation crystals and photomultiplier tubes were more readily available to university laboratories, and radiopharmaceuticals were beginning to appear on the market.

During my visits with Dr. K. Corrigan at the Harper Hospital, Detroit, and Dr. G. Bromnell at Massachusetts General Hospital, Boston, I observed pioneering work in the measurement of thyroid uptakes and external localization of brain tumors. The visit to the Oak Ridge Institute of Nuclear Studies introduced me to problems being encountered with instrumentation and standardization. It was at Oak Ridge that I met Dr. Marshall Brucer who was just as interested in what we were doing in radiation therapy in Saskatoon as I was in the many interesting projects underway at the Oak Ridge Laboratory.

It was the same Marshall Brucer who made me aware of the confusion surrounding measurement of thyroid uptake - that there were some one thousand physicians throughout the world trying to do a simple procedure and most were obtaining the wrong results. It appeared to be a simple procedure - two identical aliquots of I-131 were prepared - one was given to a patient to drink and the second vial was kept as a “standard”. The patient returned twenty-four hours after drinking the atomic cocktail. An external detector was then positioned near the thyroid gland and the count-rate observed over the thyroid was compared to the count-rate recorded for the standard. In this way, the percentage of iodine taken up by the thyroid indicated whether the gland was functioning in a normal manner (20 to 40%), hyperthyroid condition (over 40%), hypothyroid (0 to 20%).

It was only after the Oak Ridge Group undertook a world wide study using a bevy of mannekins (Fig. 1 - these beauties had names such as Drusila, Hortense and Lulu) that it was realised that there was a large variation in the measurement of uptake in various labs. The mannekins were shipped to over 300 laboratories where indeed the large variation in the uptakes measured in the labs was quickly demonstrated. The Saskatoon Cancer Clinic was an active participant in the Canadian Group which included Dr. Sternberg in Montreal and Dr. Harold Johns in Toronto.

It was during the 1952 tour that I attended the annual meeting of the Radiological Society of North America in Cincinnati and listened to Dr. B. Cassen who described a new rectilinear scanning device. The machine had a scintillation detector mounted on a motor driven trolley. The trolley was driven in the X and Y directions, and when placed over the neck of a patient who had been given I-131 the scintillation counter scanned the neck, recording the amount of radiation seen by the detector in its travel. The recording was done by a mechanical stamping device activated by the
Fig. 1 Mannekins used in a worldwide study of the iodine uptake in the thyroid.

Fig. 2 One of the world's first commercial units located in Saskatchewan, to detect I-131 in the neck of a patient.

Fig. 3 An early pulse height analyser and multi-focusing collimator.

Fig. 4 The Saskatoon whole body counter.
scaling unit. The stamper recorded a larger number of strokes on a sheet of carbon-backed paper when over the area of high activity, such as the gland itself, than over the areas of low activity in the tissue surrounding the gland. Scan speed was constant and so a positive impression or scintigram was obtained of the gland itself.

The paper was the hit of the meeting because, up until the meeting, external mapping of the thyroid was carried out in a very primitive way using a hand-held detector. Within a year, the first commercial unit was on the market and we were very fortunate to obtain two such units for the Saskatchewan program (Fig. 2).

The original detector used a 1/4" x 3/4" thick scintillator and a single hole aperture. As larger crystals became available, our interest in improving sensitivity, resolution and data recording led to the acquisition of a 2" x 2" crystal, a multi-focusing collimator and the use of pulse height analysis (Fig. 3). In addition, the mapping device was replaced with a light source which exposed a sheet of X-ray film to produce photoscans - scanning of the liver in search of metatastic cancer became a required test in the Saskatchewan program.

As the crystals became large, the rectilinear scanner grew in size. This slide (Fig. 4) shows the Saskatoon whole body scanner which had a 4" x 5" NaI(Tl) crystal shielded by 5" of lead. The number of holes in the focusing collimator had been increased to 37 and raw data were stored on magnetic tape. The data were processed and images such as that of a kidney scan (Fig. 5) were produced by photographing the oscilloscope screen. An abnormality is shown in the left kidney (displayed on the right).

It was in the late 1950's that Harold Anger at UCLA reported on his development of a scintillation camera (Fig. 6). The camera used a pinhole collimator and the location of each scintillation event occurring in the single crystal was identified in a similar location on the face of an oscilloscope screen. Thus rectilinear scanning, was being replaced by electronic scanning with the patient and camera remaining stationary during the imaging procedure.

Figure 7 is a photograph of a camera that was developed in Saskatoon by T.D. Craddock as a Ph.D. project and used an 11" x 1/2" thick crystal and 19 photomultiplier tubes (Fig. 8). The camera enabled us to study the resolution and sensitivity of scintillation cameras, and reports of our work received international attention. It also enabled us to learn first hand the horrors that can occur when a NaI crystal is exposed to a sudden change in temperature during our cold Saskatchewan winters. Today equipment in Saskatoon is located in imaging rooms that have no windows which can be left open during the night.

However, by this time, commercial cameras were mushrooming on the market and the unit shown here was purchased for the Saskatoon Cancer Clinic (Fig. 9) and by the middle 1970's, rectilinear scanning was a thing of the past, having been replaced by large crystal cameras with multiaperture collimators.

The gentleman on the right in Fig. 9 is Dr. W.B. Reid, a dear friend and colleague who passed away only five weeks ago. Bill, as a graduate student along with Dr. W. Feindel and Dr. H.E. Johns, developed a unique brain scanning device (Fig. 10). It was a departure from the rectilinear approach and had two detectors that scanned the contour of the head in search of brain tumours. The head was scanned in nine arcs (Fig. 11) and the counts as recorded by
Fig. 5 The results of a kidney scan with abnormality seen in the left kidney shown on the right.

Fig. 6 Schematic of Harold Anger's scintillation camera.

Fig. 7 The scintillation camera developed by T.D. Cradduck in Saskatoon.

Fig. 8 Photomultiplier tube assembly of Cradduck's scintillation camera.
Fig. 9 Commercial scintillation camera at the Saskatoon Cancer Clinic showing W.B. Reid on the right with a patient.

Fig. 10 Brain scanning device developed by W. Feindel, H.E. Johns and W.B. Reid.

Diagram to show the crossed axes of the detectors at various arcs of the scan as seen in coronal plane.

Fig. 11 Schematic of the nine areas scanned by the device of Feindel, Johns and Reid.

Fig. 12 Images produced from a head scan with the device by Feindel, Johns and Reid.
each detection system were compared to produce an image shown in (Fig. 12) where a localization has been recorded in the right hemisphere.

There were only three of these units built and these were used successfully for about 15 years in Saskatoon, in the Montreal Neurological Institute and in Princess Margaret Hospital in Toronto. This machine was ahead of its time and never received much international attention. I often wonder what might have happened if microprocessor technology had been available in 1956 when the unit was first put into use. Today the microprocessor is a vital component in camera imaging as is evidenced in this brain tumour localization (Fig. 13).

In 1948, The University of Saskatchewan acquired a 22 MeV Allis Chalmers Betatron to do nuclear physics research and investigate its usefulness in treating deep-seated cancer (Fig. 14) and thus began the first real clinical testing of multimegavoltage therapy. The first patient was treated on 29 March 1949 and, as M.D. Schultz (A.J.R. 124: 541-549, 1975) said: "... thus started the first really concerted clinical investigation of the usefulness of multimegavoltage as a radiotherapeutic tool." In 17 years only 301 patients were treated.

It was also in 1949 that Dr. Harold Johns turned his attention to using cobalt-60 as a radiation source in teletherapy equipment. His application to NRC for the source resulted in a letter from Dr. W.B. Lewis (Fig. 15) which indicated that some doubts were being expressed about the wisdom of applying cobalt in this way for therapeutic work.

However, the application was approved and in fact three sources were placed in the reactor in 1949. From a rather shaky beginning, Canada has gone from scratch to becoming the world's largest supplier of medical isotopes. Through AECL, Canada supplies about 80% of the international market for cobalt-60. It is a Canadian story and in particular for someone from Saskatchewan, very much a Saskatchewan story.

So often many of our good ideas are exported and someone else picks up the glory. However, as far as cobalt-60 is concerned, it is a story of where we, as Canadians, have exploited original work done in Canada. The precise chronology of the development of cobalt-60 teletherapy devices is somewhat befogged by differences in how people remember things as they were. There were, however, three principal stages and sets of actors - one in the United States and two in Canada - and the Canadians prevailed.

The beginning of our story comes with Canadian nuclear research during World War II. A new reactor became operational at Chalk River and it was the only installation in the world capable of producing large quantities of radioactive cobalt. Three sources - one for the U.S. and the other two for Canada, were put into the reactor to cook in the fall of 1949. Two years later, two sources were made available by NRC's Atomic Energy Project for experimental and clinical use in Canada. The source destined to go to the U.S. was kept in the reactor and not released until the following year.

Each source, 1" in diameter, 1/2" thick, had an approximate strength of about 1000 curies. Two quite different units were designed to use these sources for radiation therapy.

This unit (Fig. 16) was designed by the development division of Eldorado
Fig. 13 A brain tumour localised by a modern day camera with microprocessor controls.

Fig. 14 The 22 MeV Allis Chalmer Betatron acquired by the University of Saskatchewan in 1948.

Fig. 15 Letter from W.B. Lewis to H.E. Johns concerning the later's application to develop Co-60 for use in therapy.

Fig. 16 Early cobalt-60 unit designed by R.F. Errington and D.I. Green from Eldorado Mining and Refining Limited. Installed in the Victoria Hospital, London, Ontario.

Dr. H. E. Johns,
Department of Physics,
University of Saskatchewan,
Saskatoon, Sask.

Dear Johns,

I have received your application for Co60 and am putting this in hand. We expect to make the irradiation and will begin as soon as possible.

I think you make a very strong case for this application in your report. As you know, doubts have been expressed in some quarters about the wisdom of applying Cobalt in this way for therapeutic work. We need not, however, at this time commit ourselves or you on the application, and I expect that in two to three years when the source is ready ideas will have cleared considerably. We can also leave the question of price to be settled then.

Yours sincerely,

W. B. Lewis

---

Dr. W. B. Lewis
23 August, 1949.
Fig. 17 Cobalt-60 unit designed by H.E. Johns and L. Bates and built by J. McKay from Acme Machine and Electric, Saskatoon. Installed in the University Hospital, Saskatoon.

Fig. 18 H.E. Johns and J. McKay loading their cobalt-60 unit in 1951.

Fig. 19 Cancer therapy unit produced by Atomic Energy of Canada Limited and installed in the University Hospital, Saskatoon in 1972.
Mining & Refining Limited. Chief designers were R.F. Errington and D.T. Green who, incidentally, was a graduate of the University of Saskatchewan. The unit was installed in the Ontario Cancer Foundation Clinic, Victoria Hospital, London, Ontario. The unit consisted of a head pivoted between the arms of a horizontal Y which could be raised and lowered. The beam was turned on and off by an air compressor that forced mercury in and out of the reservoir, the radiation beam being off when a pool of mercury was between the source and a conical opening in the head. Field size was varied by means of four lead blocks at right angles to each other.

The Saskatoon unit (Fig. 17) was designed by Harold Johns and Lloyd Bates, a graduate student. It was built by Johnny McKay, Acme Machine & Electric, Saskatoon, for the Saskatchewan Cancer Commission and was installed in University Hospital, Saskatoon. The unit consisted of a steel-encased cylinder suspended from an overhead carriage. The source was mounted on the circumference of a wheel near the centre of the head so that by rotating the wheel, the source could be brought opposite an opening through which the radiation could emerge. The Canadian sources were released in 1951. The summer and fall of 1951 became most memorable seasons that year in Saskatoon and London, Ontario. The cobalt race was on! The American source was left "cooking" in the reactor - they were left in the starting gate and the Canadians ran for the roses - Saskatoon jumped into an early lead. The Saskatoon group received the source on July 31, 1951 and the Saskatoon Star Phoenix proudly proclaimed under this photo (Fig. 18).

"Wearing specially treated smocks and masks as protection against deadly gamma rays, Dr. Johns and Mr. McKay are seen transferring the radioactive source from the shipping container to the treatment head."

The room was hardly ready for use—plaster was still going on the walls, the concrete floor was yet to be poured, but the measurements began. The Saskatoon unit was officially opened on October 23rd and the measurements continued until November 8th when the first patient was treated by Sandy Watson.

Eldorado Mining & Smelting were most anxious for as much publicity as possible. After all, they were going to market their machine. Stories began to appear in the Eastern papers — Ivan Smith quickly treated the first patient in London, Ontario. Sandy Watson played down attempts to publicize the importance of cobalt — after all, "This is not a new thing in the fight against cancer, it is but a device that may be more efficient and economical to operate". However, he did allow MacLean's magazine to feature the Saskatoon unit with a headline for the story: The Atom Bomb That Saves Lives. The chronology of events in 1951 is shown in Table 1.

Ivan Smith passed Sandy Watson in the stretch. The battle of the cobalt bomb war was fought in the editorial pages of Canada's major newspapers. On November 7th the editor of the Star-Phoenix wrote: "We hope Messrs. Truman, Stalin, Peron et al. won't think someone is trying to steal their thunder, but we think they ought to know theirs is not the only atomic race going on in the world...with all due respect to the preservation of national peace and goodwill, that is a boast which this newspaper cannot allow to go unchallenged - especially since the Free Press is brazen enough to remark that a cobalt bomb 'is also being installed in Saskatoon, Saskatchewan'. One
is indeed, or, to be more accurate, one has.”

Scientific publications followed very quickly. The first article on the physical measurements carried out on both units appeared in the December 15th, 1951 issue of “Nature” by H.E. Johns, L.M. Bates, E.R. Epp, D.V. Cormack, S.O. Fedoruk (Saskatchewan Cancer Commission and Physics Department, Univ. of Saskatoon) and A. Morrison, W.R. Dixon and C. Garrett (Radiology Laboratory, Physics Division, National Research Council). The authors, all from Saskatchewan, tentatively said that cobalt units may prove to be very convenient sources of high energy radiation for therapy. More detailed papers were published in the Journal of the Canadian Association of Radiologists in March 1952 and the British Journal of Radiology in June, 1952.

Not everyone shared Canada’s optimism — after all, the American source was still in Chalk River.

In 1952 when teletherapy was being introduced to the manufacturers of therapeutic X-ray equipment, a poll was taken on how popular the new supervoltage cobalt-60 would be. One highly respected commercial consultant didn’t think the idea was very good and predicted that in ten years there would be only thirty machines in use. By 1981, there were about 2,200 units in routine use in the free world.

The Saskatoon unit was in service until 1972 when it was replaced by an AECL progeny (Fig. 19). In all, 6,728 patients were treated with a unit which gave us 21 years of quality service. The first patient that was treated in London died a few weeks later but the first Saskatoon patient who completed her full course of treatments on November 29, 1951, was alive and well thirty-seven years later and when she was asked regarding her condition in 1988, she replied “yes, all’s well so far”.

Canadians played an important role in the development of the cobalt unit. Cobalt-60 teletherapy units were a means of making supervoltage widely available: part of its appeal was compactness. Though born of war-time nuclear research, the “bomb” was, in practice, a ploughshare rather than a sword. It was readily available, inexpensive both to purchase and maintain and, therefore, of potential value to developing countries.

Thank you for giving me an opportunity to walk down memory lane.

Table 1. First Cobalt Treatments in World, 1951

<table>
<thead>
<tr>
<th>Saskatchewan</th>
<th>Western Ontario</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cobalt-60 source delivered</td>
<td>30 July</td>
</tr>
<tr>
<td>Unit installed</td>
<td>17 Aug.</td>
</tr>
<tr>
<td>Calibration</td>
<td>11 weeks</td>
</tr>
<tr>
<td>First patient treated</td>
<td>8 Nov.</td>
</tr>
</tbody>
</table>
It remains to thank the Learned Societies for sponsoring the excellent program we have heard this morning. I would also like to acknowledge the work of Dr. Malcolm Harvey, David Cowper and their committees who organized the session and put together such a splendid roster of speakers with a uniquely qualified moderator.

I remind you that the symposium continues with the luncheon address tomorrow of Her Honour Sylvia Fedoruk who will speak not as an expert on the Lieutenant Governorship of Saskatchewan — a position to which she graduated comparatively recently — but rather as one of Canada’s leading authorities on nuclear medicine, a disciple which she helped to found a few decades ago. Although I can’t speak for the menu, I can speak for the speaker. Don’t miss this luncheon.

No history of the last five decades of nuclear technology would be complete without hearing from the man who made it happen in Canada. I speak of course of Dr. Wilfrid Bennett Lewis who would undoubtedly have been on this platform were he alive today.

I quote from a paper entitled “Socio-Cultural Evolution”, which he presented to the Seventh International Conference on the Unity of the Sciences, Boston, 1978:

“Allowing a large fraction of the world population to starve to death is not acceptable - especially when man knows that food, water and air are essential needs and that sufficient harnessed energy is the key to providing them. Looking beyond the possible loss of sufficient harnessed energy, we see that mankind can be destroyed by something much simpler to propagate - namely mistrust.”
29th Annual Conference

29e congrès annuel

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Volume 2

Tome 2

Canadian Nuclear Association

Association nucléaire Canadienne
EDITORIAL NOTE

Volume 2 is the proceedings of the sessions of the 29th Annual Conference of the Canadian Nuclear Association, June 1989.

To commemorate the 50th anniversary of the discovery of fission, the Canadian Nuclear Association and the Canadian Nuclear Society jointly sponsored a Special Symposium as the opening session of the CNA/CNS 1989 annual conferences. The papers presented at the symposium can be found in Volume 1:


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President and CEO, Newfoundland Light and Power

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SPECIAL SYMPOSIUM:
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CANADIAN NUCLEAR ASSOCIATION

ENERGY OPTIONS
AND
SUSTAINABLE DEVELOPMENT

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Chairman, Canadian Nuclear Association

MODERATOR
Tony Patterson
President, Iconics Business Systems Ltd.

PANELISTS
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Thomas E. Kierans
President Scotia McLeod Inc. and Chairman, Energy Options

Robert W. Slater
Assistant Deputy Minister, Policy, Environment Canada

Ian A. Stewart
Consultant, previously Deputy Minister EMR and Finance

(This is an edited version of the roundtable discussion)
Paul Scholfield opened this roundtable by indicating that the title, Energy Options and Sustainable Development, tends to imply there is a solution. "Certainly it's highly probable that a solution to the latter will define what options are really available. Pessimists say that sustainable development is a contradiction in terms. I've got too much confidence in our ability to solve these kinds of problems to accept that. The degradation of the biosphere is at last forcing rational debate on energy options, and that debate transcends provincial and national boundaries. The solution is going to come, and has to come, on a global basis."

Bob Slater took up the discussion with a quote from Jim MacNeill, who was secretary general of the Brundtland Commission and is now at the Institute for Research on Public Policy: "If we change the way we make decisions then we'll change the decisions we make." He then pointed to a couple of alterations to the decision-making process of the federal government.

For example, said Slater, "in the cabinet shuffle earlier this year an environment committee of cabinet was established, chaired by the minister of environment and including ministers from economic, resource development and health portfolios. This is the first time that such a committee has been institutionalized within the system.

"The minister is also on the operations committee of Priorities and Planning, which is the most central and important decision-making body for the government as a whole. We are also examining other decision-making structures within the federal government with the intention of taking measures to see that the environment is given proper consideration at all levels."

One step removed from government are the roundtables on the environment, which have been struck by Ottawa and the provinces to provide a consultative mechanism for governments as they grapple with environmental concerns.

Putting the environment front and centre for federal policy-makers and establishing a quasi-permanent consultative group are, as Slater noted, institutional experiments.

"No one is terribly clear when you create these sorts of institutions exactly what the consequences will be, what the nature and form of decisions will be as they emerge from the new process. But the decision that has been taken by the federal government is that you've got to change the decision-making process if you want to change the decisions.

"Let me put the question: what does this change federally mean for others, and particularly the private sector?"

Tom Kierans picked up the question, but cautioned "against concentrating at this point on issues of process as opposed to issues of substance." We have to start with an understanding of what our societal objectives are. Then we must negotiate the inevitable trade-offs between environmental protection and economic development.

We must select the appropriate instruments, said Kierans, and asked, "Do we wish to continue the tried and true Canadian tradition of intervention and regulatory instruments, or are we going to make room for the market here?"
"Finally the question is who pays? If we are going to use the principle 'let the polluter pay', let's understand what we're talking about. It's not a matter of the capitalist CEO going to jail. We're talking about tens of thousands of people being thrown out of particular types of jobs as offending plants are shut down. These people pay. How are we going to internalize these costs?"

Ian Stewart intervened to pose a prior question: "How do we assemble what we know already in ways that conduce to the kinds of decisions that Tom Kierans says have to be taken, whatever order one presumes they must be taken in?

"We speak of the information revolution as empowering private and special interests but we haven't yet begun to think of how to use it in our collective interest. I think there is a direct relationship between collating that information in new ways that is absorbable by the population, thereby enabling decision-making and choice, and Bob Slater's point that in this process we must surely talk about new fora, new structures of decision-making, new processes, if we are not continuously to be locked in stalemate based on insufficient information, or ideology if you like."

Replying to Kierans, Slater was not about to buy a competition between process and substance. "I find them inextricably linked. When you look at decision-making processes you must ask, are they fundamentally right in their design so that all the information necessary to make a useful decision arrives in the proper fashion at the right table with the appropriate representatives around it? What the federal government clearly concluded in the case of environmental issues was that until the cabinet committee was created it was not well equipped to take the kinds of decisions that it wanted to take for the long term.

"If we look at the private sector, I think you'll find that some commercial investment decisions have been taken and only after the fact have environmental liabilities associated with that decision come to the fore. Then, with the benefit of hindsight, one sees that an improved decision-making process would have avoided some of the pitfalls. Moreover, I believe that getting the process right makes it more likely that substantive decisions in fact do get taken, rather than having proposals punted around in an endless process of referral from one inadequate committee structure to another one."

LEADERSHIP

At this point, Kierans shifted the focus of discussion to the issue of leadership, pointing to the nuclear industry. "I and others have been quoted on numerous occasions to the effect that we regard the nuclear impasse in Canada as a problem of political leadership. The matter has been quite literally studied to death. At some point leaders must lead. We cannot resolve these problems of enormous complexity without that.

"I agree with Ian Stewart that we need new fora through which to do this, but you can have new fora and new studies and at the end of the day somebody, on behalf of society, is going to have to decide. The nuclear issue is large, but the environmental issue is
as large as God. My concern is that in the initial stages we'll have the wrong inputs, but even if Bob Slater and his colleagues get the inputs right we still will not have the political leadership we need to proceed in the most expeditious manner."

There was a day, Stewart noted, when it seemed that The Globe and Mail would run "alternating editorials crying, first, Why doesn't the government lead?, or We need leadership, and the next day asking, Where the hell did it think it got a mandate to do that?

"We have lived for 20 years in a period of growing disesteem for governments, disesteem for bureaucracies, disesteem for political process and for public policy generally. Probably most of us would agree that this has led also in a positive direction, to allow the market to do more things unrestrained and unregulated.

"Yet all of us would acknowledge, certainly the people in this room and the Canadian Nuclear Association would acknowledge, that government will continue to meddle in decisions about the future of nuclear energy, not to speak of energy use generally and its relationship to environment. It is in all of our interests that we fashion these government decisions in ways that are as wise as they can be.

"This demands an involvement on the part of all of us in thinking about what new structures for public policy decision-making we can support. It does little good to call for leadership on the one hand, while on the other continuing to engage in reactions which demean leadership when we find it and demean the public policy formulation process."

**BIAS MECHANISMS**

Moving from the sphere of policy to that of technology, Angus Bruneau submitted the notion (endorsed by Energy Options and the Canadian Council of Resource and Environment Ministers) that "technology is the fundamental and most important variable in the scheme of things as we contemplate the requirements to move toward sustainability.

"Most of us in this room today know that there are technologies available that ideally would allow us to operate in society with a vastly reduced load of harmful effluents on the environment. But if we are to see any significant contribution made by these technologies over the next three to five decades, we need to get very busy today.

"I submit this movement needs government stimulus, national and international, regulatory prodding, and the use of performance standards that are not prescriptive of solutions but that address the real goals and issues and encourage innovation. We need to introduce mechanisms today to start biasing our technological decisions towards sustainable development rather than development that can be had at the lowest commercial risk, or lowest present cost."

Tom Kierans responded, "This raises an issue which I think is crucially important. It is the role of federal and provincial
governments in basic scientific research and development. Too often people talk about the federal deficit and the impotence of the federal government in dealing with issues of this magnitude. "There are social trade-offs to be made when we take up the environmental mandate. Given that we have such a mandate, and acknowledging Angus Bruneau's view that technologies will provide us with the economically efficient means of addressing the problems, there remains a terrific role for governments. "When we were doing the Energy Options exercise we noted that the federal government had cut back on what we believed was a thoroughly acceptable and successful in-house technology program running from basic science through to applications and demonstrations. The budget had been cut back, as I recall, from $185 million to $85 million, all in the name of restraint. But it was a mindless exercise, and not what the game is supposed to be about. "There is a role for government in scientific research, always providing that the projects are refereed and open to academic and industrial infusion. If the government is seriously interested in the environment it's going to have to change its view toward funding the scientific initiatives today that will lead us toward a better future." From the floor, Ara Mooradian commented that "identifying what the government is able to do is extremely important. The international environmental question is a multi-billion dollar problem. Government policy alone cannot solve it. But where policy can, in fact, influence it, is in providing strategic investments for research, development and demonstration. There's a lesson to be learned here from the structure of atomic energy programs. In particular, Crown corporations are useful instruments for taking technology to the point of commercial demonstration. "In fact, the environment is likely to provide the biggest and most profitable business there is over the next 10 to 50 years, as long as we can resolve the questions of backyard jurisdiction." INTERNATIONAL BACKYARD It's not always easy to tell whose backyard you are operating in, as Mooradian illustrated with a story from years past when he was a member of a team trying to define sites for waste management facilities. "We were driven to remote, apparently useless areas of the country that might be suitable, and we selected some areas to look at that were at least several hundred miles into the tundra. Then we discovered that a group in Thunder Bay claimed the area as their backyard. "I don't mind people having backyards as long as they're prepared to pay for them. I once tried to build a house and found that it was in somebody else's backyard because it affected the view." The international backyard is trickier to define. "Even the process we must go through within a municipality to build an incinerator, which requires a small fortune for studies and
forever to debate, is subject to the views and often the veto of whoever's backyard it is being placed in. The neighbour with the backyard is a legitimate intervenor. But who has the legitimacy to comment on international backyard problems?

"It's a difficult question but it's crucial. The greenhouse effect is an international problem. The way to solve it is to make it pay not to generate carbon dioxide. You can't solve it by telling Brazilians that they musn't burn forests in order to grow wheat."

This led Tom Kierans to introduce the notion of environmental credits. "Canada has," he said, "a comparative advantage in energy. That's not to say that it has lower-cost energy supplies, but simply that it has, in terms of availability and reliability, a comparative international advantage. It is to be presumed, therefore, that international business which requires such an advantage will locate here. It can also be presumed that Canada ought to be in the forefront of dealing with carbon dioxide in such a manner that we could develop a store of 'credits' allotted globally for responsible prevention of carbon dioxide emissions.

"We would then be able to trade these credits. Other nations would be prepared to acquire them to make up for the fact that their economies are not developed to a point where emissions are controllable.

"The mechanisms to produce such credits would include all of the various international financing agencies, World Bank, IMF etc., and the usual trade facilities. The stick would be that the transfer of modern technology would be tied to the trading of environmental credits, and if you haven't got any you won't be able to acquire the best plant for your production.

"Therein lies the global ecosphere solution in a broad way. I allow that it is a lot easier to say than it is to do."

EARLY DAYS

"Global solutions are not likely to be simply arrived at," said Angus Bruneau, taking up the point. "We debated credits during the Energy Options exercise but came up with no immediate answer. However another perspective I like to take is that communications systems are making the globe shrink rapidly. The perceptions of intelligent people everywhere are being focussed on the same issues, making them truly global in a way that was not the case even a decade ago. I anticipate that this will have an impact on our ability to craft solutions.

"I would like to make another point, if I may. I was fascinated by a description of this meeting as marking the 50th anniversary of fission. The technology that has grown from the science of fission is what brings us together here today. This is a very recent development in the history of human technology, very recent. Indeed, we are just at the beginning of the nuclear age. We all know how little we really understand about the fundamental processes that knit nuclei together."
"We know moreover that there is a natural information system out there that for billions of years has been storing energy and genetic codes in all living things on this planet. We don't understand this system well, but we know that the sun is its principal energy source and we have begun to link our logic systems to it.

"When we learn the secrets of nuclear and other natural systems we will change the way we live and the supplies of energy and materials we consume. The learning is the important part. We must understand that investment in science and technology is really investment in people.

"We must be part of an international search for solutions, but nationally we will need to select niches. Some of these relate to our resources, our geography or other natural advantages. But more important in today's world is building on the strengths we have in science, technology and industrial capability.

"It's important to recognize that the environment in which we all live shapes the interests and goals of the scientists and engineers who work here. It spurs processes that address the problems. The real challenge that we face is to ensure that we don't lose sight of the fact that the greatest values are in the people. Continuity of support for our community of scientists and engineers is most critical."

At this point in the proceedings, the Moderator gave the floor to Ian Stewart, who had been asked to address the topic of energy and society.

"I guess," said Stewart, "I was one of those who was optimistic in the mid-70s when we, in an explosion brought on by the energy crisis of those days, made almost an order of magnitude leap in our capacity to analyze energy matters. For the first time a large proportion of the population became literate about primary energy, secondary energy, tertiary energy, energy development and use. They began to have some broad categories of numbers and shapes in their heads about how society used energy, about the efficiency opportunities in converting primary energy to various service uses, about appropriate uses for various kinds of primary energy.

PICKING THE WORLD YOU WANT

"All of that occurred during a period of four or five years from a standing start. I won't deny that the information always trailed the events, but nonetheless there was a kind of sea-change in the sophistication of people in their knowledge of energy. This laid groundwork for the conserver society notion that one could present various matrices or spreadsheets to people and say, here's the structure of primary use, here's the structure of end use, you can have that world or you can have this world. With various value systems superimposed on the spreadsheets, you could imagine a world with a population sufficiently sophisticated that voters would look at the diagrams and say, that's the kind of world I want, with these sorts of properties."
"But now the environment adds another dimension to all of that. Moreover we must repose the Canadian system in the world system, a troubled world system that is confronting the problem after 20 years or more of economic dislocation and vast structural inequities and disequilibria within the economic world, the environmental world aside.

"How can we take all of this and get our minds around it? How can we separate out the issues that the market, private economies, private technological pursuits and private exercise of energy options will sort out by price and choice alone, as opposed to those that will demand regulation and subsidy.

"The innate conservatism of scientists is increasingly troubling in this regard. We probably would be in a better world if economists had shared that conservatism, but I find the reluctance of science to speak out about what it thinks it knows until it's certain, to refuse to discuss probabilities in front of we illiterates before truth is known, is doing all of us an enormous disservice. The appropriate science is turning out to be terribly difficult to do, and meanwhile we must all wait to gain sophistication about what our environmental choices are.

"In short, it seems to me that we are talking about issues that defy imagination. We are talking about applying resources to their solution that are far short of the resources they will actually demand of us. We have scarcely begun to think the issues through, let alone devise policies that will guide the 21st century, particularly if you put them all on a world scale."

**Frightening Stuff**

Bob Slater rose to the defence of scientists. "Good science is a pretty conservative sort of business," he said. "The double checking, the triple checking, the peer review - you don't get too many cold fusion examples in what is generally described as good science. I'm not sure that means it is conservative to a fault, which was Ian Stewart's indication.

"If you look at environmental science in particular you will see that the scientific community has been pretty consistent in its forecasts on the crucial issues. The accumulation of toxic chemicals such as DDT and dioxins, the effects of acid rain, the climate change, the ozone layer - science has been at the core of public policy deliberation on environmental issues. I think a lot of it is what single discipline scientists would describe as empirical science. Nonetheless the environment has been a scientifically driven issue. A lot of the dilemma has been, not so much the science itself, but the inability to get the science explained, understood and accepted for what it says.

"What the ozone layer and the climate change scientists tell us is really quite frightening stuff. If we stop doing tomorrow what we are doing today it will still take an estimated 200 years or more before the ozone layer recovers to the state it was in during the 1970s. Even if we put our best effort into curbing greenhouse gasses, we will still have climatic change which humankind has never previously experienced, and at a pace of
change more rapid than ever before. Perhaps this information is so overwhelming that we find it easier to question the accuracy of the forecast than to accept the probabilities which scientists are revealing.

"I'm in no way trying to deny that you can rely on enlightened self-interest or market forces to do much of the correction. If we fall back on government to just the minimum extent for what must be done on the collective behalf, then that is likely to provide solutions at the least social cost. I have no problem with these arguments, but I have to point out that we have before us right now some pretty obvious examples of how difficult it is to put market forces into practice.

"The one I like best," Slater continued, "is the pricing of water. Water in Canada costs about 25% of what it costs on average in Europe and about half of what it costs on average in the United States. No one disputes that the first criterion for a mismanaged resource is to undervalue it. Water is a mismanaged resource on certainly more than that account in this country.

"We also know - and certainly you do if you live in the City of Ottawa which has taken water rate increases in the last few months of about 100% with almost no political complaint - that there is great elasticity in the price that the market will bear for water. If we charged the equivalent of the price of a bottle of beer a day per household for water, we would have more than enough funds to finance all water supply and sewage treatment systems in Canada on a self-sustaining basis.

"Is there anyone who can tell me whether this situation is one that enlightened self-interest in the private sector can be relied upon to correct, or is this the sort of thing where the heavy hand and blunt instruments of government will be the order of the day?"

**POLLUTION TICKETS**

From the floor, Colin Lennox spoke to the question of market forces. "I don't think anyone is suggesting," he said, that a business starting up, such as a pulp mill, should not be allowed to generate any pollution whatever. The suggestion that has been written about is to issue pollution tickets. Instead of saying 'nobody may pollute', let's say 'you can pollute if you buy a ticket'.

"The Department of Environment can sell these tickets, put a value on them and generate substantial revenues. When the company that buys the ticket has developed the technology necessary to abate its pollution, it can sell the ticket to another company. The government can regulate because at any time it can buy tickets off the market, thereby pushing the price up for pollution.

"It is clear that schemes like this have been debated. I would like to know what the panel can tell us about this sort of reverse sociological approach to using open market forces to generate restrictions on pollution?"

Tom Kierans replied, "I think it has a distinct advantage. First of all, society can lower the overall level of pollutants.
But more importantly what it does is internalize within the market the search for appropriate measures to reduce pollution, as opposed to prescribing them in a set of regulations or in some other arbitrary way. I think that's a very dynamic way of addressing the problem.

"Let me just add that there has been a lot of talk in the economic literature about pricing pollution and doling out pollution rights at a cost. It doesn't avoid a lot of the technical problems, and the total load for anything one is talking about, that is how many rights are for sale, has to be decided at the beginning.

"It's very instructive in looking at the international issue because it poses quite starkly the 21st century problem of how many rights the Third World countries get free while we are paying for ours, and what equitable way can we think of to divide these rights among poor and rich claimants?"

**EXAMPLES ABROAD**

"There have been a number of places where elements of that sort of approach have worked," said Bob Slater. "I have to say that none of them have worked in Canada, for reasons that aren't terribly clear. In fact, very few of them have really had any serious attention. I think that must change, but the fact of the matter is there's very little on the ground in Canada right now.

"We know of some practical applications elsewhere. In the U.S., for instance, there has been limited but successful application of trade-offs for achieving air emission standards within what they call a 'bubble'.

"France has a long history of having private sector utilities operate sewage treatment plants, where you pay the utility a fee to discharge. This has led to some very important commercial spinoffs. France now has the world's largest equipment supply company in the water and sewage treatment business. And the private sector utilities that were created for sewage have transported their management skills to other quasi-utility operations, including transportation and cable TV companies.

"Finally, in Holland over the past ten years discharge fees have been collected whether effluent goes to sewage treatment or to open water. The objective is to make sewage treatment plants financially self-supporting, and from 10%-15% cost recovery at the outset they are now up to 70%-80%. Again, one consequence has been the development of superior pollution control devices and knowledge which are in demand internationally. Canada turns out to be one of the largest customers."
SESSION 3
CANADIAN NUCLEAR ASSOCIATION
CANADIAN NUCLEAR SOCIETY

BREAKING THE PUBLIC INFORMATION BARRIER

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M. Sasaki
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Pierre Haller
Service de la Production Thermique, Electricité de France

Rita Dionne-Marsolais
Vice President, Information, Canadian Nuclear Association
PUBLIC AND MEDIA RELATIONS ON NUCLEAR ENERGY: IS IT WORTH IT?*

Carl M. Goldstein
Vice President—Media & Public Relations
U.S. Council for Energy Awareness

Ever since the organizers of this conference so kindly invited me to speak, I have been speculating about the real meaning of the presentation title suggested to me: Public and Media Relations on Nuclear Energy: Is It Worth It?

I doubt that any of us are held in suspense, waiting for my response. The answer is an unqualified and resounding YES. Nuclear press and public relations are essential, at both the national and corporate levels. So I choose to interpret the title a little differently; to me it asks, not just whether such programs are worth it, but rather, are they effective? In the final analysis, nuclear media and PR programs are worth it only if they are effective.

My perspective is that of a media specialist for an international industry association. I was privileged to be in on the start of what passed for national nuclear industry PR in the States: Nineteen-sixty-nine—a belated and incredibly modest start, as it turned out. In that year, the Atomic Industrial Forum was authorized to spend $100,000 per annum to carry the gospel to the public and the press—yes, in those days the Fourth Estate was the press, and not the media, which belonged to the art world.

With this paltry sum, and three bodies, the U.S. industry confronted a burgeoning environmental movement on the one hand, and a shrinking Government information effort on the other.

Some of you will recall that the U.S. Atomic Energy Commission had a superb and comprehensive nuclear information program. By the mid-Seventies, it had been sacrificed on the altar of political expediency, and not until the mid-Eighties did the industry's pooled commitment to various national trade and information associations come close to matching that earlier AEC effort.

As you restructure the Canadian nuclear establishment, I hope you will be spared this loss of momentum.

From the beginning, the emphasis in our program has been on responsive, candid and personal contact with energy and science writers, editors and broadcasters. If you do your job right, as I hope we are, the news media can be a powerful amplifier of your message; if you do it ineptly, then the press becomes a filter, rather than an amplifier.

Quite frankly, in dealing with the press, public relations inevitably and quite properly turn into private relations, or perhaps I should say, relationships. It's a painstaking process—rather like filling a silo with grains of truth and goodwill during fat times, so that you can draw on this reservoir when all hell breaks lose.

Emergencies, like TMI and Chernobyl gave us gratifying evidence that our reservoir was well stocked: Each of those events resulted in several thousand media contacts before they became history. And the industry's viewpoint was fairly represented—more fairly after Chernobyl than after the TMI accident, I might add, which indicates that the press had gained perspective and sophistication on things nuclear.

*Presented June 6, 1989 at the annual conference of the Canadian Nuclear Association and Canadian Nuclear Society, Ottawa, Canada
A national public-relations presence is important, also, for what might be described as optical reasons. The nuclear industry has always lacked a focal point because it cuts across so many lines of business, yet it constitutes—in my country alone—a current-worth asset of roughly $150–billion.

A national communications association gives a face and tangible presence to the disparate business interests that helped create and manage this enormous investment. Especially if it is active and visible to the public, a national advocacy group demonstrates that it represents a vibrant, optimistic industry. That is the image that USCEA tries to project.

What is a reasonable and adequate commitment to a national media and PR program? I'll duck that one, except to say that the U.S. industry has never come close to its initial target of around $40–million per year, including, of course, a substantial advertising budget.

Even at our current budget level, the question of how individual corporate members fund their support is a sensitive topic of much discussion; there's a great temptation for a chief executive officer to tap his corporate public-information program, which we believe is the wrong approach. A national nuclear PR effort should be viewed as part and parcel of the total nuclear operation, and therefore should be considered an operational expense.

A colleague of mine has suggested, only half jokingly, that a national nuclear information program, in essence, buys you insurance, and so its cost should be considered an insurance premium.

There's one more thought on this question of cost. A national program can be effective only if it is sustained and consistent. I believe the USCEA effort is a good example of both: the industry's commitment is now well into its sixth year, and our energy-independence message track has served us well in all that time.

Persistence is vital, because we're working in an extremely volatile environment. Reporters come and go at an alarming rate, and, as our polls indicate, the vast majority of the public—perhaps 90 percent—hold views on nuclear energy that are both superficial and fragile. In any six-month period, fully half of these citizens may change those views, both pro and con.

I emphasize commitment, because when times get hard, the first casualty often is support for the industry-wide effort, and the second casualty is the corporate communications program. Slighting either one of them is foolhardy, in my opinion, and has been compared to putting a snooze button on a smoke detector.

Finally, is nuclear media and public relations effective?

With just a trace of modesty, let me picture for you what the national U.S. nuclear information scene would be like without that most unique part of our program—third-party advocacy. In a word, it would be a vacuum.
In 1988 alone, we would not have had the benefit of 135 media tours by prominent specialists in many fields; we would not have seen more than 150 op-ed page commentaries and letters to the editors in publications that yielded 100-million reader impressions; we would not have seen our videotaped news features on 194 TV stations, with a viewership of 22 million, or heard our radio actuality feeds on a total of 17,000 stations. And our allied organizations, such as the American College of Nuclear Physicians, would not have been able to carry facts about nuclear energy to audiences totalling 35 million.

Not to end on too somber a note, but the absence of an umbrella industry effort would not even be the worst of it. Chances are our critics would work hard to fill the information vacuum. Let's not delude ourselves: This is no rag-tag, hand-to-mouth coalition of idealists; the critics of nuclear power are every bit as institutionalized as we are, and they are linked globally in a sophisticated, disciplined and well-funded campaign.

We have come too far even to entertain the thought that our combined efforts are not worth it. For the first time in a decade, I sense that the nuclear pendulum is swinging away from emotion and mindless obstruction toward rationality. This is precisely the time when we should all be redoubling our efforts—and then double it again!
THE SITUATION OF ANTI-NUCLEAR MOVEMENTS
AND THE CONCEPT FOR DEVELOPMENT OF PUBLIC
ACCEPTANCE IN JAPAN

M. Sasaki
Tokyo Electric Power Company

It gives me pleasure and honor to be here today to participate in the annual conference of this distinguished association, CNA.

I should like to use this opportunity to give you a brief presentation on the present state of anti-nuclear movements and the concept for the development of public acceptance strategy in Japan.

Let me first begin by outlining the current status of Japan's nuclear power generation.

There are 36 commercial nuclear reactors now in operation with a total capacity of 28,700 MW.

In 1988 alone, of all power sources available, nuclear represented 17.4 percent in terms of capacity and 26 percent in terms of electricity generated.

The capacity factor of nuclear reactors in Japan turned out to be 71.4 percent, and if we set aside the necessary inspection and refuelling periods, it is fair to say that we have been achieving safe and sound operation at near full capacity.

As for the future, a revised version of "The Long-term Program for Development and Utilization of Nuclear Energy" published by Japan Atomic Energy Commission in 1987 forecasts that nuclear power will reach 53 GW or 25 percent of generating capacity and approximately 40 percent of the total electricity generation in the year 2000, eleven years from now.

Despite a mere 2 percent of the world population, Japan accounts for roughly 5 percent of the world energy consumption. Japan, however, is a nation with scarce indigenous resources and, therefore, must rely on import for approximately 80 percent of its energy.

Given that background, Japan has undergone a process of promoting energy conservation as well as developing nuclear energy as an oil-substituting energy source with government support since the oil crisis in 1973. As a result, we have been able to install nuclear facilities that produce savings of
about 700 thousand barrels a day, which amounts to roughly 20 percent of the total oil consumption.

The oil prices have declined significantly since then, and energy supply and demand has relaxed into a stable condition. However, needless to say, not only oil but all fossil fuels represent a precious and finite energy source to every country in the world. Thus, there is a universal responsibility among our generation to help conserve as much natural resources as possible for our future generations.

Furthermore, when considering such global environmental problems as acid rain and the greenhouse effect caused by the accumulation of carbon dioxide, nuclear certainly stands as an important technologically created energy source, and we intend to promote nuclear energy, regarding it as an indispensable primary energy supply source in the 21st century.

Recent Growth of Anti-Nuclear Movements in Japan

Let me now turn to the anti-nuclear movement in Japan. The disastrous accident at Chernobyl in 1986 no doubt shocked the world deeply.

Since immediately after the accident, we in the electric utility industry have publicized not only the special circumstances of the Chernobyl reactor but also how devoted we are to ensuring safety in the designing of every stage of nuclear plants in Japan, and how all possible safety precautions are followed and how thoroughly employees are being trained in order to maintain high standards of safety. And we seemed ostensibly to have succeeded in gaining public understanding.

However, when one of the Japanese utilities planned to perform a load following operation test early last year, agitators began to campaign actively against the plan claiming that it could result in a Chernobyl-type accident. And this triggered a quick spreading of anti-nuclear movement all over Japan.

It is fair to say that even though we have experienced over the years anti-nuclear activities centering on the obstruction of construction at nuclear power plant sites, Japan has been able to promote nuclear energy development relatively free of trouble while securing the general public's support and confidence.

That no longer remains the case. The anti-nuclear activities of today seem much larger in scale, and many kinds of activities such as protest rallies, signature-collecting
campaigns and lectures are being conducted and many of these demonstrations are taking place in the cities.

Bearing in mind this new and unprecedented trend of anti-nuclear, no-more-nuclear movements, we are now promoting our public acceptance activities with a back-to-basics stance, going back to the fundamental issue of reconfirming the necessity and safety of nuclear power.

Let us now look at some of the actual PA activities conducted in Japan by taking TEPCO as an example.

The Characteristics of Anti-Nuclear Movements

I will first start by stating some of the characteristics of anti-nuclear movement.

There are dozens of points which the anti-nuclear people tend to stress, and they can be categorized into 4 major points.

No. 1 - Big doubt about safety of nuclear power generation aroused by the Chernobyl accident.

No. 2 - Fear of their life being threatened by contaminated food imports.

No. 3 - Incoherence they see between the promotion of nuclear power which they feel is not definitely safe or economical and the surplus capacity of electricity supply.

No. 4 - Mistrust in utility companies who do not make public facts and figures about power plant mishaps and troubles.

There is no need to tell you that these are groundless, incorrect arguments. Let me then explain why anti-nuclear movement did not spread right after the Chernobyl accident but with a time lag in Japan.

For one thing, over 15 years have passed since the oil crisis, and the sense of crisis due to excessive dependence on imported oil has been eroded. The oil price and supply/demand situation added to such tendency. The very fact that nuclear energy successfully achieved its role as an oil-substituting power source resulted ironically in bringing about anti-nuclear attitude.

Secondly, there are increasing opportunities for people, women especially, to express openly their perception of issues by
way of social participation. This is all due to the greater affluence and stable livelihood brought about by rapid economic growth and the resulting changes in people's values. No longer do women take economic reason and efficiency as the basis for managing the home. Greater emphasis is now placed on the pursuit of quality and safety of life against the background of advances in home electrification, information network, education and culture.

Thirdly, there seems to be an undercurrent of uneasiness and suspicion about advanced technologies such as biotechnology, computerization, space development, nuclear energy development and mega-technologies because of uncertainties about the future course of science and technology which to date has contributed to improved living of mankind.

And most importantly, utilities' system of public information on nuclear energy needs to be improved. As far as public information system on nuclear energy is concerned, we have been working at it as an important component of our effort to promote smooth nuclear power development. However, there remains much to be desired from the vantage point of providing the public with easy-to-understand and comprehensive information on aspects of nuclear energy that most people feel uneasy about.

According to an opinion poll taken by a Japanese newspaper on the question, "Do the government and utilities provide satisfactory explanation on what you want to know about nuclear power?", 76 percent answered negatively. Of course, we can attribute this to their lack of knowledge that information is available, and that we have not failed to provide information.

Whichever the case may be, we must accept the factual results humbly and incorporate them in the future planning of our action.

The Concept of Future PA Activities and Strategy for its Implementation

Let me now briefly explain what we are and will be doing to promote public acceptance under these circumstances.

Before going into this issue, one needs to identify and bear in mind the general awareness level of the public on nuclear generation as a whole.

Setting aside slight differences that may come out from different surveys, a general trend shows that approximately 80 percent of people feel uneasy about nuclear power. On the other hand, approximately 60 percent of people including those who
strongly favor nuclear do agree that nuclear energy is in fact an important energy source, but they think there still remains a lot of uncertainty prevailing and nuclear power development should be promoted with care.

Thus, central to our PA strategy is the premise that the majority view in our country is, "We are for nuclear power, but we feel uneasy about it. Nuclear, therefore, should be developed cautiously."

It goes without saying that the first step to take, fundamentally, is to strive constantly to attain an even higher level of safety and stability in operation. It is also essential to provide easy-to-comprehend information in a manner the public will be able to visualize the state of stable operation.

We have realized just how important it is to provide firsthand information to the public and the mass media such as newspaper and television as promptly and courteously as possible in the event of a failure or trouble. We are now in the process of improving this method of public announcements. Following the French example, classification of nuclear power plant accidents and troubles is being considered with the government playing a central role, and it will soon be implemented.

In order to gain the public's understanding of nuclear energy in general, we have been promoting various activities such as increasing the selection of easy-to-understand pamphlets, videotapes and books, improving the quality of speakers on nuclear energy, expanding lecture programs, while placing opinion advertisements in nation-wide newspaper and sending periodic information to opinion leaders. We must also be fully aware that the interest of the general public, especially the awareness of housewives, goes far beyond nuclear energy. We, therefore, must try to keep in mind to cover a wide range of issues from global environmental problems, future image of Japan's energy structure, to the question of energy security in the developing world.

The next task is the visualization of nuclear energy. How best can we make the invisible nuclear power and radioactivity visible? In our service area nuclear power plants typically are located far from the major energy consuming districts such as Tokyo and its vicinities. Therefore, the city residents do not see nuclear as something close to home.

In fact, we encourage opinion leaders and general public to come on plant tours inside the reactor buildings in order to gain their confidence, as our experience has proven that firsthand observation of a nuclear power plant in operation is most effective.
For those who cannot make it to a plant, we have set up close-to-life exhibits on nuclear power at electricity exhibition centers located in the city.

Furthermore, we have engaged in other activities such as displaying in the cities the readouts of radioactivity monitoring posts located around our nuclear plants, and holding mobile classes on radioactivity in vans equipped with geiger counters.

The third task is to address the mass media.

It is no exaggeration to say that the anti-nuclear movements has been amplified by the dissemination of incorrect and misleading information by the mass media at a time when the movement was rapidly gaining momentum.

We have been working hard to supply accurate information expeditiously and to correct misguided and scientifically groundless information and arguments. Fortunately, these efforts seem to have led more recently to a visible increase in opportunities for fair opinions to appear in the mass media.

The fourth task is to cope with the anti-nuclear activist groups.

The leaders of these organized opposition groups are stubbornly against nuclear energy no matter how we approach them. But there is a more likely possibility to untangle the misunderstanding on the part of the believers within the group such as housewives by providing them with correct information. We are making efforts to that end in good faith and with perseverance.

Last but not the least, confidence in nuclear power has to be founded in safe and stable power generating operation.

The recent accident involving the third reactor at TEPCO's Second Fukushima Nuclear Power Plant has significantly affected confidence in nuclear power especially in the areas around that plant. We, therefore, are all the more aware of our responsibility to maintain safe and stable operation.

I think the ground rules for gaining broad understanding of the nuclear option as an important energy source of the future is to improve public confidence in the power utilities as a responsible player in the supply of electricity, and to engage in well-balanced PA activity that fits the feelings of a broad strata of the people, and by so doing to increase the number of nuclear sympathizers.
1. THE ROLE OF PLANT VISITS IN NUCLEAR COMMUNICATIONS

It is an established fact that the nuclear industry needs to communicate with the general public. In the long run, the public will either accept the industry, or it won’t. The worldwide slowing down of nuclear programs is due essentially to problems of acceptance by national public opinion. In France, where the situation is less critical than in other countries, over 65% of the public was in favor of nuclear power in 1985. A marked change has occurred since Tchernobyl, with less than 50% in favor in 1989. This phenomenon is also partially due to an electricity surplus in France, the result, among other factors, of a greater than expected plant availability. Whatever the reason, public opinion is sensitive and characteristically volatile; it is open to considerable change at any moment, in the event of an incident in France or elsewhere in the world, for example.

A multi-pronged strategy has been implemented by the French authorities and Electricité de France to promote nuclear power. Activities aimed at national, regional and local authorities, leaders of public opinion, local associations and industries and, above all, the general public. During the development phase of the nuclear program, with the construction of 48 units totalling almost 50,000 MW by 1989, action on the national institutional framework was required to set up the necessary industrial installations. On the local level, it was particularly important to promote acceptance of new plants and population influx by emphasizing their economic value.

In the current phase of nuclear power production, we must be prepared to answer questions from the public and the media in the event of an incident and to continue the ongoing work with the public which was initiated at the very beginning of the nuclear power program.

Visits to nuclear plants are an essential part of our communications program. Logging 300,000 visitors every year, our plants have more visitors than any other industrial facility. This is comparable to the number of annual visitors to the Musée de l’Homme (Museum of Mankind) in
Paris. An estimated 5 million people have visited nuclear plants over the last 15 years - almost 10% of the population of France.

The plant visit is also a special event in qualitative terms. Much more effective than any lecture or theoretical demonstration, this concrete experience provides direct contact with the reality of nuclear power and with the men and women doing their daily work in the installations. Contact with this reality attenuates the imagined picture of nuclear power, helping to eliminate the fear of the unknown, and enabling the individual to make the transition from myths to the formation of a coherent mental picture, based on fact.

Visiting a plant is a voluntary act, freely made, which involves the visitor in the plant's activity. Involvement and coherent mental definitions are important components in any process of psychological adherence to an idea, achievements, or action. An idea becomes acceptable to the individual once he or she understands it sufficiently to discuss it. The individual also accepts an idea once he or she has performed a concrete act in relation to it, even if this is simply paying a visit to a nuclear plant. Studies into social psychology demonstrate that "individuals confirm or modify their attitudes in order to harmonize with the opinions and behavior of others" (J.L. Beauvois and R.V. Joule, La psychologie de la soumission, La Recherche, Septembre 1980).

Plant visits play a vital role in the nuclear industry's communications and should therefore be made as effective as possible. This can be achieved by applying structure and rationale, based on 15 years' practical experience, and by calling on the services of professionals in the field of communications.

THE PUBLIC...THE EXPRESSION OF A MESSAGE

Visits provide an opportunity to meet the public - the general public, schoolchildren, representatives of official organizations, specialists and laymen alike. The language we use must be adapted to each group and explanations must be accessible to everyone, so that each individual can construct that all-important coherent mental image.

The message put forward must be of the highest quality - a "four-star" discourse. The four stars are: a warm reception; an open attitude; the ability to listen; willingness to inform. These four values must underpin the message and organizational approach to visits. They will be expressed both explicitly and implicitly in the attitude of the reception staff. This is essentially a way of being.

Électricité de France is the message. The message expressed must underscore the fact that the company is working to serve the public, providing the electricity required for the welfare of all and fulfilling the economic needs of the nation under optimum safety conditions in relation to both people and the environment. Nuclear generating plants are equipped with the most ultra-modern technology available and contribute to France's independence in terms of...
energy. Nuclear power protecting the environment is a strong message which must be promoted.

In local terms, nuclear plants influence the economic development of the region in a major way. A 4 x 900 MW unit site, for example, provides FF 100 million in salaries and expenditures, FF 50 million in work undertaken by regional firms and earnings of FF 90 million in regional taxation. Nearly 50% of the site's employees would typically be drawn from the local area and a further 2,500 local jobs would be created through the activities of the nuclear generation plant.

The message is human.
It is important to create a human image of nuclear installations, which are often seen as unapproachable giants. A major problem during visits is the absence of people in the plant. The fact that visits to the control room cannot be authorized is a drawback, since here lies the very nerve center of the plant. This can be compensated for by installing cameras and broadcasting to the exterior. This problem must be dealt with by negotiation with those operating the plant.

It is important to depict those working in the plant as ordinary people, who accept their professional and family responsibilities in exactly the same way as any other citizen. The presence of women in the plant must be underscored and, if possible, increased. Whenever possible, technicians should be available to speak to the public about their work and to reinforce a reassuring image of competence. This is important for both the public and for the technicians themselves.

The men and women in the plant are also the media, in both their professional and extra-professional activities. Consciousness of this aspect should be developed.

Words carry the message.
Choosing words to describe a thing is not a neutral act. Words are imbued with connotations which shape peoples’ perception of their meaning. The technician's choice of words will not correspond to the same criteria as that of the plant's PR representative. It may be impractical to revise official jargon which has entered current usage in the industrial setting. However, in dialogue with the public, technical language should be used with the utmost care.

Acronyms and abbreviations should be avoided altogether. Even when explanations are given, acronyms are neither understood nor remembered. New concepts must be explained through analogies adapted to a general level of technical knowledge (alternator = generator for example). Designating the plant as a "Nuclear Power Center" leads people to think that the end product is nuclear production; what's more, over 30% of visitors are under the impression that they are visiting a nuclear research center. A more appropriate designation would be "electro-nuclear generation center".

It is also advisable to avoid the use of words with very negative connotations, such as "pollution, contamination, accident, aging, breakdown, waste, etc.", or at least to be extremely cautious in their application.
it is difficult to communicate the concept of "security" or "safety" to the public, since the opposing idea of "danger" is automatically inferred. A permanent discrepancy exists between the technician's rational discourse on safety and the anxiety associated with the perception of danger. For this reason, the prime argument in favor of nuclear power should not focus on safety, although it is essential to provide factual answers to all questions in this area.

In general, the public is oblivious to figures and units of measurement. Concepts such as order of magnitude and probability, which are used by technicians in the evaluation of energy requirements or the impact and threat to the environment of an installation, are difficult to communicate to a wide audience. Images are required in the discourse, for example: "Every single person in France consumes as much energy as they would if they had 200 slaves working for them. To replace the Paluel plant (4 x 1,300 MW) with windpower would require windmills every 100 meters, stretching from Dunkirk to Biarritz - a distance of 1,500 kilometers - and even then, additional means of electricity production would be required for days with no wind. The water emitted by nuclear plants is less radioactive than certain mineral waters available in any store".

The site is the message.
The places visited play a crucial role in the overall experience. These areas create an impression which will shape the mental image of the plant carried away by the visitor. It is vital to ensure that visitors know where they are and what they are seeing throughout the visit. Visitors want reassurance, they want to penetrate the mysteries of this site surrounded by barbed wire. Buildings and installations must be signposted and explained with an efficient signage system. The itinerary must be marked with arrows.

It is also important to ensure that the visitors feel welcome. The provision of refreshments, toilet facilities or a comfortable corner in which to picnic is greatly appreciated by the visitor arriving after a long journey.

Having outlined the general principles of the four-star visit, we can now go on to examine their practical application.

2. STAGES OF THE VISIT

The principle is that the visit takes place in two stages: the passage through the Information Center and the actual tour of the site.
The quality of communications during the visit will depend on three factors:

- timetable management
- site layout
- accessibility of information.

Timetable management. It is important to ensure that the various stages of the visit are balanced. Too much time is frequently allocated to films and lectures. Visitors are passive during these stages and may
become bored, however good the speakers or films may be. The public is used to television broadcasting and therefore expect a relatively rapid succession of events. The visit should also be entertaining. We therefore suggest that visits are divided as follows:

- reception: 10 minutes
- exposé + film: 30 minutes
- tour of exhibition: 20 minutes
- tour of Installations: 30 minutes
- question time in the lecture hall: 20 minutes

**total 110 minutes**

This breakdown is given as a general guideline, open to adaptation according to the situation.

Site layout. As already mentioned, “the site is the message”. The public must be made to feel “at home” in the premises during the visit. At home, through an efficient sign system, freedom of movement inside the Information Center, high quality facilities and the opportunity to leave their own mark.

Accessibility of Information. This is where the visitor feels the full impact of the four-star message. The way the visit unfolds and the site layout should produce a sense of freedom, giving visitors an opportunity to penetrate the apparently monolithic order of the electro-nuclear industry, to ask questions - possibly awkward ones, pick up a brochure from a stand, voice their opinion.

**HOW THE VISIT WILL UNFOLD**

The visit actually begins **several kilometers** from the plant. A sign with the EDF logo shows which route to take.

Near the plant, the sign become more precise, directing traffic to the parking area and Information Center. A sign details the plant’s timetable and reservation method.

In the parking lot, signposting remains consistent, with a map of the plant and description of the surrounding buildings.

In the Information Center, a tour leader will meet visitors as they arrive. A warm reception can be underscored with a luminous sign, saying, for example, “Welcome to the Oshawa Senior Citizens Club”. Cloakroom security is very important to visitors and coats and bags should be locked in cupboards.

In the lecture hall, a 10 to 15 minute exposé, adapted to each audience, will outline the role of the plant in the overall nuclear program. This is followed by a film of no more than 10 minutes duration and question time. This segment of the visit should not last for more than 45 minutes.

Visit to the exhibition, lasting approximately 20 minutes, provides an opportunity to pick up further information and consolidate the information already given.
Visit to the Installations, a tour leaflet outlines the itinerary and main technical points to be made. The route is marked by signs and technical information at key points. The tour should stop three or four times to enable the tour leader to give a short explanatory talk (3/4 minutes each).

Return to the Information Center for refreshments and on to the lecture hall for discussion and question time, distribution of reference material on the plant and an EDF bag. Visitors are invited to freely note their comments in the visitors' book.

JUNIOR SPECIAL
Children are welcome to installations once they are old enough to benefit educationally from a visit (8 to 10 years). Sufficient supervision is required to ensure that neither the children nor the installations are at risk. If children are not authorized to visit installations, the perception of danger in the public mind is reinforced. In certain instances, children's visits can be prepared in conjunction with teachers, providing an opportunity to enter into dialogue with this important profession.

The various stages of the junior special are:
- dialogue to assess the children's level + exposé
- 5 to 10 minute film on electricity with explanation
- film on nuclear power, with explanation
- presentation of scale model
- brief tour of the installations
- play atmosphere in the Information Center, cushions on the floor so that specific activities can be organized, for example:
  - organized games, answers to questionnaire
  - questionnaire count
  - walk around the Center, distribution of candies during the count
  - the three winners receive an EDF pen, the others receive stickers, a cartoon strip, etc.

3. COMMUNICATION TOOLS

Signs. This is an extremely important area, in which every effort must be made, particularly at the reception stage.

Directional signs must show the EDF logo. They require:
- setting up road signs
- signs painted on the ground at installations, for visitors' reassurance and guidance.

Information signs are of two types, comprising:
- reception signs, giving timetables and welcoming visitors
- technical information boards inside the installations.

Visitors' information pack. In addition to the usual technical information, two leaflets are distributed to visitors, one on the exhibition, the other on the tour, with a diagram of the electrical plant and a few key figures.
A good quality plastic bag, with "string" handles, featuring the EDF logo and a visual of the site, is useful to the visitor for carrying the documents during the visit and can be used again later.

Means of self-expression Each plant should be as inventive as possible, to maximize results in this area. A few suggestions:
- Luxury visitors' book for important visitors
- Information display stand
- Comments board, e.g. a word processor, providing visitors with a copy of their comments
- A questionnaire evaluating the visit.

4. TOUR LEADERS

In the communications program outlined, much of the factual and technical information is provided by signs and other documents. The role of the tour leader will therefore be to reinforce this information through discussion and answering questions.

The five tasks of the leader:
- reception
- lecture and film commentary
- visit commentary
- discussion, answering questions
- organizing children's games.

Tour leader training must be developed. This currently includes:
- basic technical training on the operation of nuclear power plants
- two training sessions in communications: hand movements, voice, eye contact, structuring discussions, visual language and variety of explanations, analysing group expectations, the ability to improvise.

Future additions:
- introduction to public relations, covering visits and the overall company message
- regular retraining (18 months).

Information update. It is vital that tour leaders are aware of current events in the field of nuclear power and in the company in general.
It is important, therefore, to ensure that they receive all internal memos concerning national or local information on the subject (press kits, internal publications, press reports, news telexes, etc.). In addition, the public relations team should compile information spanning the widest possible range of subjects, based on this material.
The team's smooth integration into the life of the nuclear plant and interaction with technicians will enable them to remain informed and motivated.

5. CONCLUSION

The answer to the question: "Are plant visits and nuclear site visits cost effective?" is undoubtedly "Yes", although no means exist to assess their
effectiveness. The role of public relations work in the wide range of social and political factors shaping public opinion on the question of nuclear power is also difficult to evaluate.

Unlike other countries, the problem of public acceptance of nuclear power in France is not serious. The policy of voluntarily providing information, in operation for over 15 years, has unquestionably contributed to this fact.

The electro-nuclear industry has opened the way for the technologies of the future. It has developed a science of safety, dealing with risks to people and the environment throughout the various stages of design, construction, production and dismantling of installations. Through in-depth study and fastidious monitoring, the impact of the industry on the environment is absolutely minimal under normal conditions and the consequences of a possible accident are strictly limited. Most importantly, the nuclear industry is one of the first to communicate with the public. It can justify its choice of technical methods. The industry is strictly monitored on a permanent basis by the Safety Authorities and therefore satisfies both their requirements and those of the public.

This forms part of the historic evolutionary process of those democratic societies benefiting from ultra-modern technology. We have a duty to explain and from this meeting between the technician and public opinion, there arises healthy debate, the airing of ideas and the realization that both parties must adopt a responsible attitude. We are aware that opposition to nuclear power is often the expression of other problems, such as the desire for recognition felt by certain social groups, or a tool employed to combat a political adversary to gain power. The technician is often placed in a difficult position in a debate where the role of the "unsaid" is so important. It is no less true, however, that clear answers must be given to the questions raised by the public in the face of increasingly complex technological developments whose implications are increasingly difficult to define.

As the mathematician Alfred N. Whitehead once said, a clash of doctrines is not a disaster, it is an opportunity.
Big question? Does it work for what? If we focus on the nuclear industry, one should ask "can public understanding be achieved with advertising"?

Research has indicated that Canadians do not know our industry, that they do not trust its people nor its products. Yet in the same breath, Canadians tell us that scientists are the most respected and trusted people in society. One cannot but notice the dichotomy here. Our industry has probably one of the highest percentages of scientists. It is quite strange that science is appreciated and its application, technology is feared. We think it is the case only with nuclear energy; I would say it is the same with any application of technology new or old.

Human nature is such that we must become comfortable with something or someone before we can learn to understand, accept or even discuss it on reasonably sensible terms. This is where we look for help in gaining this level of comfort. This is why, in a sophisticated and educated society, advertising must play an important role. Salespeople tell us that "everything is marketing". Maybe so. However, before one markets a product, someone must be receptive or alert or open-minded to receive the message. When we fear something we have barriers to it and are not receptive to the message. Think about a child who fears a cat. There is little logic here, only ignorance and instinctive fear.

Can advertising work in those circumstances? I believe it can. If we use advertising as one of the tools available in our modern society, it can help by offering a counterbalance to the occasional negative and scary media coverage of our industry. Good news is no news; all that gets reasonable coverage is generally bad news from our industry or others for that matter. If by advertising we can balance the scale of the negative impression created by news reporting and increase the level of comfort with our industry, the net result will be a greater understanding and acceptance.

That is one reason. There are others that relate to the use of advertising to convey our message. The most important one is that we live in a modern society with instant communication, and where advertising has become a need or a vehicle to bring to different publics a wide array of information. Some of this information is more essential than others and some is more urgent than others. It is important to clearly understand what
is the nature of the information we convey and how the use of the tool of advertising will support the other efforts of our overall information program, which aims at increasing the support for nuclear energy in Canada by improving its understanding by Canadians.

We believe nuclear energy is an important option for producing electricity in Canada for now and in the future. It is a clean source of electricity; it is efficient and cost effective. Our objective is to share that belief with all Canadians.

We have identified three sub-objectives and the three corresponding phases to achieve them.

(1) establish credibility and trust in our industry;
(2) establish acceptance and confidence in our products and in our people; and,
(3) increase the level of public support for the nuclear industry.

We have also identified a specific target group:

adult men and women in Canada with a particular focus on opinion leaders.

Research performed by us and by some of our members has guided us in establishing four major strategies that over a time period of 3 to 5 years could help us achieve our general objective. These strategies are:

(1) to open an on-going dialogue with the Canadian public;
(2) to introduce and familiarize the Canadian public with specific applications of nuclear energy;
(3) to present specific facts about nuclear energy; and,
(4) to monitor the results of our efforts in order to maintain and improve the on-going dialogue with the Canadian public.

These strategies are organized in a program where each element is designed to emphasize our effort and increase our chances of success. Our program is spaced out over a three-phase period which will run at least three years, and if need be, five years.
Advertising is only one of the tools we use. It is a tool that
is available to us like everyone else and it is a tool that we
use with discretion, knowledge and efficiency. Upfront we
understand that we are dealing with an issue not a product.

With this in mind, we research the Canadian public taking into
consideration the realities of market receptivity to an issue
and tolerance level in relation to advertising about issues.

Our research indicates that a non-advocacy, relatively "soft"
approach can best meet our objective and reach the Canadian
public. We want our advertising to represent our commitment
to share information with the Canadian public in the fashion
most likely to have a favourable impact.

We deal with two mainstream cultures in Canada. The message
in our advertising takes into consideration these two cultures
and presents our message in a fashion sensitive to the cultural
framework of contemporary French and English Canadian life.

In all of our advertising effort, we want to ensure that
Canadians understand that we are in the process of information
dissemination, not of advocacy. This nuance is extremely
important.

Our research indicates clearly that advocacy per se, statements
that imply the selling of a product, would not be believed and
most likely would prove counter-productive for the nuclear
industry. In fact, our research finds that people who are
modestly disposed towards the industry but have some doubts,
and those who are modestly opposed but not beyond hope have a
strong desire for more information, more data, more background
and more understanding. This has been consistent in all our
surveys.

Information is not an enemy for people who believe that the
nuclear industry has a role to play in Canada's future but
rather it is a fundamental ally. In fact, Canadians have
always believed that nuclear energy has a role to play in
Canada's future. They are perhaps more skeptical about its
role today.

As in all the components of our overall Public Information
Program, the advertising element has its own goal. It is to
establish the nuclear industry as:

- committed to the sharing of information with the Canadian
  public;
- understanding the public's requirement and desire for that information; and,
- putting forward information in a comprehensive manner so as to be a constructive participant in an on-going dialogue.

Once we agree upon a signature for our advertising effort, we pursue our choice of media. We have opted for print media and television. Magazines are chosen in preference to newspapers and other kinds of print material because of their longevity and because their format lends to greater in-depth coverage of articles and issues. The sense of urgency found in newspapers is not present in magazines. The messages that are developed for the print media concentrate on what our research indicates as being the concerns of Canadians. We want our industry to have faces, names and real people. We then move directly to a message which appeals to the general public and introduces diversity in the applications of nuclear technology and addresses the public's questions. All our print advertisements are tested with focus groups representing the public we are trying to reach.

Our television message must be by definition, quite general and is focused on general aspects of the nuclear industry. Our television message is aimed at a very wide audience and because we have to deal with certain time constraints in television advertising, it strikes many members of our industry as not being terribly complex. Like our magazine ads, our TV commercial has also been tested in great detail. From an industry point of view, we have a tendency to think that the general public is more interested in our issue then they really are and that their level of knowledge of our industry is higher then it really is. That is not to say that the Canadian public is not capable of understanding our industry, it is only to say that it is not particularly interested in our industry and in its scientific aspect. When properly introduced however, the Canadian public becomes quite interested, and in some cases, very curious about our industry.

In conclusion, I believe public confidence and acceptance can be achieved with advertising. I believe such an approach is essential in today's global village of instant communication. We provide Canadians with facts using advertising to bring these facts to their attention. This is our right as an industry. Because of the actual energy dependency of our country, where Canada ranks second in the world in electricity consumption per capita, and because Canada can no longer, as it has traditionally, rely to a
great degree on damming large rivers to produce electricity (most of our economic sites having already been harnessed), it is our duty to provide information on our industry to the Canadian public in order for them to know all the facts when they are asked to choose between different forms of energy to produce electricity. In our country, electricity production is a provincial concern. Many elements are taken into consideration when a choice to proceed with one electricity production source is taken. What we provide with our information program and particularly with the advertising dimension is knowledge, to help Canadians make more enlightened decisions. Our industry has a duty to provide that information. Our results indicate that our information is reaching more and more Canadians and where we are advertising, we are increasing their level of understanding.
SESSION 4
CANADIAN NUCLEAR ASSOCIATION

CANADIAN INDUSTRY PERSPECTIVES

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Babcock and Wilcox Canada Ltd.

SPEAKERS
A.A. Wilkinson
Department of Nuclear Medicine, University Hospital, Saskatchewan

W.A. Gatenby
Chairman, President & CEO, Cameco

R.G. Connelly
Senior Regional Director, Federal Environmental Assessment Review Office
FUTURE DIRECTIONS OF NUCLEAR MEDICINE

Dr. A. A. Wilkinson
Department of Nuclear Medicine
University Hospital
Saskatoon, Saskatchewan

In making this presentation I recognize that many of those present have little direct knowledge or experience of the field I am to discuss. Therefore, largely utilizing the contributions of the basic sciences, I will attempt to place Nuclear Medicine in its present day context and, from that, project into the future.

Nuclear Medicine is that medical specialty which utilizes radionuclides and radiopharmaceuticals in the diagnosis and treatment of disease and in research. It is multi-disciplinary with contributions from physicians, physicists, radiopharmacists, computer scientists and technologists.

Before proceeding further I should characterize nuclear medicine in practical terms. Diagnostic procedures in nuclear medicine are minimally invasive, the majority requiring a single intravenous injection of a radiopharmaceutical. The associated morbidity and mortality rates are vanishingly low and the radiation absorbed dose to the patient is acceptably low. The procedures are applied to all ages, although the incidence of disease we investigate dictates that the greater proportion of our patients are in the older age group. Therapeutic procedures, which deliver much greater absorbed doses of radiation, represent only a small fraction of total procedures at present.

Many of you from industry and research laboratories will be familiar with the radiation safety procedures applicable to the handling of open and sealed sources of radionuclides. Doubtless you have suitable protocols to manage the situation. In contrast, we deliberately introduce radionuclides into humans. The inevitable result is events at the medical/environmental interface with which you are not normally faced in industry or in the laboratory. This demands an extended form of radiation surveillance beyond the physical monitoring protocols familiar to you. It transgresses the conventional boundaries between physics and biology and should be fully understood by those who seek to regulate activities in the field of radiation surveillance.

The first use of radioactive tracers to study biological systems was by Hevesy in 1923 (1). He used an isotope of lead and followed its distribution within plants. During the course of his work, he established a principle which applies today and is the basis of
diagnostic nuclear medicine. This involved the observations that radioactive tracers may be introduced into biological systems without disturbing normal physiological processes, and that the distribution of the tracer could then be measured by external means. Work continued on biological systems, including some studies of the human thyroid gland, through the 1930s. However, it was not until after the second world war that the availability of large quantities of cheap 131 Iodine provided a major stimulus for the utilization of radionuclides in the investigation and treatment of human disease. The subsequent development, in the 1950s, of the rectilinear scanner and later the Anger (scintillation) camera enabled the distribution of radionuclides to be imaged in vivo (2,3). The more recent introduction of computer technology has permitted extensive developments in data acquisition and manipulation.

In parallel with the advances in instrumentation has been the rapid development of a wide variety of radiopharmaceuticals, suitable for use in humans. Although numerous radionuclides are used in nuclear medicine, by far the most commonly utilized is 99mTechnetium. This has a half-life of 6 hours, much too short to be clinically useful if it has to be transported a significant distance. Therefore, the development of the 99Molybdenum/99mTechnetium generator system by Strang in 1957, which made available a continuous supply of 99mTc even in the more remote locations, was a milestone in the development of nuclear medicine (4). The source of radionuclides used in medicine would be of interest to this particular audience. The lists are not exhaustive (Table I, Table II).

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<th>TABLE I</th>
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<tr>
<td>Reactor Produced Radionuclides</td>
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<tr>
<td>125 Iodine</td>
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<tr>
<td>131 Iodine</td>
</tr>
<tr>
<td>99 Molybdenum (99Mo/99mTc)</td>
</tr>
<tr>
<td>133 Xenon</td>
</tr>
<tr>
<td>51 Chromium</td>
</tr>
<tr>
<td>59 Iron</td>
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<td>60 Cobalt</td>
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In discussing future directions of Nuclear Medicine I wish to dwell mainly on those imaging techniques which are available to the vast majority of the population. When I use the word imaging it embraces all applicable forms of data acquisition and manipulation whether or not the final outcome is presented as an image or some other form of diagnostic data. However, first I would like to discuss briefly two other techniques.

Positron emission tomography (PET) involves the detection of the annihilation radiation associated with positron decay, the coincident emission of two photons of 0.511 MeV at 180 degrees to each other. The data is reconstructed to produce transverse sectional images of the radionuclide distribution. The principal radionuclides involved are 11C, 13N, 15O, 18F and 68Ga. The first four have half-lives ranging from about 2 minutes to 110 minutes which means they have to be produced on site, requiring the presence of a cyclotron. This technology is expensive in both instrumentation and manpower and therefore its distribution is restricted to several major centers. However, it has tremendous potential to lead the way in non-invasive study of a variety of pathological conditions.
Radioimmunoassay (RIA) is a well established technique widely used in medicine to measure the concentration of a variety of compounds (such as hormones) in serum. I will not go into the technique other than to say that, as the name implies, it involves immunochemical processes and competitive binding which are followed and evaluated through the detection of the emissions from trace quantities of certain radionuclides. It is an in vitro technique, that is no radioactive material is administered to the patient. In the future, RIA will be extended to include the serological evaluation of malignant disease, utilizing the detection and measurement of the levels of cell surface antigens which have been shed from malignant cells into the circulation.

Before continuing further I would like to consider certain factors which determine the "tools" we use in nuclear medicine and demonstrate its relative uniqueness in the general field of medical imaging.

Crucial to successful investigation by these techniques is the selection of the radionuclide. For diagnostic purposes, it should be a pure gamma emitter, with an energy that can be readily detected externally to the patient. Preferably it should be mono-energetic although there are data manipulation techniques emerging which may be more successful with radionuclides having two or more photopeaks. The half-life of the radionuclide should bear some relationship to the duration of the biological process being measured. There is no point in using a radionuclide with a physical half-life of a few minutes if the biological process being assessed extends over hours or days. Conversely it would be inappropriate to measure a biological process of very short duration using a radionuclide with a prolonged physical half-life (unless the biological half-life is short). If a radionuclide is to be used for therapy, the requirements are different as will be indicated later.

Nuclear medicine techniques depend mainly on detecting changes in normal physiology to demonstrate and diagnose the presence of pathology or disease. This is in contrast with other imaging modalities such as conventional diagnostic radiology, CT and ultrasound which rely largely on anatomical changes to demonstrate the presence of disease. In certain clinical situations, abnormalities can be demonstrated by nuclear medicine techniques up to several months before there is a sufficient anatomical change to demonstrate them by conventional radiology. On the other hand, the number of photons available for detection by nuclear medicine techniques are orders of magnitude less than those in diagnostic radiology and CT. Thus, nuclear medicine images do not and cannot have the spatial resolution produced by radiological techniques. This is not such a handicap as may at first appear, because the detection of physiological abnormality does not require the same detail as does the detection of anatomical abnormality. However, in assembling data for interpretation, the aim is to optimize spatial resolution, contrast resolution and temporal resolution.
Factors which affect the former two will be discussed later. In terms of temporal resolution, individual events can be timed in microseconds or less and groups of events in tens of microseconds. This far exceeds the resolution we require to follow physiological events currently amenable to investigation. However, the introduction of temporal information in certain processes permits a reduction in spatial and/or contrast resolution but at the same time considerably expands the volume of clinical information available for diagnosis. I had hoped to illustrate this by demonstrating a contracting heart in motion. Unfortunately the facilities are not available. A stationary image of the heart may provide some diagnostic information but put the heart in motion by incorporating temporal data and much more valuable information becomes available. For example we can see how well or poorly the walls or segments of the walls of the cardiac chambers are contracting. Utilizing the digital information available we can make numerical calculations to estimate how well the heart is functioning as a pump. The phase relationships of contraction in various regions of the heart can be evaluated.

We now return to considering the eternal physical problems of nuclear medicine imaging about which tremendous advances have been made but much remains to be done to approach the ideal situation.

There are two fundamental questions to be answered:

If an event is registered at a particular point in a detector, what are the probabilities it arose at various individual locations within the object?

If an event occurs at a discrete point within an object, what is the probability of it being detected, given the detector system?

These are complex questions and a partial and very workable solution has been obtained utilizing collimation and energy discrimination. In seeking answers there are a number of factors which must be considered, at the photon energies at which we predominantly operate (Table III).

<table>
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<tr>
<th>TABLE III</th>
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<tr>
<td>Compton Scatter</td>
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<tr>
<td>Photoelectric Effect</td>
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<tr>
<td>Characteristics of the Detector System</td>
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<tr>
<td>Spatial Relationships (Detector/object)</td>
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In the appropriate energy range, Compton scatter degrades all images derived from x-or gamma rays to a lesser or greater extent. Work continues on this problem in several directions. However, I would like to put before you work of Dr. Georges Dupras and his colleagues at the Montreal Institute of Cardiology and at University College Hospital Medical School, London, England (5,6). They have developed a new system which they call Holospectral imaging. Data is obtained not just from a narrow window over the photopeak but in contiguous segments throughout the energy spectrum of the radionuclide. All this data is then utilized to reconstruct an image relatively free of the effects of Compton scatter. This methodology is promising. It is now undergoing an initial multicentre trial. It appears, at least on preliminary evidence, that detectability of lesions will be considerably improved.

Absorption of photons within the patient considerably degrades the data available for processing. In addition, the photoelectric effect is more likely to involve those photons traversing considerable distances in soft tissue compared with those emitted near the surface and therefore closer to the detector. There are a number of protocols which seek to correct attenuation within the patient. None yet approaches the ideal solution but work continues.

The photoelectric effect is also important in the crystal of the detector. An increased efficiency of the detection of incident photons would yield improved diagnostic data at no increased radiation dose to the patient. The position circuitry within the detector is presumably amenable to electronic and other improvements but this is well beyond the scope of my comprehension.

We have discussed so far common situations in which the detector is in a fixed position in relationship to the object. In fact in many standard procedures in nuclear medicine, images are obtained in multiple projections relative to the patient. The spatial relationships between the detector and the patient during the acquisition of data are therefore of vital importance. Unlike conventional radiology or CT, photons in nuclear medicine procedures are emitted in 4π geometry. Thus a scintillation camera, set in a single plane will intercept relatively few of the total events. In addition, for reasons discussed earlier, objects closest to the camera will contribute more precise information than objects at some depth in the patient. In an attempt to optimize the diagnostic data this means that in conventional imaging of certain organs at least four static projections are obtained, for example anterior, posterior and both laterals. Most of you will be familiar with the concept of transmission computed tomography (CT). Utilizing an x-ray source of photons rotated 360 degrees around the patient, Housfield demonstrated that transverse sections could be reconstructed from the acquired data. This revolutionized diagnostic radiology. However, I should point out that several years prior to this Dr. David Kuhl in the United States had successfully demonstrated the same concept utilizing nuclear
medicine techniques and mapping, in transverse section, the distribution of radiopharmaceuticals within various organs. I might interject here to say that Dr. Kuhl acknowledged the contribution of the late Dr. W. B. Reid and Dr. H.E. Johns who, in 1958, demonstrated that images could be obtained other than by detector motion in a single plane \((7,8)\). They used concentric parasagittal arcs to image the brain. I mention this because their work was performed in Saskatoon. Unfortunately a suitable computer system was not available at the time otherwise who knows what strides that small group of physicists in Saskatoon would have made towards developing (emission) computed tomography. The techniques to which I am currently referring are classified as single photon emission computed tomography (SPECT), as opposed to the previously described PET.

The simplest SPECT technique utilizes the rotation of a scintillation camera through 360 degrees around the patient, acquiring data at 64 or 128 locations. This data is then reconstructed, usually by the filtered back projection techniques, to produce multiple transverse tomographic sections. It is then a simple matter to reconstruct tomographic sections in the sagittal or coronal, or any oblique plane. The technique requires the patient to lie still for a significant period of time. However, developments are now advanced, utilizing multiple detector systems, which will considerably reduce the acquisition time and enhance the quality of the data. There still remains the problem of correction for attenuation and scatter. As will be seen later, it will become particularly important to be able to measure accurately the amount of radioactivity in a particular lesion in order that valid dosimetric estimations can be made.

All that has been said previously about the actual and potential development of nuclear medicine instrumentation would be of no avail without a corresponding and appropriate development of radiopharmaceuticals specific for particular organ systems or physiological or pathological processes. Since the 1940s and 1950s, progress in radiopharmaceutical development has been such that today nuclear medicine diagnostic procedures impact across the broad spectrum of medical practice. In addition, radiopharmaceutical experiments are at the leading edge of a number of exciting areas of medical research. Historically, and even today, many of these techniques are highly sensitive but at the same time of relative low specificity; that is they are very efficient at demonstrating the presence of disease, but less capable of defining the true nature of the disease. This handicap can be considerably lessened by the proper application of interpretive techniques. However, I do not wish to discuss the complexities of medical diagnosis in this presentation.

The future in radiopharmacology demands progress on several fronts. It is necessary to refine and improve established techniques, to enhance the quality of diagnostic information. The search will continue for suitable radiopharmaceuticals with which to
investigate systems which are not currently amenable to such investigation. There needs to be developed radiopharmaceuticals more specific to particular disease processes.

Until recently, brain imaging in nuclear medicine was an indirect process and had been largely replaced by CT in many centers. This was because no radiopharmaceutical was available which crossed the blood brain barrier. Recently compounds have been developed to cross this barrier and subsequently distribute within the brain in relationship to the local blood supply. Utilizing SPECT techniques this has immediately opened a large new area of investigation of neurological and psychiatric diseases. This technique is very promising and in some clinical situations is providing diagnostic information not available using CT or nuclear magnetic resonance imaging techniques. This is just a beginning. There is considerable research being done on neuro-receptors and neurotransmitters which, if successful, will have a considerable impact on the diagnosis and treatment of neuro-psychiatric disorders. Imagine, for example, being able to diagnose schizophrenia by demonstrating an abnormal distribution of a particular neuro-receptor in the brain. Imagine then being able to follow the progress of treatment by the same imaging techniques. It is prospects such as these that are driving much of the research in this area. I should add that these methodologies are particularly applicable in an aging population where the incidence of stroke and dementia, particularly of the Alzheimer type, are increasingly common.

Another disease which is common and increases with age involves the reduction in blood supply to heart muscle leading to myocardial ischemia or myocardial infarction. Up to the present, this condition has been investigated utilizing 201 Thallium as thallous chloride. Although it meets many of the physiological requirements for such investigation, the photon energy of 201 thallium is less than optimum for producing high quality images. A class of compounds called isonitriles has been introduced and is about to enter general nuclear medicine practice in Canada. Although the physiological pathways it follows are slightly different from thallous chloride, labelling is with 99mTc which considerably improves the quality of the acquired data.

There is a considerable amount of research going on worldwide to investigate the utilization of monoclonal antibodies in diagnosis and in certain instances in therapy. The theory is that such labelled monoclonal antibodies would concentrate in pathological cells carrying specific antigens and not normal cells. Needless to say, this ideal situation has not yet been achieved. The major thrust is directed towards the diagnosis of malignant disease but applications are also evolving applicable to benign disease. Although considerable progress has been made, there are still major problems to be overcome in these techniques. It is difficult translating what can be demonstrated in tissue culture or by immunohistochemistry into the intact animal.
These are but three of the directions in which radiopharmacy is moving. Of course, there are many others.

Before closing I would like to discuss briefly the possibility of utilizing monoclonal antibodies in cancer treatment. You are probably aware that, depending on the type and distribution of the cancer, treatment may or may not include surgery, radiotherapy, chemotherapy, hormone therapy and possibly immunotherapy. I will not go into detail but several of these techniques inevitably damage normal tissues to a lesser or greater extent. The ideal situation would be to deliver a therapeutic agent directly to the malignant cells and not to normal cells. Conceptually this would be possible with monoclonal antibodies transporting appropriate radionuclides (e.g. beta emitters), chemotherapeutic agents or other cellular toxins. Such a methodology would require accurate dosimetric calculations based on SPECT techniques, to which reference was made earlier. There remain a number of problems to be solved before such therapeutic techniques enter routine medical practice. However, there is a reasonable expectation such solutions are not beyond the ingenuity of mankind.

To close, I would like to leave with you with the following thought. If I had performed this exercise each year of the last 20 years my forecasts would have been on the pessimistic side, that is the actual developments have far exceeded reasonable expectations. I optimistically expect that the predictions in this presentation will fall into the same category.
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Thank you very much for your invitation to speak here today. It is indeed an honor to participate at a conference with such learned and distinguished speakers.

I join with the Canadian Nuclear Association and the Canadian Nuclear Society in celebrating the discovery of fission, a truly remarkable exercise which has made so many achievements possible during the past 50 years.

As some of you know, I am a newcomer to your industry. I became a part of it last year when I was asked to become the Chairman of Cameco. Cameco is one of the world’s largest uranium companies and I was invited to preside over the corporation’s creation and its initial transition to the private sector.

As a rookie in the nuclear industry, I won’t try to enlighten you with my knowledge of fission. That would add another 5 to 10 seconds to a speech that is already a bit lengthy.

What I would like to bring to this conference is a different view of the nuclear industry. My perspective is not a scientific one, but rather, a business and production one, which has been shaped by a few important influences in my life.

First, it has been shaped by many years of engineering and management experience in the oil industry. Before joining Cameco, I spent 39 years in the oil business, mostly with Texaco, in Canada, the US and Venezuela.

Secondly, it has been shaped by living on the Prairies, the heart of Canada’s energy sector. There, you’ll find a wealth of resources, whether it’s oil or uranium. And there’s enough to fuel most of North America’s requirements, if needed.

We on the Prairies tend to forget this extraordinary fact about our energy capabilities. However, we are rarely allowed to forget the close relationship between our energy resources, the world’s energy marketplace, and our local economy.

To us, a $10-dollar barrel of oil means that petroleum megaprojects are delayed, or even cancelled. And that translates into a slower local economy and fewer jobs.

These days, as uranium spot prices hit record lows, we are all starting to wake up to the fact that there’s a similar cause-and-effect relationship between yellowcake that sells for less than $10 dollars a pound and cutbacks in the uranium industry such as those which we are seeing now.
The ripple effect of those cutbacks is very unpleasant, especially in Northern Saskatchewan and Northern Ontario, where our industry is a key and predominant part of the local economy.

Thirdly, my perspective has evolved from a business philosophy which is much like the football coaching philosophy of Vince Lombardi, the late, great, NFL coach and manager. Those of you who are American football fans will remember Lombardi from back in the 60s when he steered the small town Green Bay packers to five NFL championships and two Super Bowl titles.

Lombardi once told a reporter that "the only personal satisfaction that a person receives in football is being part of the successful whole." You can be the greatest player in the world, he said, "but if your team doesn't win, you're a nothing. So the only satisfaction you receive is being part of a winner."

I think Lombardi had something to tell us all. His message about winning, or just wanting to win, is an important one, whether you are talking about football, business or life.

To win, you need to have a positive outlook. Our industry has been much too negative in the past but believe me, a change is coming.

I bring to you a view of the nuclear industry which is the same as my view of any other business. It is all about winning. In my world, where I am in charge of a large company in a volatile industry, being a winner means efficiencies of production and a healthy bottom line.

At Cameco, I believe we have the making of a winner. And I would like to tell you why I think this is so.

Cameco, which is an acronym for A Canadian Mining & Energy Corporation, was formed last October from the merger of two experienced mining companies. These were Eldorado Nuclear Limited, a federal Crown corporation, and Saskatchewan Mining Development Corporation, or SMDC, which was a provincial Crown corporation.

The merger created one of the world's largest uranium companies. Cameco still has only two shareholders -- the Government of Canada and the Province of Saskatchewan -- but it will be fully privatized during the next six years through a series of share offerings.

This merger brought together a wealth of history, natural resources, skills, and potential. The number of employees from the two companies added up to more than 1,200 people. The new corporation's uranium reserves are the richest and largest in the world, totalling some 460 million pounds. That's enough to maintain the Western World's current uranium needs for the next five years. These, and other Cameco assets are pegged at more than $1.5 billion dollars.

From Eldorado, Cameco inherited 100% ownership of the Rabbit Lake uranium mine in northern Saskatchewan plus the world-class uranium reserves at Eagle Point which are located nearby.

As well, Cameco acquired Eldorado's uranium refining and conversion facilities here in Ontario at Blind River and at Port Hope. These facilities are the
newest and the most technologically advanced in the world, from both a processing and environmental protection perspective.

From SMDC, Cameco inherited a majority interest in the Key Lake mine, the world’s largest and lowest cost uranium mine. Key Lake has an annual capacity of 12 million pounds of yellowcake.

Cameco also acquired a 20% interest in the Cluff Lake uranium mine, which is operated by its major shareholder, Amok Ltd.

In addition, Cameco acquired SMDC’s interests in many gold and other precious and base metal projects.

One of our company’s most promising assets is our share (48.75%) of the uranium reserves at Cigar Lake, also in northern Saskatchewan.

Testmining is underway on this orebody which is the world’s richest uranium deposit. The average grade of the ore found at Cigar Lake is an extraordinary 12%. This quality is phenomenal when you consider that the other uranium mining operations I’ve just mentioned -- Key Lake, Rabbit Lake and Cluff Lake -- are the current world leaders using millfeed with average ore grades under 2.5%. In fact, a significant amount of the world’s uranium production is obtained from ore with grades of less than 0.5%.

To summarize, Cameco began its life with an impressive list of assets:

-- various levels of ownership in three top-class operating uranium mines in Saskatchewan

-- varying interests in at least 30 other mineral exploration projects

and

-- the most modern uranium conversion facilities in the world.

With its involvement in three operating mines, Cameco produces almost three quarters of the uranium that comes out of Saskatchewan. During 1988, Saskatchewan’s total production of 21 million pounds of U308 supplied more than 18% of the Western World’s uranium needs.

What amazes me, as a former oilman, is that all of that Saskatchewan production came from only 10 square miles of land in a northern, isolated part of the province.

And to compare uranium to the petroleum industry, in one year, Saskatchewan produced as much energy fuelled by uranium as all of Canada did in oil. For a former oilman like myself, those are amazing and not well-known statistics.

When the merger creating Cameco took place, plans called for the first share offering to be issued later this year, likely at the end of 1989. However, the continuing decline of the spot price has forced our shareholders to reconsider the timing.
That, however, is a Cameco problem. I would like to take a few moments of your time to discuss, more generally, the overall issue of privatization.

The effect of the privatization initiative underway around the world is forcing citizens across the globe to reassess some long-held beliefs.

Privatization is particularly significant in a province like Saskatchewan where public corporations have enjoyed, for the most part, a relatively high degree of public support in the past.

Cameco's management sees privatization as a very important policy which will allow our corporation to achieve its full potential. There is great promise for Cameco as a private company, for a number of reasons.

First of all, as a Crown corporation in a competitive industry regulated by government, we see it as both unfair and bad business to be part of an authority which is both player and regulator. It is especially bad when the industry is one which is constantly being put to the test of public scrutiny. We all know how difficult it is to establish public confidence in the nuclear industry.

From an investment perspective, we believe the interest in mineral development is such that, given reasonable tax and regulatory regimes, there is no need for direct government involvement in this commercial sector. And the sale of Cameco shares will generate funds which are essential for the development of new resources, such as the Cigar Lake deposit I spoke about earlier.

And, while I hesitate to speak for the governments, the sale of their interests in Cameco can generate funds which can be better applied to other government priorities. Perhaps even a little deficit reduction.

As well, we believe that broad ownership of Cameco by people and companies who choose to become shareholders will help entrench support from a diversified audience for the uranium, and hence, the nuclear industry.

As I indicated earlier, the falling spot price has had an effect on the timing of the share issue but full privatization of our company will occur by 1995.

In the meantime, we continue to adapt to the new realities which the merger and the marketplace have created.

Turning to this world marketplace, the international spot market price for uranium sits at $9.85 (US) a pound.

It hasn’t been this low since 1974 — 15 years ago — and in terms of real purchasing power, the spot price has never been this low. To compare, in 1978, it peaked at $43.40 (US) per pound.

Now normally, the industry can adjust to some fluctuations in the price. After all, contracts traditionally have been negotiated for the long term, at prices higher than the spot price of the day, but with mechanisms for adjustment, if needed. And traditionally, utility customers have been concerned about security and diversity of supply when they made their buying decisions. Today, unfortunately, it’s only price.
Also in the past five years, we have seen some strange things happen to uranium inventories. In 1985, for the first time in the history of nuclear power, consumption of uranium exceeded production. This was the start of slow drawdown of inventories held by many utilities, and this process is continuing.

You would think that depleting inventories would lead to rising prices, and many experts believed this is just what would happen by now.

However, increasing trade with Russia, as well as the emergence of China into the competitive uranium market, has slowed the pace of inventory reduction.

In addition, we are seeing some unlikely transactions between producers and utilities. Transactions such as ...

Utilities with excess inventories selling or lending uranium to producers whose production costs are above the current spot price ... or lower cost producers selling to higher cost producers ... or utilities selling or lending to other utilities.

So, inventories that were being depleted are, unfortunately, being reduced at a much slower rate than expected because of these unusual transactions and because of the new and unexpected entries on the market.

And we are suffering.

Because of Cameco’s large size and its corresponding place in the uranium industry, we can be seen as representative of the industry as a whole.

Whatever happens to us is bound to be repeated to some extent throughout the world’s uranium industry. So if Cameco is suffering -- and we’re considered to be a low cost producer -- we expect the same or worse is happening to our competitors.

I don’t believe there’s a company today that could mine uranium and make a reasonable return on investment selling it at the current spot price.

In view of the foregoing bad news, what has Cameco done with the assets it inherited eight short months ago?

Well, we have done a number of things:

To begin with, the two founding companies each had a head office -- one was here in Ottawa, the other in Saskatoon -- and these were amalgamated into one back in Saskatoon. As part of this consolidation, we are also relocating some of the Ottawa research and environment group to Saskatoon, and a smaller technology group to Port Hope.

Secondly, in March, we announced we would cut production and that almost 200 positions would be terminated -- 113 of them in Saskatchewan and 77 in Ontario.

In Saskatchewan, most of the terminated jobs are at our Rabbit Lake uranium minesite. In four weeks time, we will close the mine for at least six months, perhaps longer. Besides the obvious savings realized by stopping production, the shutdown will accomplish a number of goals.
It will cut this year’s production at the mine by 50%, and put a dent in the worldwide inventories. It will give us an opportunity to upgrade the mill and focus the remaining manpower on development work at new ore deposits located nearby.

When the Rabbit Lake mine reopens, the mill capacity will rise from 9 to 12 million pounds of uranium oxide per year, and we believe the entire operation will be more efficient.

Thirdly, we have cut production at our plants in Blind River and Port Hope. Here, as well, we are in a position to resume larger-scale production as soon as market conditions dictate.

Fourthly, we are in the process of becoming the operator of the Key Lake mine, the world’s largest, high-grade, uranium mine. This, in effect, is another consolidation. When the merger created Cameco, the new corporation found itself holding a two thirds interest in Key Lake. With such a significant stake in the mine, and considering it’s only 150 kilometres from Rabbit Lake, it makes good business sense that Cameco take over as operator.

The streamlining, amalgamation, and restructuring have produced a leaner corporation which is poised to take advantage of the anticipated upswing in the market. When it happens, we will be ready. When will it happen? We don’t know. Nor, it seems, does anyone else.

Last year, Cameco recorded a net profit of $52.8 million dollars. We will consider ourselves lucky if we make a modest profit at the end of 1989, but we can’t even rule out a break-even scenario. And that’s after considerable downsizing and reductions, in both staff and production.

What I am saying is that with these factors against us -- low spot prices, continuing over-supply -- we are looking at a very, very tough year.

Discussions will continue about nuclear energy, public perceptions, safety, mining technology and environmental impacts. But unless North American uranium companies receive more for their product than what it costs to produce, the uranium industry will have fundamental problems. Once mines or conversion plants are fully shut in, they cannot come back on stream quickly or cheaply.

However, I am optimistic things will turn around. The world’s electricity needs are increasing, fuel is required to feed those electricity demands ... and uranium is achieving some new-found respect as a clean fuel that does not contribute to acid rain or the greenhouse effect.

In fact, statements made by many respected sources reinforce this evaluation. For instance, William Reilly, who is the head of the US Environmental Protection Agency, recently made some suggestions about what should be done to limit the emissions which contribute to the greenhouse effect.

Among his recommendations, he says we should build 600 new 1,000-megawatt nuclear power plants around the world -- this from the man who is the US environmental watchdog.

Well, Mr. Reilly, new reactors are being considered, even in Saskatchewan.
Efforts are underway to promote the merits of building a CANDU-3 unit in the province. The public debate has been lively and I believe it is opening many people’s eyes to the benefits of nuclear powered electricity plants.

In addition, a six-month study will be conducted on the feasibility of using a Slowpoke reactor to heat buildings on campus at the University of Saskatchewan in Saskatoon.

These are all positive signs which add to our optimism about the future of the uranium industry.

Lastly, we like to remember that 10 years ago today, the spot price for uranium was $43 (US) and that the good times have a habit of returning.

When they do, Cameco definitely will be ready.
ENVIRONMENTAL ASSESSMENT REVIEW OF THE
CONCEPT OF DISPOSAL OF NUCLEAR FUEL WASTES
IN CANADA

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INTRODUCTION

The long-term management of nuclear fuel wastes is an important issue to
Canadians. For this reason, a public review process is being undertaken to
provide scientists and the public the opportunity to examine, and discuss
thoroughly the concept of geological disposal of nuclear fuel wastes along
with a broad range of nuclear fuel waste management issues. This review is
being conducted under the federal Environmental Assessment and Review Process
(EARP) and may span a four to five year period.

The Environmental Assessment and Review Process, established in 1973 by the
Government of Canada, created an environmental impact assessment process that
attempts to ensure that:

1. environmental effects are taken into account early in the planning of new
   federal projects, programs and activities;

2. environmental assessments are carried out for all projects which may have
   adverse effects on the environment, before commitments or irrevocable
   decisions are made;

3. the results of these assessments are used in planning, decision-making and
   implementation.

Environmental impact assessment is an important tool to implement the concept
of sustainable development as determined by the World Commission on
Environment and Development in its report entitled "Our Common Future." A
more detailed description of the federal EARP is provided in Annex I.

BACKGROUND

High-level nuclear fuel wastes in Canada are predominantly produced by CANDU
reactor operations and are contained in fuel bundles. The used fuel
discharged from reactors is presently stored in water-filled bays at each
reactor site. As early as 1958, Atomic Energy of Canada Limited (AECL)
scientists felt that permanent disposal of these fuel wastes would eventually
be necessary since this material is potentially hazardous for thousands of
years. At the same time, it was realized that long-term management of the
fuel wastes was an issue of high public interest.
Early in 1973, AECL asked the Geological Survey of Canada of the Department of Energy, Mines and Resources (EMR), to review the current status of waste disposal technology in geological formations and to recommend appropriate geological media for disposal in Canada. Subsequently, EMR recommended that geoscience investigations focus on plutonic rock found in the Precambrian Shield in Ontario. In 1978, the Federal and Ontario governments agreed that AECL would be responsible for research and development relating to immobilization and permanent disposal of fuel wastes. The bulk of Canada's nuclear fuel wastes is produced in Ontario. Ontario Hydro, a provincial utility, was to be responsible for the interim storage of these wastes, and final transportation to any permanent disposal facility.

Since 1978, as directed by the Government of Canada, AECL has been pursuing a research program on the immobilization and disposal of nuclear fuel waste, to establish the scientific basis for an assessment of the disposal concept, and to establish technical siting and design criteria for a future disposal vault. This program integrates research from the many scientific disciplines needed to assess the concept of permanent disposal, including studies in fuel immobilization, applied and fundamental chemistry and geochemistry, environmental research, environmental safety assessment and geotechnical evaluation.

Ontario Hydro has supported AECL's research and has pursued its own related studies in interim storage and transportation of high-level nuclear fuel wastes.

In April 1981, the Canadian government approved, in principle, a ten-year generic research and development program in nuclear fuel waste management. The objectives of this program were:

1. to develop and demonstrate the technology for the storage, transportation, immobilization and disposal of nuclear fuel waste;

2. to develop and demonstrate the methodology and technology to characterize and select disposal sites;

3. to assess the environmental and safety aspects of the disposal concept and

4. to develop the basis for securing acceptance of the disposal concept through scientific and regulatory review and public information, interaction and participation.

In August 1981, the Minister of Energy, Mines and Resources and the Ontario Minister of Energy issued a joint statement which set out the process by which the disposal concept will be evaluated. The process would involve:

1. a regulatory and environmental review;

2. a full-public hearing and

3. a decision by the governments involved on the acceptability of the concept based on information and recommendations coming from 1. and 2.

It was agreed then that no site selection process for a permanent disposal facility would be started until the concept had been accepted as safe, secure and desirable by both governments.
After approximately 10 years of research work, AECL has developed the disposal concept for nuclear fuel wastes to the point where it considers that the proposal is now ready for public environmental assessment and review.

This process was initiated in September 1988 by the Minister of Energy, Mines and Resources. He requested the Minister of Environment to form an Environmental Assessment Panel to undertake a review of the proposed concept of nuclear fuel waste disposal along with a broad range of nuclear fuel waste-management issues. In making this request, the Minister of Energy noted that this will be one of the most important environmental assessments ever undertaken in this country and will provide an essential foundation for future decisions on energy policy.

SCOPE OF THE REVIEW

The panel is expected to be composed of five to seven individuals knowledgeable about issues likely to be raised in the review. They will be appointed by the Minister of the Environment to conduct the public review and will report their findings to the Ministers of Environment and Energy, Mines and Resources. These individuals must have no conflict of interest upon selection as Panel members and must not place themselves in a conflict of interest position while serving on the Panel.

The Federal Environmental Assessment Review Office (FEARO) will appoint an Executive Secretary to assist the Panel and to provide it with administrative support. Environmental Assessment Panels are customarily chaired by the Executive Chairman of FEARO or his delegate. The Panel's Terms of Reference will be released by the Minister of the Environment when the Panel appointment is announced.

Most Panels also engage experts to advise them and other participants in the review of scientific and technical issues. Given the complexity of the issues associated with this review, a Scientific Review Group (SRG) of eminent, independent experts will be established and given Terms of Reference by the Panel. The Scientific Review Group may consist of as many as ten to twelve individuals recognized for their specialized expertise in subject areas relevant to the concept review. Each individual is expected to serve in his or her own capacity and not as a representative of any organization. The Scientific Review Group will conduct specific, thorough and critical examinations of the safety and acceptability of the disposal concept and provide advice to the Panel on other issues when requested. The Scientific Review Group will report its findings to the Panel for consideration and input to the public review.

Because of the great variety of scientific disciplines, and specialization within each discipline involved in assessing the concept of nuclear fuel waste disposal, the Scientific Review Group is expected to solicit advice from other sources including governments, universities and consultants.

The Panel review will be conducted in accordance with the procedures of the federal EARP and it will likely concentrate its activities in those
provinces where nuclear reactors are located (i.e. Ontario, Quebec and New Brunswick). Specifically, the Panel will review the safety and environmental impacts of the concept of geological disposal of nuclear fuel wastes along with a broad range of nuclear fuel waste management issues.

In addressing the broader issues, the review is expected to examine questions such as:

- the appropriate criteria by which the safety and acceptability of a concept for a long-term waste management should be evaluated;
- the general criteria for the management of nuclear fuel waste, as compared to those for wastes from other energy and industrial sources;
- approaches to long-term waste management, including long-term storage with a capability for continuing human intervention in the form of monitoring, retrieval, and remedial action; and the transition from storage to permanent, passive disposal;
- the degree to which we should relieve future generations of the burden of looking after the wastes;
- the social, economic and environmental implications of the nuclear fuel waste-management program;
- the use of different geological media and the experience of other countries in addressing their own nuclear fuel waste-management issues;
- the impact of recycling or other processes on the volume of waste;
- a recommended process and criteria for siting an eventual long-term fuel waste-management facility;
- the next steps to be taken with respect to the management of nuclear fuel wastes in Canada and
- the potential availability of sites in Canada and the methodology required to characterize them; and the future siting process and costs and benefits to host communities. (Since no site selection will occur until a disposal concept has been accepted as safe, no specific potential sites will be considered).

In establishing terms of reference for Panel reviews, it is customary to indicate the policy framework within which the Panel is expected to conduct its review and those issues that the Panel should not address.

These include: the energy policies of Canada and the provinces, the role of nuclear energy within these policies, including the construction, operation and safety of new or existing nuclear power plants, fuel reprocessing as an energy policy, and military applications of nuclear technology.
THE REVIEW PROCESS

The procedural steps in the review process are outlined in this section. Diagrams, illustrating the process steps are included in Annex II. Once the Panel is formed its first task will be the development of specific guidelines for the preparation of the environmental assessment documentation. This documentation will include a Concept Assessment Document (CAD) prepared by AECL as well as any other material that might be provided by government agencies or commissioned directly by the Panel. To assist in the development of the guidelines, the panel will hold a series of general public and technical meetings, known as scoping workshops (the number and locations to be determined by the Panel). An issues document is under preparation to assist participants in planning for these meetings. Participants are expected to include the general public, and organized groups, government agencies and the Scientific Review Group.

The Panel will then draft the guidelines and invite written comments by the review participants. After reviewing comments received from participants, the Panel will finalize the guidelines and distribute them along with a compendium of review comments to all the participants.

AECL, in response to the guidelines, will complete and submit the Concept Assessment Document to the Panel, who will in turn distribute the documents to the review participants and the Scientific Review Group for evaluation and comment. The Scientific Review Group will submit its review of the Concept Assessment Document to the Panel for input into the public review process.

The Panel will assess the adequacy of the Concept Assessment Document on the basis of its own review, the results of the Scientific Review Group, and the reviews submitted by public participants and government agencies. If the Panel finds that the Concept Assessment Document has adequately responded to the guidelines, it will then proceed to the public hearings stage of the review. However, if the Panel feels that some issues have not been addressed adequately, it may require AECL to supply additional information or explanation prior to the convening of the public hearings.

When the Panel considers that the information received adequately addresses the guidelines, public hearing dates and locations will be announced. The hearings offer a public forum to discuss issues of concern and to allow for supporting and opposing views on the proposal to be aired. To encourage the broadest public participation, hearings are held in locations and at times that are as convenient as possible for participants. The hearings are structured and follow pre-announced procedures but are not quasi-judicial in nature. A variety of hearings may be held. Some would allow for general discussions while others would be more technical in nature.

Following the public hearings, the Panel will review all the information received and will prepare its report for the Ministers of Environment and Energy, Mines and Resources. The report will contain a history of events associated with the concept of nuclear fuel waste disposal, an examination of environmental, safety, health and socio-economic implications of the concept,
and the Panel's conclusions and recommendations.

The Panel's report will address whether AECL's Concept for geological disposal of nuclear fuel wastes is safe and acceptable or should be modified. It will also indicate the future steps that should be taken in the management of nuclear fuel wastes in Canada.

The Panel's report is advisory to Ministers and will be made public. The Governments of Canada and Ontario will decide whether or not to accept the Panel's recommendations.
WHAT IS THE EARP?

In Canada, environmental impact assessment (EIA) is recognized as an important tool for insuring that economic development activities can occur in an ecologically and socially sound manner. It is a mechanism that permits the integration of environmental and economic considerations in decision making and it has become a major component of project planning and resource management in Canada. Basically, EIA is designed to identify, predict, interpret and communicate information about the impact of a project on human health and well-being, including the well-being of ecosystems on which human survival depends.

For more than a decade this process has been used by the Government of Canada to determine the potential environmental impacts of proposals that require a federal government decision.

The federal Environmental Assessment and Review Process (EARP) deals with the physical and biological aspects of development proposals: air, land, water, plants, animals and people. Its scope covers the potential environmental and directly related social effects of proposals; effects that could bring adverse changes to the natural environment and the effects that these changes could have on people. The scope of a public review may be extended by the ministers concerned to cover the broader socio-economic effects, assessment of technology, the need for the proposal, or other relevant issues.

Over its fifteen year history, EARP has been in continuous evolution. Changing priorities of governments and increasing public demand for an accessible and credible forum for addressing the environmental effects and related socio-economic effects of development proposals, particularly those of major large-scale undertakings, have been the stimulus for that constant evolution. The government is currently engaged in re-examining its process, with an eye to improving it. One of the conclusions reached to date is that the greatest need for change is at the initial assessment phase of EARP, where a need for a greater rigour in procedures and more accessibility by the public is apparent.

Canada was the second country in the world after the United States to adopt an administrative mechanism to conduct environmental impact assessment. Unlike the United States, Canada chose not to legislate its process. Many countries and Canadian provinces modelled their process on the Canadian EARP but chose to enshrine it in legislation. Canada is currently considering the possibility of legislating its process, including certain changes resulting from the ongoing public examination of EARP.

EARP was initially established by a decision of the federal Cabinet in 1973 and adjusted by Cabinet decision in 1977. On June 22, 1984, the process was
strengthened and updated with the issuance of the Environmental Assessment and Review Process Guidelines by an Order in Council, an administrative order made by the Cabinet under the authority of the Government Organization Act, 1979.

This Guidelines Order is now the authority for the process. It reaffirms those aspects of the original policy and procedures that proved their worth and incorporates others that came about through evolution. Roles and responsibilities are more precisely defined and public participation is reconfirmed as an essential element of the process from beginning to end.

EARP is a planning, rather than a regulatory, process. It is a planning tool intended to help administrators make good decisions, just as economic and engineering studies are planning tools. It helps to ensure that Canada's resources are not inadvertently wasted or irretrievably lost through lack of awareness or poor planning.

The Guidelines Order applies to all departments, boards and agencies of the Government of Canada. Parent corporations (mostly former proprietary Crown Corporations) are expected to make EARP a part of their corporate policies, unless this is not possible under their legislation.

A department with the authority to make a decision about a development proposal is called the initiating department or initiator, while the organization (or the initiating department itself) that intends to undertake the proposal is called the proponent.

WHEN IS EARP USED?

EARP is used when a department:

° intends to undertake any proposal of its own; or
° has the authority to make a decision about a proposal of another organization (private or public corporation) that:

-- might have an environmental effect on an area of federal government responsibility;
-- would require federal government financial commitment, or
-- would be undertaken on lands administered by the federal government, including the offshore.

Departments are also expected to ensure that Canadian activities do not bring about adverse effects in other countries, including those benefiting from foreign aid.

HOW DOES IT WORK?

Initial Assessment

Initial assessment is the first step in the process, encompassing everything a
Department does to determine what potential adverse environmental effects a proposal may have. It begins with a screening: an assessment of potential environmental effects and public concerns carried out by the department that has decision-making authority for the proposal being examined (see figure 1). Initial assessment may lead to an additional detailed study called an initial environmental evaluation (IEE).

After a proposal has undergone initial assessment, it will either proceed, be abandoned, or be referred for review by a panel.

This phase of the process has been often criticized because it is not implemented consistently throughout the government. Furthermore, individual government departments' decisions are not subject to public input. Suggestions for reform concerning the initial assessment phase include creation of a list of projects which should be automatically subjected to an initial environmental evaluation (IEE) with an opportunity for the public to request a public hearing should there be dissatisfaction with the decision taken as a result of the IEE. Suggestions have also been made to include, at this stage, opportunities for the public to provide comments and concerns regarding the proposal under review by the initiating agency.

**PANEL REVIEWS**

When an initial assessment leads to the decision that a proposal's potentially adverse environmental and directly related social effects are significant, or that public concern is such that a public review is desirable, the minister of the initiating department refers the proposal to the Minister of the Environment for a public review by an environmental assessment panel.

Public reviews of proposals may differ in type and focus, but two characteristics are always present: the proposal undergoes detailed examination by an independent panel of experts, and there is opportunity for public involvement, including participation in public hearings.

Each panel has a specific mandate, describing the nature and scope of the review, which is set out in the terms of reference issued by the Minister of the Environment.

The nature of the proposal and the scope of the review will be specified in the terms of reference. Most often, a specific project, is carefully examined and the panel eventually recommends whether the project should proceed, and if so, under what conditions. In some cases, when the Cabinet decides in advance that the proposal must proceed in the national interest, a panel review results in terms and conditions for the project rather than a decision on whether it should proceed.

**THE PANEL**

Panel members are appointed by the Minister of the Environment for the duration of the panel review. Anyone can be chosen, provided certain requirements for objectivity and competence are met. Members must be free of potential conflict of interest or political influence, and have special
knowledge or relevant experience that is useful for reviewing the anticipated effects.

Normally, a panel is chaired by the Executive Chairman of the Federal Environmental Assessment Review Office (FEARO), or his delegate. When there is a joint review with a province or territory, the panel may be co-chaired by persons appointed by the two jurisdictions. Panels are supported by a secretariat from the FEARO staff. Administrative and financial arrangements for panels are managed by FEARO.

In 1987 a Study Group, composed of a retired judge, a lawyer and a university professor, was appointed to review the procedures used by panels for their public reviews and recommend whether these procedures should be formalized to become quasi-judicial. On the subject of panel members the Study Group report reaffirmed the principles stated above including the following recommendations:

"Members of environmental assessment panels must be unbiased with respect to the proposal being reviewed and yet collectively have special expertise related to the proposal; they must be able to function as members of an interdisciplinary panel; they must understand and respect the purpose of the review process generally and the hearing process specifically; and they must be able to get the necessary information from the public hearings.

Panel members must be independent of the federal government and of the proponent. Neither the proponent nor a special interest group should have the right to a representation on the panel."

PANEL PROCEDURES AND ACTIVITIES

Each panel establishes and publishes its own operating procedures, based on FEARO's Procedures and Rules for Public Meetings.

FEARO's procedures and rules help ensure policy and procedural consistency between reviews. They may be modified by FEARO for a federal-provincial review; or in special circumstances, as for example, when the Office negotiates provincial or territorial participation in a review, federal participation in a provincial review, or participation in any other cooperative study of a proposal.

To conduct reviews, panels must have appropriate information upon which to focus. Typically, this begins with an environmental impact statement (EIS) describing the proposal and its potential effects.

A panel may issue guidelines for the preparation of an EIS by the proponent and hold public meetings beforehand to determine the scope and relative importance of issues to be covered by these guidelines.

Throughout the review, the panel secretariat disseminates information about panel activities and the review process. This is done through personal contact, letters, press releases, advertisements, libraries and local
information centres. The public are encouraged to contact the secretariat for information and to participate in the public meetings. The Study Group on Hearing Procedures mentioned earlier also recommended 13 general principles of ethics that should be adopted de facto by any panel conducting a hearing under EARP.

THE ENVIRONMENTAL IMPACT STATEMENT

All essential elements of a proposal are contained in one document, usually an environmental impact statement (EIS), which provides the focus of the public review.

An EIS generally describes the proposal; shows the need for the development under consideration and states alternatives; describes the present environment, resource use, and social patterns; predicts potential impacts; and indicates how the adverse impacts will be mitigated or avoided altogether. The EIS states where the proposed development will occur, how long it will last, how it can be carried out and the preferred way to do this so as to minimize any potential adverse impacts during construction.

The EIS is submitted to the panel and made public. Indeed, all material submitted to a panel during this, or any other stage of the review, becomes public information. The panel also allows sufficient time for review participants to examine and comment on the information received before it holds public hearings.

If the information in an EIS is adequate, a panel goes ahead with its public hearings. If it is considered deficient, the panel requests more information and the hearings are delayed until the material is received and reviewed.

THE PUBLIC HEARINGS

Public hearings, held by panels, fall into two categories:

* special meetings seek public input on issues requiring further study during the review, and receive comments on draft guidelines for the preparation of an EIS;

* final hearings provide the principal forum for public comment on the proposal and assist the panel in the eventual preparation of its report.

The hearings offer a public forum for supporting and opposing views of the proposal. To encourage the broadest public participation, hearings are as informal and flexible as practicable and are held in the area affected by the project. No one can be subpoenaed to appear before the panel or asked to take an oath. There is no cross-examination in the legal sense and no need to be accompanied by legal counsel. However, the panel may question the relevancy and content of any information submitted to it. The Study Group on Hearing Procedures however, recommended that panels be given the power to subpoena in
order to provide them with better access to information they may need if such information was known to be withheld. The Study Group also recommended against cross-examination in the legal sense as it recognized panel members' ability to question information being presented to them.

Participation in the hearings by both the experts and the public is vital to the review. For, while a panel needs technical and scientific analyses from experts, it also needs to hear from people who could be affected by the proposal, particularly those who live near the proposed site. Although an impact may not be significant to the "experts", it may be so for people living and working near the site. Local residents may have information and insights not available to an outsider. Recognizing the importance of public participation in environmental assessment and more specifically public hearings, the Study Group on Hearing Procedures recommended against the use of judicial procedures which would turn a hearing into a trial and therefore would reduce if not eliminate spontaneous public participation. In fact the Group stated that "a public hearing is not a privilege granted to the population, but a service requested of the public by the government to help it make a better decision and to favour a harmonious relationship between economic development and environmental protection".

THE PANEL REPORT

When the public hearings are completed, the panel writes a report for the Minister of the Environment and the minister of the initiating department or, in the case of joint reviews, for additional ministers or organizations that may also be involved. A panel's report is advisory; the ministers make the final decisions.

A report usually contains:
- a brief description of the proposal;
- the characteristics of the proposed site;
- the potential impacts;
- comments, issues and analysis; and
- conclusions and recommendations.

RECOMMENDATIONS

It is the responsibility of the two federal ministers receiving the report to make it public.

The initiator decides to what extent panel recommendations must be adopted before the proposal can proceed. These are incorporated into the design, construction, and operation of the proposal. The initiator must see to it that decisions on suitable implementation, mitigation measures, inspection, and monitoring programs are carried out.

The proponent must make certain that any post-assessment monitoring, surveillance, and reporting, laid down as conditions for proceeding with the proposal, are undertaken.

Decisions stemming from the panel's recommendations are made public. The initiator decides how this is done.
Figure 1  ENVIRONMENTAL ASSESSMENT AND REVIEW PROCESS

Lead Responsibility
For Decisions and
Providing Information
to the Public

MINISTER OF THE ENVIRONMENT
(PUBLIC REVIEW PHASE)

RESPONSIBILITIES AND PROCEDURES VARY WITH CIRCUMSTANCES
(PUBLIC REVIEW PHASE)

PANEL
(PUBLIC REVIEW PHASE)

INITIATING DEPARTMENT
(IMPLEMENTATION PHASE)

AUTOMATIC EXCLUSION
MISCELLANEOUS ADVERSE EFFECTS
ADVERSE EFFECTS MITIGABLE
ABILITY TO MITIGATE UNKNOWN
SIGNIFICANT ADVERSE EFFECTS
SIGNIFICANT PUBLIC CONCERN
AUTOMATIC REJECTION

PROJECT PROCEEDS WITH ANY NECESSARY MITIGATION AND FOLLOW-UP
PROJECT ABANDONED OR POSTPONED
PROJECT PROCESSES BUT WITH MODIFICATIONS
PROJECT PROCEEDS

TERMS OF REFERENCE PREPARED
PANEL FORMED
GUIDELINES FOR EIA PREPARED
ENVIRONMENTAL ASSESSMENT DOCUMENTS PREPARED
PUBLIC REVIEW
PANEL REPORTS RECOMMENDATIONS TO ENVIRONMENT AND INITIATING MINISTERS

PROPOSALS TO BE SCREENED

NO SIGNIFICANT EFFECTS
UNACCEPTABLE EFFECTS

INVESTIGATION TO ADDRESS UNKNOWNS RESULTING IN INITIAL ENVIRONMENTAL EVALUATION

POTENTIAL SIGNIFICANT EFFECTS

INITIAL ASSESSMENT PHASE
INITIAL ASSESSMENT PHASE

PROJECT PROCEEDS WITH ANY NECESSARY MITIGATION AND FOLLOW-UP
PROJECT ABANDONED OR POSTPONED
MODIFY AND RESCREEN

MINISTRY OF THE ENVIRONMENT
(PUBLIC REVIEW PHASE)

RESPONSIBILITIES AND PROCEDURES VARY WITH CIRCUMSTANCES
(PUBLIC REVIEW PHASE)

PANEL
(PUBLIC REVIEW PHASE)

INITIATING DEPARTMENT
(IMPLEMENTATION PHASE)
ANNEX II

CONCEPT REVIEW
OF
NUCLEAR FUEL WASTE MANAGEMENT

LIST OF CONTENTS

- LETTER OF REFERRAL
- OUTLINES SCOPE OF REVIEW
- PANEL APPOINTMENT AND TERMS OF REFERENCE ISSUED BY MINISTER OF ENVIRONMENT
- APPOINTMENT OF SCIENTIFIC REVIEW GROUP (SRG) BY PANEL
- SRG MEMBERS HAVE SPECIALIZED EXPERTISE IN RELEVANT SUBJECT AREAS
SCOPING WORKSHOPS

- General Public and Technical Meetings
- Review Participants (Government Agencies, SRG and Public) assist in identifying issues for Concept Assessment Document (CAD) and other Review Documents

DRAFT REVIEW DOCUMENTATION GUIDELINES

- Panel prepares review documentation guidelines
- Written comments by review participants invited
CONT'D

FINAL REVIEW
DOCUMENTATION GUIDELINES
AND
COMPRENDIUM OF COMMENTS

- PANEL ISSUES GUIDELINES TO AECL
- PANEL DISTRIBUTES GUIDELINES AND
  COMPRENDIUM TO REVIEW PARTICIPANTS

DOCUMENTATION REVIEW

- AECL SUBMITS CAD TO PANEL
- PANEL DISTRIBUTES DOCUMENTATION
  TO REVIEW PARTICIPANTS FOR
  COMMENT
- SRG SUBMITS REPORT TO PANEL FOR
  DISTRIBUTION TO REVIEW PARTICIPANTS
CONT'D

Panel Assessment of Adequacy of Doc'n

YES

- Panel requests additional information

NO

AECL Response

- Panel distributes response to review participants

Review of Response

- Panel and review participants evaluate response

If additional information is adequate, panel announces public hearings

Public Hearings

- General, technical and community sessions
PANEL REPORT

CONT'D

• PANEL SUBMITS CONCLUSIONS AND RECOMMENDATIONS TO MINISTERS OF ENVIRONMENT AND EMR

DECISION

• MINISTER OF EMR COORDINATES THE GOVERNMENT'S RESPONSE TO THE PANEL REPORT

• GOVERNMENT'S RESPONSE MADE PUBLIC
SESSION 5
CANADIAN NUCLEAR ASSOCIATION

NUCLEAR ENERGY: THE NEXT FIFTY YEARS

SESSION CHAIRMAN

J.M. Campbell
Donlee Precision

SPEAKERS

Jean-Claude Charrault
Head, Nuclear Policy Division, Directorate General for Energy,
Commission of the European Communities

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John C. DeVine, Jr.
Electric Power Research Institute, USA

Kirk R. Smith
Environment and Policy Institute, East-West Center, Hawaii

O.J.C. Runnalls
Chairman, Centre for Nuclear Engineering and
Professor of Energy Studies, University of Toronto
THE FUTURE FOR NUCLEAR POWER IN EUROPE

Jean-Claude Charrault
Head, Nuclear Policy Division
Directorate-General for Energy
Commission of the European Communities

If you browse through the lines of novels in your local bookshop, you will notice an ever growing proportion of "sagas". These usually follow one particular family - or a specific individual - for a considerable period of time. They usually comprise more than one book with the three book series or "trilogy" being the most popular. I would like to follow fashion by splitting my presentation into three parts or periods. However, in this case I will be trying to predict what could happen rather than describing events which have already occurred. It may turn out to be a true story.

The first part of my trilogy - like all such stories - will set the scene and introduce the main characters as they prepare themselves to meet the challenges the future will put in their way. We then will meet our hero again during the first decade of the next century, but rather than describe the path he takes we will consider what alternative ways he could follow. The future is not yet clearly defined. After another interval we come to the rousing finale when, after battling against great odds and triumphing over adversity, our hero looks forward to a long and happy life. The "hero" is, of course, nuclear power.

Book One - Now and the next few years.

At the present time, nuclear power accounts for some 35 % of electricity production in the European Community. It is the biggest primary energy source used for that purpose, ahead of hard coal (admittedly by a narrow margin). Hence its important role within the Community.

However, as you know, the Community is made up of twelve Member States. Five Member States of the Community (Denmark, Greece, Ireland, Luxembourg and Portugal) have, in practice, decided from the start not to take up the nuclear option. These countries include nearly all the smaller Member States - a fact which could well have had some influence on their decisions. The electrical output of these States represents less than 6 percent of the output of the Community as a whole and some of them import nuclear electricity. The situation in the other Member States could be described briefly as follows:

In Belgium it is now clear that there will be no early decision in favour of building another nuclear station (NB). The utilities recent plan included the construction of an additional nuclear power plant, but the government requested additional economic studies and favors in principle development scenarios which do not call upon more nuclear energy. As is the case in the Netherlands, gas appears to be the preferred option for domestically generated electricity, though importing electricity is an alternative solution. The existing nuclear power stations contributed 65.5% of the total electricity production in 1988.

In view of the presently growing reserve margin available in France, the government has stretched its ordering schedule, committing to one unit every 18 - or more - months. This rate of ordering is thought to be the
minimum necessary to maintain the economical viability and technical expertise of the French manufacturing industry. It will likely be stepped up again in a few years when the reserve margin disappears or it is justified by more long-term electricity export contracts. The 55 units coupled to the grid have a combined capacity of 53 GWe. A further 8 units are under construction (PWR-1300s and PWR-1450s). A load following system has been developed so that nuclear units will no longer be operated solely for baseload supply. It is worth noting that French generating costs support the construction of nuclear units over conventional plants when the plants are used for more than 2000 hours per year (i.e. utilization factor greater than 25%).

In Germany, nuclear power also produced 39% of the electricity in 1988. Only one reactor - the SNR-300 FBR - is under construction (its completion has been delayed on account of legal problems) and there is a general consensus that there will be no immediate follow-up to three nuclear units of the convoy currently entering commercial operation. No new nuclear capacity additions are expected during the next ten years. The choice of the new investment necessary to meet the demand after 1995 is unclear. However, the Government of the Federal Republic has, on several recent occasions, reiterated its position in favour of the continued use of nuclear energy.

In Italy, after Chernobyl, nuclear energy became a hotly debated political issue. Following the referendum of November 1987, a policy calling for a halt of 5 years on nuclear power stations construction has been established. The existing nuclear power stations of Trlno and Caorso are presently shut down and it is now planned to convert the plant under construction at Montalto to burn fossil fuels. (Meanwhile over 25 TWh of electricity will continue to be imported each year - mainly of nuclear origin. This represents around 15% of the country's needs.)

In the Netherlands, the government and Parliament decided after Chernobyl that it was necessary to reconsider the use of nuclear energy and to carry out first a number of studies. The studies have now been published and independent advice is being sought. Thereafter the government will formulate a standpoint on the construction of new nuclear power plants. This will then be discussed with the Parliament. In the meantime, rational use of energy is emphasised in electricity planning, imports of (nuclear) electricity will continue (3.6 TWh in 1987) and the use of natural gas will probably be increased.

In Spain, the Trillo I plant was brought into commercial operation in August 1988. As a result the net electricity production from nuclear plants increased by 22% over 1987. In total nuclear contributed nearly 50,000 GWh of electricity in 1988 - representing 39% of all electricity generated. The average load factor of the Spanish stations was nearly 80%. However, there are presently five nuclear stations on which work was started but has since been halted. There appears to be little possibility of work restarting on two of the stations (León and Valdeballvros) and the second plant at Trillo may be required to cover some of the 7 - 7.5 GWe capacity gap that could open between 1995 and the end of the century.

In the United Kingdom, Sizewell B is under construction and the CEGB has applied to the Secretary of State for Energy for consent to construct the second PWR power station at Hinkley Point. Plans have been announced for two further PWRs, one at Wylfa and a second at Sizewell (Sizewell C). The privatisation of the electricity industry will bring a new working
environment for the CEGB's successors and it is presently unclear whether
the new organisation will continue to stimulate the development of nuclear
power.

On the whole, the nuclear capacity in the Community will continue to grow
over the next years from the current level of nearly 100 GWe although at a
reduced rate. A total of about 115 GWe can be expected for 2000.

In 1985, the Commission published its "Illustrative Nuclear Programme for
the Community" in which it proposed industrial and strategic objectives for
the development of nuclear energy and made a number of recommendations both
to Member States and to companies involved in the nuclear industry. This
document continues to be the basis for the nuclear policy of the
Commission.

The PINC - as it is known - set out two objectives, one concerning the
nuclear share in electricity generation and the other relating to the fast
reactor, and a number of recommendations and observations. For the moment I
will leave aside the matter of the fast reactor - I will return to it later
- and deal with the nuclear share and its implications.

In a section of the PINC covering the outlook to the year 2000, the report
stated that by 1990 nuclear energy "should account for about 35% of
electricity production in the Community" though it should have done better.
The reasons for it not doing better were "uncertainties in the demand for
energy and owing to difficulties of various origins, particularly the
acceptability of nuclear energy to the public and the conflict between the
powers of local authorities and national authorities. Moreover, in certain
cases, priority was given to using domestic sources of fossil fuels".

It went on to say that "In view of the existence of such difficulties, care
must be taken not to set too optimistic objectives. It is for this reason
that the Commission proposes the adoption of the following lines of
development for nuclear energy:

I) to produce about 40% of Community electricity in 1995, and

II) subsequently to increase its share in electricity production
to 50% around the turn of the century."

While we are well on target for the 35% in 1990, we calculated that the
Community (which was then 10 Member States) would require 120 GWe of
nuclear capacity by 1995 to meet the objective. However, even with the
addition of the nuclear reactors in Spain (following the enlargement of the
Community to twelve Member States) we do not now expect the installed
nuclear capacity to exceed 107 GWe. The main reason for the difference
between our latest and our earliest prediction results from the change in
the Italian programme, where nearly 11 GWe had been planned by 1995. There
have been other less significant changes (in Belgium and the Netherlands)
also as a result of Chernobyl. As, at the same time, electricity demand is
also growing quicker than we expected, it is likely that the share of
nuclear energy in electricity production will tend to stabilize and will
probably not easily reach the level of 40% by 1995 - nor the 50% by the
end of the century.

Does this then mean that we start our story with gloom and despondency over
nuclear apparent failure to meet its objectives? I think not. Some people
characterise the next decade as "lean years" for nuclear power. To some
extent this is true - especially if we compare it to the "years of plenty"
of the past decade and more. An alternative view would be to look on them as growth in a different direction. As children grow older they increase most quickly in size in their earlier years. This growth often occurs in spurts and is followed by periods when very little upward growth occurs. However, just because their size is not increasing during these periods it certainly does not mean that they are not "growing up". On the contrary, they are learning every day new skills, increasing in experience and maturity and becoming more and more integrated into society.

Over the next ten years while nuclear power's "upward growth" in the Community may be relatively slow it will be growing up in other ways. While we may not be bringing many new reactors on line, we will be opening other new nuclear facilities. At the front end of the fuel cycle Spain will increase its uranium production capability, several Member states will increase their capacities for fabricating mixed-oxide (MOX) fuel, including the construction of the Franco-Belgian MELOX facility at Marcoule in France to produce 100 to 120 tonnes per year of MOX fuel by around 1993. URENCO expects to more than double its gas centrifuge enrichment capacity and there is continuing work on isotope enrichment using lasers (SILVA or AVLIS). At the back end of the fuel cycle there will be major expansions to reprocessing capacity in France and the UK as new plants come on stream in the early 1990s. By the end of the century, our reprocessing capacity for light water reactor fuel in the Community will be around seven times what it is today.

Our capacities for storing spent fuel will also be increasing and work will be well advanced on long-term storage facilities. Finally, as our Member States will start disposing of high level radioactive wastes soon after the year 2000, a Community programme of research and development work on disposal concepts for different geological formations is underway. Site characterization – including, in some countries, in situ characterization – will continue apace. There is no doubt that a demonstration that HLW can be disposed of safely will do much to relieve the public's concerns – however irrational – over this aspect of nuclear power. Finally, a great deal of experience will be gained in the field of decommissioning nuclear facilities as some of the older reactors in France, Germany and the UK are taken out of service.

Therefore, while the number of nuclear reactors may not be increasing, there is a continuing very substantial investment in nuclear power in the Community and a wealth of experience being gained. While our hero may not be growing taller he is certainly building his muscles and his mind!

So what about nuclear's 40% share in electricity generation? Must we shrug our shoulders and hope for better in the future? Maybe not!

In the PINC the Commission encouraged "the acquisition by electricity producers of holdings in nuclear power stations installed in neighbouring countries". One of the benefits of this "cross-frontier acquisition" would be the that it gave "the electricity producers the opportunity to obtain greater benefits from having an interconnected grid".

This text was written over one year before the adoption by the Member States of what is known as the "Single European Act" in (December 1985) by which the completion of the internal market became the central priority for the Community.

In June 1987, the Council of Energy Ministers agreed that the Commission should compile a list of obstacles to an internal market for energy. Based
on this inventory a report, the "Internal Energy Market", was prepared which not only lists the obstacles but also outlines the main priorities in each of the different energy sectors which should be addressed in order to achieve the opening of the market. This report was approved by the Energy Council in November 1988.

Regarding nuclear energy, the report found very few obstacles to an open market. The one priority area where improvements could be made was identified as in the market for equipment and components. We found that too many of our Member States were rather "nationalistic" when it came to ordering nuclear plants. This was not at all desirable – especially when large production overcapacities meant that rationalisation and possible greater integration of the Community's nuclear industries would be the better route to follow. One problem that would need to be faced and overcome in the context of this rationalisation could take place is ensuring that the different construction standards are compatible. In other words we have to get our act together!

This however, will take a little time and will not affect our achievement of the 1995 40% objective. For this we must look more closely at what the report on the Internal Market for Energy says about the electricity sector, in particular "Interconnection". It found that the internal high-voltage system interconnection is highly developed within national boundries, that - whatever the ownership of the network - its operation is generally managed on a national basis and that although some systems have a certain degree of "common carriage" obligations all systems are effectively concessionary monopolies. The report goes on to say "there is no obligation to use international interconnections, although it is true that such interconnections are used to achieve better economy and security of the national systems". It also states that interconnection may provide more economic and secure supplies to the distribution utilities on a more Community-wide basis. There are also arguments in favour of interconnection systems on a "common carrier" basis the benefits of which are said to include the ability of consumers (or distribution utilities) to obtain supplies from any suitable low-cost production source.

The main contribution to be made by nuclear power is to "base-load" electricity generation. This field represents more than half the Community's electricity production. However, even at the presently depressed fossil fuel prices, nuclear energy's economic advantage is such that in most countries in the Community it could still be cheaper to generate electricity using nuclear power at a load factor below 50%. In France, as I said earlier, nuclear stations can be economic at a load factor of around 25%! This means that nuclear could generate over 60% of the Community's electricity more economically than any fossil fuel. This will not surprise those people living in Ontario. These calculations were done using 25 to 30 year lifetimes for the nuclear plants whereas it appears possible that a technical lifetime of 40 years is realistic.

While the availability of nuclear reactors is usually above 70% – and can be much higher – the load factor for several of the reactors in the Community are often somewhat short of this figure. This is particularly true for the French reactors which presently have a load factor of close to 60%. However, because of the major economic benefits from building these standardised reactors in series they are capable of producing some of the cheapest electricity in the Community. If, through opening up or completion of the internal Market, electricity from nuclear stations could be more widely market throughout the Community and the reactors could increase
their load factors to a figure closer to their expected availability, then it is still possible that the 1995 objective could be achieved.

What about the 50% objective for around the turn of the century? Increased utilization of existing plants and those under construction could help us on our way to this target but will not achieve it, especially if electricity demand continues to climb at its present rate. On the other hand, while we must rule out the possibility of any reactors not yet under construction having any impact on the nuclear capacity by 1995, I am not yet ready to do the same for the end of the century. Decisions can be taken and reactors built in less than 10 years. Though, to be realistic, the target is probably too far away for the time that we have left to reach it.

But whether or not we reach or approach 50% of our electricity by nuclear power by 2000 must not be seen as a major preoccupation for the energy sector. That would be like a runner in a 10,000-metres race — or the marathon — trying to break the record for the 1500 metres without thought to the remaining distance. A good start is important — but we already have that. We are not only up with the leaders but out in front. What is needed now is a calm reappraisal of the situation, planning for the future and finally, and most important, decisions.

For the Community we see decisions concerning the opening up of the Internal Market, the decisions to build new capacity or import from neighbours (these must be taken soon in many of our Member States), what fuel to use and, if nuclear is the choice, the standards to which the reactors should be built and the structure of the nuclear industry itself. There are many paths which our hero might take into the next century. Which one will he follow?

Book Two - The next decade.

Because of the relatively long lead times, especially in some countries, for nuclear power plant approvals, construction and licencing, for many energy planners the next century is just around the corner. While that may be the case, it doesn’t make it any easier to forecast what electricity generating capacity we will need. In the Community, for example, we had a negative electricity demand growth in 1981 (-0.17%) which had changed to a very positive (+4.07%) growth by 1984. Since then it has averaged about 3.2% (against a GDP growth of 2.8%) and we are predicting a growth of around 4% this year.

In our most recent electricity demand forecasts, as part of the Commission’s "Energy 2010" study*, we opted for a range of possible growths. For the period 1990-2000 we used from 1.5% to 3.0% and from 2001-2010 we used 1.0% to 2.5% (these figures are rounded). Such values may seem conservative in the light of present demand growth and the optimistic predictions by many economists for GDP growth through the period. In other words we may be underestimating demand — which could have important repercussions.

* This study is “ongoing” so the numbers presented here from the study are subject to revision. A report on the work will be presented at the next World Energy Conference (Montreal, September 1989) and in a future edition of “Energy in Europe".
At the present time our net electricity production in the Community is approaching 1,700 TWh/year. We would expect this to rise to between 1,900 and 2,400 TWh/year by the year 2000 and to between 2,200 and 3,000 TWh/year by 2010. We could be required to produce at least 50% more electricity in 2010 than we are now – and more if our figures are under-estimates. How will we do this and what rôle will nuclear power play?

For the Energy 2010 study we used two different scenarios: a so-called "nuclear moratorium" and a "nuclear revival". These terms need a little explanation. The moratorium means different things to different countries so that while some Member States would continue not to build nuclear stations others, such as France, would construct fewer stations than in the past. In other words, it is not a shutting down of existing stations or even a freezing of capacity at its present level but a slowing down of nuclear growth. It is, in fact, a continuation of the situation in which we now find ourselves. The "revival" on the other hand does not necessarily mean that all countries would take up or accelerate the move to the nuclear option. The countries without nuclear power now could still be in that situation in 2010.

So what does this mean for nuclear? With a low electricity demand growth and a "moratorium" our forecasters estimated that we could only be producing as much nuclear electricity in 2010 as we are now. This means that the nuclear share would fall below 30% of total electricity production. At the other extreme, with "high" electricity demand growth and a nuclear revival we could be producing at least twice as much as we are now and the nuclear share would be around 45%. In other words, depending on the scenario chosen, nuclear may not grow at all or double in the next twenty years.

The study which produced these forecasts is not yet finalised so I would not like to go into any more detail as to its breakdown of the fuel mix used for generating electricity. However I would like to look briefly at the options that the utilities have.

It is an energy objective of the Community – agreed by the Council of Energy Ministers – that the Member States continue with, and step up, the measures taken to reduce as much as possible the share of hydrocarbons in the production of electricity. The present objective is to reduce the proportion of electricity generated from hydrocarbons to less than 15% by 1995. This has, in fact, already been achieved. While no objectives have been agreed for the longer term there is no obvious reason to increase hydrocarbon's share, especially as our imports of oil and gas are already growing – and will continue to grow – in both absolute and percentage terms.

Given that the possibility of expanding the rôle of large scale hydropower is very limited in Europe and renewables are not likely to make a significant contribution in the time frame, the objective would appear to put the emphasis on solid fuels, in particular coal, and on nuclear power. If we did so – and the "moratorium" scenario was followed – we could be producing around 50% of our electricity from coal in 2010.

This would appear to be well in line with another of the Community's energy objectives – that of increasing the share of solid fuels in energy consumption. However, since the Energy Council agreed these objectives a new player has not only appeared on the sidelines but has been introduced into the game. I refer, of course, to the rapidly growing concern about the environment and, in particular, to the "greenhouse effect".
course, has already taken a lead rôle in this area by hosting the conference on the changing atmosphere and its implications for global security in Toronto in June of last year.

If we had to identify one single issue which we believe could influence our lives most over the next twenty years, many of us would say growing concern about the environment in which we live. It is something which affects every one of us as we cannot isolate ourselves from it. Not only that, the decisions we take now (and those we have already taken) will affect our children, grandchildren and great-grandchildren for many generations. We've got to get it right!

In its conclusion on the Internal Energy Market in November 1988, the Council of Energy Ministers made it clear that "the achievement of a satisfactory balance between energy and the environment - in accordance with the Single European Act - must constitute a major goal of the Community's work". This has become a priority area in the Community and in individual Member States.

As part of our electricity study we estimated emission of SO₂, NOₓ and CO₂. We have already shown that we are bringing SO₂ from power stations under control. We do not expect the same progress with NOₓ though progress is technically possible. The technology is available to reduce these emissions using selective catalytic reactors (SCR) but the cost of doing this is high with a capital cost similar to flue gas desulphurisation (FGD) and high operating costs. CO₂ is another problem altogether. The weight of the gas emitted is more than double the weight of the fuel consumed, so even if it can be taken out of the flue gases how would we dispose of the material?

The "greenhouse effect" has become something about which we are all becoming experts (to one degree or another). Concentrations of the "greenhouse gases" are increasing due to human activities. This is modifying the composition of the atmosphere at an unprecedented rate. The thermal balance of the earth is being modified so global warming and associated climate changes could follow. There is considerable uncertainty as to exactly what changes could take place - but they could well be drastic.

The Community presently produces about 2.8 billion tonnes of CO₂/year. Of this total the power generation sector accounts for about 30% of these emissions. This is a smaller share than it was in the past. For example, the quantities of carbon emitted during electricity production in the twelve Member States in 1987 was the same as it was in 1973 - in spite of a close to 50% increase in electricity production. Assuming that we would have burnt coal in place of nuclear power our emissions now would have been over 70% higher than they now are (360 million tonnes/year in place of 210 million tonnes/year). In the case of using gas in place of nuclear the emissions would have been 25% higher and 50% higher in the case of oil.

In the future, the emissions of CO₂ from power stations could then vary tremendously depending on the scenario followed. For example, we calculated that with a low electricity demand growth and a nuclear revival, the emissions in 2010 could be around the same level as now. On the other hand, with a high demand growth and a nuclear "moratorium" the emissions from our power stations could more than double. This would bring about a close to 20% increase in the Community's total CO₂ emissions.
EMISSION OF CARBON DURING ELECTRICITY PRODUCTION IN THE E.E.C.

(EUROSTAT, OECD & IAEA Data)
This increase would, of course, be less if oil-fired capacity was built instead of coal and even less if gas-fired stations were used. Together with major increases in global reserves of natural gas and the relatively low investment cost and short construction time for a gas-fired station, this explains why, in several of our Member States, there has been growing interest in building new gas-fired capacity in recent months and pressure on the Energy Council to repeal or review a Directive against the use of natural gas for this purpose. Similar events have taken place in the USA and I believe that a significant increase in gas-fired capacity is forecast — and that the utilities are already having some difficulty in finding gas producers who are willing to sign long term supply contracts! There is little doubt in my mind that the more gas-fired stations we build the more the price of natural gas will increase. Also, in the case of the Community, we already import around 30% of our gas and this will increase to around 55% by 2010. In addition, gas still produces CO₂ and, of course, NOₓ in quantities which cannot be ignored. It is not — as many people would like to believe — an environmentally benign energy source.

I said earlier that the period 1990–2000 was one of major decisions. Before 2000 much of the capacity that will come on line in the first 10 years of the 21st century will already have been planned and under construction. I can see some utilities and governments being drawn to gas partly because they are leaving — or have left — too late the decision to build new capacity. They will also be lulled by the soft market for hydrocarbons and the relative environmental benefits of gas. This “benefit” will only be realised if gas-fired capacity is built instead of coal (or oil). It will be a “debit” if it is built instead of nuclear.

I also said earlier that if our forecast was an underestimate it could impact on the choice of fuel mix. It is likely that a low demand forecast would favour gas as being faster and cheaper than nuclear in the construction phase — even though the electricity it generates is more expensive over the lifetime of the plant, rather more flexible in meeting fluctuations in demand and has less impact on the environment than coal or oil. If our forecasts then turned out to be too low, it is possible that gas would again benefit because it could be built in time to meet greater than expected demand.

So, just as nuclear spectacular growth during the 1980s was a result of decisions in the 1970 — and the slower growth in the 1990s follow on from the events of this decade — the major influence on nuclear future in Europe will depend on what is decided in the next few years. Present low fossil fuel prices and lingering doubts in the minds of the public at large following Chernobyl will be balanced in the decision makers mind against the concerns over burning ever increasing amounts of fossil fuel and the continuing safe exploitation of a large nuclear park. I feel that nuclear may well be approaching a cross-roads in Europe. I also feel that while nuclear power may not double its present level of electricity generation by 2010 — though it still could — it will certainly increase substantially.

Book Three — The nuclear future.

There has been much talk, especially since Chernobyl, about the so-called "inherently safe nuclear reactor". Now, following on from the growing concerns about the impact of energy on the environment, even many of the so-called "greens" are advocating increased use of nuclear power — but linking the call to more research to find better and safer reactor systems. This can give the public the impression that the present generation of
reactors are not safe. This is not true. They are safe. Moreover, there is really no such a thing as an "inherently safe reactor". Reactors may possess one or a number of passive safety features, but it would be dishonest to lay too much emphasis on any such features as it is the safety of the system as a whole – including both its design and its operation – that is important. If a reactor is licenced to operate in the Community, then its overall safety is certainly satisfactory.

This is not to say that the present generation of nuclear reactors cannot be improved upon. They can – and a great deal of money is being spent on nuclear research in the Community. Much of this money is being spent on reactor safety research aimed, in particular, at making easier the construction and operation of reactors and narrowing the uncertainty margins. Community funding financially supports a significant portion of this work.

Before I uncover my crystal ball and give you my thoughts on the longer term future, I must tell you something about one particular line of research which we believe will play possibly the major rôle in the long term future of nuclear power. I refer, of course, to the fast reactor.

I know that this reactor with its very economic use of uranium does not endear itself to the hearts of uranium-rich Canadians. However I would ask you to bear with me for a few moments while I extol some of its merits.

The major advantage of the FBR over other types of reactors is that it can extract over 50 times more energy from natural uranium. Thus the energy extractable from known uranium resources would be at least twice that from exploitable coal reserves and more than an order of magnitude greater than for the world's oil resources. However, while the uranium economy of the FBR is not disputed, the reactor type is often perceived as being less safe than water cooled reactors, producing more plutonium and being more expensive. This has led some people to suggest that its introduction, if ever, should at least be delayed until uranium becomes in short supply.

I said above that reactors are only licenced in the Community if their safety is satisfactory. Therefore the FBR and the LWR are equivalent in that they are both safe reactor types. However, there are some differences between the features of the two reactor lines. With the FBR, for instance, of particular importance are the benefits from the point of view of safety of using liquid metal as a coolant. The fact that the metal remains liquid over a wide range of temperatures even at atmospheric pressure, together with its very high thermal conductivity (about 80 times that of water) and its high latent heat of vaporization means that it is a very efficient material for carrying the heat away from the core. Because of these features, together with the large volume of the liquid metal and its low viscosity, the system has considerable thermal inertia which would give the operator several hours in which to take action after any shutdown before the core overheated and was damaged. Another important advantage of the FBR is that it is even easier with this reactor to meet the emission limits for release of radioactive effluents to the environment and it also releases less heat. Furthermore, it actually produces less plutonium than an LWR of the same size and much less than a heavy water reactor.

The present major drawback to the FBR is its initial capital cost. For example, Superphenix cost around 2.5 times more per installed KW than comparable PWRs in France. Figures such as this have caused some people to dismiss the reactor type as uneconomic. However, this does not make allowances for the potential benefits of building reactors in series nor of
possible improvements in design leading to substantial savings in construction. Studies carried out with the support of the Commission have already shown that series construction, including reducing construction times and savings on interest payments, could reduce the cost of building a series of reactors identical to Superphenix by at least 45%. At this price the FBR would still be 50% more expensive than conventional light water reactors so the next step is to look at the actual design of the reactor itself.

Design work on a European Fast Reactor (EFR) started in the Spring of 1988 at the request of the European Fast Reactor Utilities Group (EFRUG) and the companies involved are halfway through a two year conceptual design phase. This will be followed by a three year period to produce a detailed design. You may well have read in recent press reports about the major agreements which were signed recently (February 1989) in Bonn in the field of fast reactor R&D. These agreements provide for the sharing out of design tasks for the coordinated provision of R&D support and for the protection and licencing of know-how developed in the EFR project. Organisations and companies from Belgium, France, FRG, Italy and the UK participated in the signing. There is also a Dutch interest in one of the companies involved.

This is a major advance. We will now have one fast reactor design which will bring together the best features of the different national designs while reducing duplication of effort and saving on the costs. As you would expect, the Commission strongly supports the objective of a single design European Fast Reactor having advocated this for several years – in particular in the previously mentioned PINC. This report, possibly as its most important conclusion, called for an economically competitive design for a FBR to be available (on offer) by 2005 which could be brought into commercial service towards 2015. There must still be a possibility that the EFR could, if required, meet this deadline.

So my vision of the post-2010 period sees the fast reactor ready for commercial introduction if not already under construction. Will it have a rôle to play?

At the time of the first oil crisis in 1973, the World’s annual primary energy consumption totalled well below 6 billion tonnes of oil equivalent (6 Btoe). The OECD region accounted for well over 60% of this. Consumption has now risen to over 8 Btoe with the OECD share falling to around one-half. By 2010 the World will probably be consuming well over 13 Btoe/year with the fastest growth in demand – by far – being in the lesser developed countries (LDCs). So in 20 years time the LDCs will account for about one-quarter of all energy demand. In the longer term this growth will continue as 95% of the World’s increase in population will occur in the developing world over the next 50 years.

These changes are important to the Community, but also to many other countries, for two main reasons. The first is that being a major importer of fossil fuels – by 2010 we will be importing well over 50% of our gas and coal and over 80% of our oil – the Community can be very much affected on the world energy markets. With the rapidly growing demand for energy, we will see increasing competition in world markets for what must be a declining resource base. I am not preaching gloom and doom from the point of view of security of supply. Technically, the resources are such that demand could be covered for many years beyond 2010, but demand will be growing, fossil fuel resources are finite and it is possible that the number of important suppliers will be decreasing. It therefore seems very likely that our energy will – at least – become more expensive.
The second reason is concern about the environment. The large majority of the increasing energy demand will be met by carbon-based fossil fuels. Not only will this mean more $\text{CO}_2$ going into the atmosphere but I wonder how many of the developing countries will be able to afford flue gas desulphurisation or selective catalytic reactors to remove $\text{SO}_2$ and $\text{NO}_x$? As the number of vehicles per capita increases in these countries, how many of the governments will introduce and monitor tight emission control standards? On top of this, there is the impact on global warming of deforestation which so far continues unabated.

We must not try to give the impression that nuclear power can solve the problem. It can’t. We must, first of all, learn to produce and use our energy better — certainly more efficiently — to try to reduce the growth in our demand for fossil fuels and try to persuade the developing world to follow our example. However, we will reach a limit as to how far we can go before we start impacting on our economic growth and the lifestyle to which we have become accustomed and that to which many other nations aspire. Here then is the rôle for nuclear power.

While some of us may feel nostalgic about the steam engine and regret its passing, we must admit that the electrification of the railways has given us faster and more efficient transport and much cleaner stations. A candlelight supper for two (prepared on a wood burning stove?) may be very pleasant — but the electric light is much brighter and more convenient on most other occasions! We now take the tremendous benefits of electricity very much for granted — but there are many others who have not yet had this opportunity. It already does a lot for us — but it can and must do more if we are to reduce our dependence on fossil fuels. My vision is of a world increasingly dependent on electricity for its development. However, while we think of electricity as a clean energy source this is only in its use. In its production it can be one of the most polluting — unless we generate it with nuclear power.

So an increasingly "electric" world needs to be a nuclear one. If we expand our nuclear programmes we will need to be certain we can fuel them. "Ah!" — I hear you thinking — "This is the point when he tries to justify the introduction of the fast reactor because of a lack of uranium resources". You are only partly correct. Uranium resources — like those for the carbon-based fossil fuels — are finite, certainly at the prices we can afford to pay. However, we see no obvious reason for supply failing to meet any likely level of demand for at least forty or more years. Though, if I was to be responsible taking the decision to build a new reactor in 2010 — a time around which many such decisions will have to be taken if only to replace our existing park — and this reactor, once built, would have a life of 40 years, would I be as confident of my fuel supply? Remember, nuclear power will probably be expanding rapidly and those countries without fast reactor technology could be limited to using the conventional — inefficient from the point of view of uranium consumption — reactor lines.

The fast reactor will be needed not to depress uranium demand, but to maintain it at a level which could be met by supply for a much longer period of time. The world will need all the uranium from your remarkable deposits — and more. But without the fast reactor the nuclear star — the hero of our story — could burn very brightly but become extinct far too soon.
About five years ago, a core group of U.S. utility leaders, working with the Electric Power Research Institute (EPRI) in Palo Alto, California, set in motion a far-sighted plan for the revitalization of nuclear power in the United States. Their vision was clear - an improved and simplified light water reactor (LWR) design, building on the proven experience of over 100 LWRs operating in the United States.

These utility leaders gathered support from their U.S. utility colleagues, and they structured a formal industry initiative, called the Advanced Light Water Reactor (ALWR) Program, to develop this concept. The work was funded by U.S. utilities through EPRI, and the Program was staffed by EPRI and utility participants. A Utility Steering Committee of senior U.S. utility executives was established to guide the team's activities.

The ALWR team began work in 1985 to craft, from their sponsors' vision, the specific technical foundation for the next generation LWR. At the time the task must have seemed very ambitious indeed. No U.S. utility had expressed serious interest in building new nuclear plants, and the U.S. media and public at large had already welcomed as inevitable and irreversible the demise of the U.S. nuclear industry.

Since the ALWR utility founders' initial act of faith, the ALWR Program has taken root and grown to the point that it is now a major factor in the emerging new direction of LWR technology. Their early vision of the next LWR has crystallized as a plant design concept which is:

- Substantially simplified compared to current nuclear units.
- Rugged and forgiving, with substantial design margin.
- Based solidly on proven technology. It is "advanced" in the sense of applying the best experience of existing plants.
- Sharply focused on the man-machine interface, and on the needs of the operators who must assure its safe, efficient performance.

To implement these principles, the ALWR Program team took on the task of creating a Utility Requirements Document, a comprehensive statement of performance and design requirements for an "Evolutionary Plant" version of the ALWR. This Evolutionary Plant is envisioned as a large, (approximately 1200MWe) unit, employing substantial simplifications and improvements in plant safety systems and Instrumentation and Control systems. It is a direct descendent of today's ALWRs and its safety systems and regulatory base largely follow conventional approaches. As part of the creation
of this Requirements Document, the ALWR team worked closely with the Nuclear Regulatory Commission (NRC) to identify and resolve significant, outstanding issues of reactor safety and to incorporate these resolutions into the ALWR Requirements Document.

This Requirements Document effort is still working toward its final objective of a completed and comprehensive document reflecting a consensus of prospective utility users and approved by an NRC Safety Evaluation Report. Twelve of its 13 chapters have been prepared, and already it is serving as a reference for U.S. and worldwide users, and it is influencing the design of reactors which could be in place by the end of this century. The structure of the Requirements Document (including its Passive Plant component discussed below) is shown in Figure 1.

Along with the technical progress achieved by the ALWR Program to date, it has emerged as the focal point and catalyst for the cooperation of U.S. and international utilities in directing the course of their future nuclear reactor designs. From its early days as an embryonic EPRI/U.S. utility program, the ALWR work has gained visibility and respect among U.S. participants, and it has attracted the financial and technical support of a number of Asian and European utility participants. The synergistic effect of this growing U.S. and international collaboration has been very significant. As international partners have joined the Program, its technical strength and credibility has increased directly; as that has happened the influence that the Program's technical output has been able to exert on U.S. and international NSSS vendors has increased as well, encouraging in turn even more U.S. and international support.

THE PASSIVE ALWR - A MAJOR STEP FORWARD

As the work proceeded in developing Evolutionary Plant requirements, the ALWR Program team also began a to explore a new LWR concept, which they called the "Passive Plant". This Passive Plant was envisioned as a smaller reactor which would employ primarily passive means - gravity, natural circulation and stored energy - for its essential safety functions.

The Passive ALWR design concept was considered to be potentially attractive to utility investors, for several reasons:

- Due to the fundamental simplicity of the passive safety concept, it could offer an opportunity to effect wholesale simplification (in the form of reduction of many valves, pumps, tanks, instruments, etc.,) with attendant improvement in construction cost and schedule, plant operability, and maintainability.

- By eliminating reliance on active components and human intervention, the Passive Plant could accommodate a wide range of upset conditions and internal and external plant threats, such as loss of all electrical power.
A reference size of 600 MWe was selected by the Passive Plant studies. In theory the passive plant could be of any size, but it is likely that ratings significantly higher than 600 MWe will prove impractical or not cost effective, because of the relatively large component sizes (such as reactor vessel, cooling water tanks) involved. Furthermore, a smaller plant size may prove to be an advantage in its own right in that plants of 500 to 600 MWe may fit more easily into the capacity planning schemes of most U.S. utilities. Also, smaller plant sizes offer a potential for shorter construction time, more extensive modularization of plant equipment, replication learning curve and other factors that can improve overall plant economics.

Phase 1 of ALWR Passive Plant work was a design competition from which two promising conceptual 600 MWe passive designs, a PWR and a BWR, were selected for further development. Both concepts employ passive safety systems and offer fundamental advances compared to existing plants. The most important of these is the design criterion that no operator action shall be required for a period of three days following a design basis event to protect the plant or public.

Phase 2 of the ALWR Passive Plant development conducted in collaboration with the U.S. Department of Energy, involved fleshing out the details of these concepts through extensive technical studies and equipment and system development activities. Figure 2 shows, schematically, the phases of Passive Plant development.

These Passive Plant designs, while still at a preliminary engineering stage, are already bright prospects for successful development. Brief descriptions of the PWR and BWR design concepts follow:

The Passive PWR

The passive PWR concept, referred to as AP-600, is being developed by a design team led by Westinghouse with assistance of Avondale Industries, Burns & Roe, and others. This 600 MWe PWR features an improved reactor coolant system configuration utilizing canned motor reactor coolant pumps directly coupled to the steam generator outlet. This configuration removes the "cross-over leg" from the reactor coolant system, lowering the overall system flow resistance, and improving the small break LOCA performance. This simplified arrangement also allows a single support for the combined steam generator and pump assembly, greatly simplifying the RCS loop support configuration.

The AP-600 also features a natural circulation heat exchanger that removes decay heat from the reactor coolant system at full temperature and pressure, eliminating the need for a pumped emergency feedwater safety system. A gravity-driven ECCS system with full pressure core makeup tanks, in-containment refueling water storage tank, and depressurization capability eliminate the need for a pumped ECCS system.

The containment cooling function is also accomplished passively in the the AP-600 concept. The cylindrical steel containment building is surrounded
by a vented concrete shield building; airflow between the two structures removes heat from the containment shell. Water is allowed to gravity drain to the outside of the steel shell to increase the heat transfer coefficient by evaporation for about the first day after an accident.

Together these features accomplish all the plant's necessary safety functions by passive means, and with substantial reduction in the necessary pumps, valves, and supporting electrical and cooling systems.

The Passive BWR

A design team led by the General Electric Co., and including Bechtel and MIT, is well along in developing a BWR version of the Passive ALWR, called SBWR.

The SBWR is a 600 MWe unit designed to meet similar ambitious targets including no dependence on operator action for three days after a core-damaging event, and a three-year construction duration.

The SBWR reactor is designed for operation at full power without recirculation pumps. Elimination of recirculation pumps and piping permits a simpler reactor vessel design, reduces vulnerability to LOCA events, and it reduces maintenance requirements. The larger reactor vessel needed for natural circulation provides the additional benefit of greater inventory of water over the core at the initiation of any upset conditions.

SBWR's safety features are imaginative and, at the same time, very simple. They include:

- A Gravity Grain Cooling System that will keep the core covered and cooled in the event of a loss of coolant accident.
- A Steam Injector System, which uses residual steam as a driving force for injecting makeup water into the reactor to make up for leakage when no AC power is available.
- An isolation condenser located in an elevated water tank, providing capability for residual heat removal by natural circulation.
- A containment of the pressure suppression type, which is passively cooled under accident conditions.

The capability provided by the passive safety features in the AP600 and SBWR plants can accommodate all design basis events, and there is no need for a safety-grade emergency diesel generator or a class 1E ac distribution system.
GROWING INTEREST IN THE U.S. AND AROUND THE WORLD

Initially this Passive Plant development was an intriguing but low priority part of the ALWR Program. It seemed to be a promising concept, but one which required much development, and more importantly, a fundamental shift in technical philosophy and direction in reactor design. After all, reactors had been growing steadily in size from the small, early prototypes of the 1950's to today's 1300 and 1400 MWe unit.

However, as the work has proceeded, there has been a clear and remarkable shift in the interest of the utilities in the U.S. and around the world. The Phase 1 and Phase 2 work have been technically very successful and have shown promise of better things to come. The designs which have emerged clearly meet the vision and technical principles established by the ALWR Utility Steering Committee at the Program's outset. The Passive Plants can be dramatically simplified compared to today's plants, with rugged design and conservative design margins. They provide outstanding "operator friendliness", particularly in terms of providing long lead times before operator action in required in event of upset or emergency conditions. At the same time they are rooted in fundamentals and proven technology - the passive reactor is a "back to basic design" incorporating the key lessons learned (some perhaps temporarily forgotten) since the beginnings of this excellent LWR technology.

THE WORD AHEAD

At the completion of the current phase of work (Passive Plant Phase 2), scheduled for the first half of 1990, the Passive Plant concept will have been thoroughly investigated. A Passive Plant ALWR Requirements Documents will have been produced and approved by the ALWR Utility Steering Committee, and the passive safety regulatory foundation will have been developed. Conceptual designs for the Passive Plant concepts outlined above will be complete. Together these will constitute an excellent foundation for further Passive Plant development. However, more work will be needed before this Passive Plant can be considered attractive to investors, either from a technical or licensing standpoint.

A follow-on program - Passive Plant Phase 3 - is needed to take the Passive Plant to the point that it truly is a viable ALWR candidate, one demonstrated to meet the needs of utilities, regulators and the public, and a sound basis for utility investment.

EPRI and DOE are working together to create such a follow-on ALWR Passive Plant Program. DOE is proceeding with a major Passive Plant design certification effort, contingent upon industry cost sharing. EPRI and the ALWR Utility Steering Committee are structuring an international partnership of utilities working together to support the design and development of passively stable advanced light water reactors. The objective of this coordinated Program will be to carry existing Passive Plant design concepts to the point that they are considered "investment ready". More specifically, Phase 3 will achieve:

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• A well understood and stable set of regulatory requirements, confirmed via NRC certification of one or more Passive Plant designs.

• A utility-approved and NRC-endorsed Requirements Document setting forth the design requirements of U.S. and international utilities.

• Detailed construction plans supporting the target 36-month Passive Plant construction schedule.

• Probabilistic Risk Assessments (PRAs) and accident analyses demonstrating that accident prevention and mitigation objectives have been met.

• Man-machine interface well defined via functional-based task analyses.

As part of Passive Plant Phase 3, the utilities will make available financial and technical resources for direct application to Passive Plant design work.

EPRI and the current ALWR Program participants are working now to structure this important next phase of the Passive Plant Program. The work is being closely coordinated with a major Department of Energy-sponsored program to support certification of plant designs, and across the board there is enthusiasm, building momentum, and very high expectations for this next phase.

Thomas Edison once said "Nothing is as strong as an idea whose time has come." The time has come for a revival of new nuclear power in the United States and around the world. The motivations are compelling – growing need for electricity, combined with diminishing supplies of fossil fuels and growing concern over the implications of their use. And at the same time the technical concept of simpler and better new reactors is becoming a reality. The Passive Plant is an idea whose time has come.
Figure 1. ALWR Requirements
Document Structure

VOLUME I - ALWR TOP-TIER REQUIREMENTS
- EXECUTIVE SUMMARY
- ALWR POLICIES
- ALWR KEY REQUIREMENTS

VOLUME II - EVOLUTIONARY PLANT ALWR REQUIREMENTS
CHAPTER 1: OVERALL PERFORMANCE AND DESIGN REQUIREMENTS FOR EVOLUTIONARY ALWR PLANTS
CHAPTER 2-13: REQUIREMENTS FOR SYSTEMS AND STRUCTURES

VOLUME III - PASSIVE PLANT ALWR REQUIREMENTS
CHAPTER 1: OVERALL PERFORMANCE AND DESIGN REQUIREMENTS FOR PASSIVE ALWR PLANTS
CHAPTER 2-13: REQUIREMENTS FOR SYSTEMS AND STRUCTURES

Figure 2. Passive Plant Implementation Steps

PRELIMINARY CONCEPT

PHASE I

FEASIBLE CONCEPT

PHASE II

INVESTMENT READY

PHASE III

CONSTRUCTION & SITE SPECIFIC DESIGN DETAIL

FIRST UNIT
Let us start by turning the clock back 15 years, to mid-1974, and imagine ourselves at an energy conference back then, just after the first oil shock. What would we have thought if some speaker had told us that in the next 15 years the USA would hardly increase its total energy use in spite of substantial increases in population and economic activity? Or that, adjusted for inflation, the world oil price would be less in 15 years than at present? That, despite prognosis to the effect that natural gas would run out before oil even as early as the 1980s, globally there would in fact be more gas available than oil, and that gas would be seen as a likely candidate to become the transition fuel into the 21st century? That woodfuel would become the most rapidly increasing energy source in the USA? That wood burned in household heating stoves would be the chief source of air pollution in the USA for some important pollutants? That the USA and other industrial countries would have during much of that period a tremendous oversupply of electrical power plants and refineries?

In regard to nuclear power, we would have been equally sceptical to hear that not one further nuclear power plant would be successfully ordered, built, and put into operation in the USA from this day onward and that the U.S. breeder reactor program would be abandoned? As for the rest of the world, it would have hardly been credible that only three developing countries in the world that did not already have nuclear power plants would build them in the next 15 years, but that two of these would have among the most intensive nuclear programs in the world.

While there were people advocating one or another of these directions in 1974, few would have seriously agreed with them as a set of predictions. Indeed, they would have seemed outrageous to many people.

Why were we wrong? On the supply side, many of us tended to think, for example, that the energy crisis was either a matter of geology or technology. Some of us spoke as if everything depended on whether or not there were sufficient reserves of fossil fuels, or whether some particular energy technology, such as advanced nuclear power, was available. These are indeed important factors, but they are not the only ones. Among others that were recognized at the time is size; e.g., power grids being large enough for an energy industry to take advantage of economies of scale. Another of these limiting nutrients for energy development, now more painfully obvious than then, is the ability to finance new energy systems, especially ones that are highly capital intensive.

The global experience of nuclear power during this period, however, illustrates some important factors that are not confined to geology, finances, or technology in the narrow sense. Outside the already industrialized countries, there are only two countries where nuclear power seems so far to have been a firm success: Taiwan and South Korea.
Each has had rapid development of nuclear power since 1974. Characterizing their economic development in general and nuclear development in particular—but not that of many other countries that have entered the field—are factors such as a dedicated and steadily productive workforce, an efficient level of professional management, national economic planning, and, a means of achieving and maintaining societal consensus. Although now showing signs of eroding, these national organizational resources have been at least as valuable in the short term as more traditional domestic energy resources of oil and coal, and, in the long term, substantially more valuable, as evidenced by Japan, the model of development in Asia.

Another area of the world that has also seen steady growth of nuclear power is Eastern Europe. There is a parallel with Northeast Asia, i.e. Taiwan and South Korea, in that each of these areas had close connections to nations with active nuclear power programs. These special "client state" relationships allowed fairly easy transfer of financing and technology from the USSR in one case and the USA in the other.

This paper deals briefly with three issues: the role of electricity in economic development; the role of nuclear power within the electricity component of development; and the way nuclear power risks are perceived at different levels of development. The definition of development here is a narrow one--economic well-being (per capita income)--recognizing all the problems of its calculation and interpretation.

ELECTRICITY IN DEVELOPMENT

A comparison of world income with population, reveals the phenomenon called by Harrison Brown the "fissioning" of world society. After World War II, by measures of average national income and consumption (energy, electricity, and steel), there came to be two major population groups: the poor with the majority of population but nevertheless distributed from the very poor to only slightly poor, and the rich countries with a similar distribution of their own. Few countries lay in the middle.

In this context, it is worth asking the following question: If all the world's income is divided by the total number of people, what is the average income and what country has such an average income? In 1986, the world's average was about $3100 per capita, about the same as Taiwan with Greece next highest at $3700. According to the World Bank (which does not separately list the eastern European countries), between $3700 - $6200 there was an income gap in which only two countries with a total of less than 3 million people were represented.

The bulk of the poor countries—for example China and India, which represent a large fraction of the world's population—are slowly growing richer in per capita income. In recent years, they have even been growing slightly faster than the rich countries, so in that sense the relative gap has been slightly narrowing. Then there is a group of poor countries, mainly in Africa, which are becoming poorer, where per capita income is dropping. There are also several large and small Latin American economies that have had severe financial problems in recent years. Finally, there
is a group of newly industrializing countries (often called NICs, but more recently renamed as NIEs, newly industrializing economies) that are growing quickly and transiting the gap between the two humps of the curve. These countries are mostly in Asia and include South Korea, Taiwan, Singapore, and Hong Kong.

The trend, then, is toward the appearance of three groups and two gaps. This is due to a second "fissioning, in this case of the developing countries. By stagnation or actual decline, the desperately poor are becoming distinct from the bulk of the developing country population. While the older gap remains below the rich countries, it will become somewhat less distinct as the NIEs continue their transit.

Many developing countries would like to find that magical take-off point and experience the same kind of rapid economic growth that the NIEs have achieved. They would also like to do so by maintaining income equity, as have the Asian NIEs, particularly Taiwan. Indeed, some developing countries today have characteristics that make them look like Taiwan and South Korea in the 1950s when their development was just starting. Today, countries such as Malaysia already seem to be moving along such a growth course with Thailand following behind. Electric power plays an important role in these efforts.

Economic development occurs, by definition, when the economy grows faster than population. In general, when this happens, energy use grows even faster and electricity use grows faster than total energy consumption. Thus, electric power can sometimes be growing at fantastic rates—15-20% for years at a time, as in South Korea and Taiwan. This puts enormous pressure on the management of the power sector, on financing, and also potentially on the environment.

Among the surprises over the last 15 years is the recognition that there is relatively weak link between per capita income and total energy use, but a much stronger one with electricity consumption. There is some reason to think that the two may become less coupled in the future, however, because there seem to be many advances in electric efficiency that have not yet deeply penetrated many markets. There are also important temporal shifts in development patterns that affect fuel demand. Compared to the development period of the presently developed countries, for example, there is an earlier large demand for transport fuel in the presently developing countries. Nevertheless, substantial growth in power demand can be expected in most rapidly growing economies.

There is no question that there is a great potential demand in the developing countries for electric power. At early stages of development, most of the expressed demand is in irrigation and industry. When countries reach a stage achieved not long ago by Taiwan and South Korea, there is a shift in demand from industry to household and commercial and from winter to summer. The peak loads in recent years in Taiwan and South Korea, for example, are determined by air conditioning. Because of geography, much of the Third World will eventually need air conditioning. There is thus a tremendous potential demand for electric power and value to assuring its efficient generation and use.
The importance of the electric power sector for development can also be seen on the opposite side, that is, in the negative effects caused by lack of power availability. It is clear, for example, that the economies of India, China, and Turkey are severely constrained by the lack of sufficient electric power. Some of the major states of India have 50% average power cuts to their industrial sectors. There are investment repercussions including, for example, flights of capital from power-poor areas of a country to relatively power-rich areas. Although determining "unmet demand" is not straightforward, the Chinese government estimates that its hope to increase the power system from about 100 GW in 1987 to 240 GW in 2000 will still leave it almost 20% short.

The high growth rate of the electrical sector eats up tremendous quantities of capital. It is the largest consuming and investment sector in the economy in some nations and is the largest part of the energy sector in most nations. In Taiwan, for example, the power sector has annually soaked up 15-20% of all fixed capital formation since 1974. It represents an even higher fraction of public sector investment, as in most of the five-year plans of India and China.

The importance of electric power can also be seen in the activities of international lending agencies, such as the World Bank. Of the $10 billion lent for energy by the World Bank between 1979-1982, $7.3 billion went for electrical power, even though it makes no nuclear power loans.

In summary, electric power investment in developing countries makes up 5-15% of total capital formation, 15-25% of public sector investment, and 65-90% of development assistance in the energy sector. It is noteworthy, as Jyoti Parikh points out, that about 25% of power equipment in developing countries must be imported. In the early 1980s this amounted to about $10 billion—a significant fraction of the total oil bill for developing countries. Capital equipment is not usually mentioned as part of the energy import problem, but is actually plays a significant role in the balance of trade between many developing countries, even those that import most of their oil.

One implication is that improving the efficiency of the power sector can greatly improve overall economic performance. If the growth of power demand can be lowered or higher generating and transmission efficiencies can be achieved, tremendous amounts of capital can be saved just at a time when such developing countries most need it. Room for improvement can be seen in the low load factors that typify developing countries. In the Asia-Pacific region, for example, half the countries have system load factors of less than 60%.

Developing countries also have bigger problems with power losses, that is, with electricity generated but never billed to a consumer. In the USA, Japan or Taiwan, such power losses run about 5-6%. Loss rates in developing countries run much higher. Many countries could save a whole year's investment in capital equipment and new supply if they could cut these power losses by 10% or more.

One implication of these conditions is that the highest marginal return on investment in most developing countries' power systems is actually
increased efficiency of the system; better management rather than new capacity. It is not clear, however, how much can be done with the power system in isolation from changes throughout the economy. Indeed, it is almost a tautology to say that developing countries generate and use electricity inefficiently since they use all resources inefficiently, e.g. labor, capital, energy, land, etc.

The electric power sector has been the most important sector in dealing with the principal need generated by the oil shocks in most developing countries—moving away from oil, in particular imported oil. It is the power sector that has been the most successful in this task. This is partly because it is the most homogeneous and regulated sector, and therefore the easiest for governments to get a handle on, to control. Also, most of the major substitutes for oil are in power generation while few substitutes exist in transportation and other oil-using sectors. These substitutes include coal and nuclear, but also the renewables such as hydro, geothermal and wind.

This oil substitution in the power sector, however, has some interesting effects on the international oil market. The big changes in the world oil picture that occurred in the 1970s were in the so-called "upstream" market, that is, in the production of oil, the ownership of oil fields, and the marketing of crude oil. The changes in the 1980s have been in the "downstream" market; in refining, in petrochemicals, and in product transport. OPEC countries, for example, have been building refineries because they want to be more dominant in the market for products as well as that for crude. As a result, there is relatively more product and less crude being traded internationally.

Stated simply, running a particular crude oil in a particular refinery will result, within fairly narrow limits, in a certain fraction of light products such as gasoline, another fraction of middle distillates such as kerosene and diesel, and another fraction of heavy products such as the oil used in large power plants. If most of the substitutes for oil are in the power sector, however, then only the demand for heavy oil is actually being changed. Thus, there is not necessarily a decrease in overall demand for crude oil depending on the demand for other products.

It is possible to reduce the relative fraction of heavy oil produced by buying a different type of crude oil or by upgrading or building new refineries. This is expensive, however, and requires an understanding of what may happen in the world product market before making decisions with long-term consequences. This prediction is difficult, for example, because the demand for the middle distillates, kerosene and diesel, is growing rapidly but erratically in developing countries. In summary, ownership and control of product transport and marketing are changing, complicating energy planning in developing countries. The largest pressures behind these changes come from the power sector.

Finally, electricity plays a critical role in what might be called the "dialectic of development" an important part of which is the tension between "top-down" and "bottom-up" development approaches. Should development proceed by having experts make analyses and then governments institute programs that implement their plans downwards; or does
development proceed from grassroots efforts to communicate needs and desires upwards? Ideally, of course, both processes operate. Electrical power sits in the center of the dialectic because it is generated centrally but distributed to every household. Electric light in the village is a symbol of development, regardless of any economic benefit that may accrue from it. Consequently, there is great political pressure in many countries for rural electrification, regardless of cost. It is expensive to electrify locations with low loads; lights and a few small motors. But it is considered important in order to provide for basic human wants. This often places extreme financial and operational demands on power systems.

An important part of this dialectic relates to economics. Electrification schemes often do not recover their costs and thus utilities in developing countries are continually going after new capital. The World Bank believes they should charge the real costs of production to all their consumers, even their rural consumers, so that they can repay loans and improve economic efficiency. The politics are such, however, that it is impossible to set such rates in rural areas, for the poorest parts of the population. It seems there has to be a willingness to subsidize rural electrification; it will not pay for itself, at least not at this stage of development. Obviously, however, there are limits and important decisions to be made, for example how much subsidy is justified.

NUCLEAR POWER IN DEVELOPMENT

Having briefly addressed the overall role of electricity in development, let us consider the role of nuclear power. Among its many disadvantages, nuclear power has two potential advantages: cost and energy security. Unfortunately, it is extremely difficult to reliably determine and compare costs of power projects, especially since these are spread over 20 years or more. Even though there are financial techniques for considering costs over time, they cannot make an evaluation that is truly independent of accounting practices, value judgments about future costs, and general economic conditions. Experience has shown how large the range within which the economic climate can change over such a period.

Using these techniques, however, it is possible to generalize that, over a 30-year lifetime, a nuclear plant will be a bit cheaper than coal in many parts of the world. There are exceptions and qualifications, to be sure, particularly depending on location with respect to coal mines and the amount of non-plant infrastructure that is often required for nuclear plants in developing countries. In addition, this advantage has been steadily eroding in many parts of the world.

An important financial disadvantage to nuclear, however, is that the government or utility and their banks involved in the financing must initially spend much more money per year for the nuclear plant than for the coal plant. Construction times for nuclear power plant are estimated now at 10 years. And it takes another 12 years or so until the annual expenses of the nuclear plant are actually less than the costs of a comparable coal plant. Of course, if all works as planned, after that 22 years there are savings from the nuclear system. But given economic
uncertainties and the relative instability of many governments, this extra requirement for capital in the short term may be perceived as too daunting inspite of the results of standard economic calculations that may show long-term economic benefits for nuclear power.

Another factor that must be considered before purchasing a nuclear but not a coal plant is that, as pointed out by Alvin Weinberg, the nuclear industry is a global one. If an accident happens in a nuclear plant in India, news of it will be in the papers in Beijing, Tokyo, and New York. If an accident happens in a coal plant in India, however, few hear about it. And when a nuclear accident occurs, similar and sometimes even quite different plants will be closed down around the world, at least while immediate investigations are undertaken. A nuclear utility is vulnerable to the safety record of utilities everywhere else in the world and thus to factors outside its control. This is like the aircraft industry: if a DC-10 crashes in Chicago, all DC-10s around the world are grounded.

The accidents at Three Mile Island and Chernobyl, for example, have had, and will continue to have, negative impacts on nuclear industry worldwide, although apparently the latter, paradoxically, in a way that is almost inverse to the radiation dose received, i.e. more impact in Western than in Eastern Europe and more in the USA than in the USSR. It affects LDC perceptions as well, of course, as shown by the Philippines government's citation of Chernobyl as one of the factors in its decision not to begin operation of its essentially completed nuclear power plant. This global vulnerability translates into financial penalty for nuclear power.

The global nature of nuclear power has expression also in regulation. When the Philippines, for example, were intending to embark on nuclear power, they were not in a position to establish a complete regulatory apparatus to cover all aspects of construction and operation. They intended to use the USNRC's regulations and updates, especially those covering the same type of reactor in the USA. In other words, the U.S. government (or any other nuclear exporter) essentially exports ongoing regulation as a part of nuclear technology export, as it does with aircraft. Not that it physically inspects the foreign systems, but the needed information is transferred as a result of the regulation of the same type of system in the USA. This became somewhat of a problem for the Philippines, however, because the USA cancelled the only reactor of that exact type being built, in Puerto Rico. While this was probably only a minor factor in the government's 1986 decision to abandon the plant, it illustrates the complexity of this global industry. It also illustrates one of the factors that the industry will need to address if nuclear power is to find an expanded role in many other developing countries.

How do things stand for nuclear power? There are several ways of looking at the question. One is by generating capacity. As shown in Table 1, early this year there were about 310 GW in operation worldwide of which LDCs account for less than 14 GW or about 4%. Of that, 80% is in two countries: South Korea and Taiwan. The total capacity in operation and in construction is about 400 GW globally, with about 23 GW in the LDCs, or 6%, more than half of which is in South Korea and Taiwan.
Table 1
Global Nuclear Power as of Early 1989

<table>
<thead>
<tr>
<th>Status</th>
<th>World</th>
<th>LDCs</th>
<th>LDC%</th>
<th>Asian NIEs</th>
<th>% of LDCs</th>
</tr>
</thead>
<tbody>
<tr>
<td>In operation</td>
<td>310 GW</td>
<td>13.9 GW</td>
<td>4</td>
<td>80</td>
<td></td>
</tr>
<tr>
<td>Under construction</td>
<td>88 GW</td>
<td>8.9 GW</td>
<td>10</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>398 GW</td>
<td>22.8 GW</td>
<td>6</td>
<td>53</td>
<td></td>
</tr>
</tbody>
</table>

Less-developed countries (LDCs) are defined here to be non-European nations with less than the world mean per capita income ($2900 in 1984). The nuclear Newly Industrializing Economies (NIEs) of Asia (Taiwan and Korea) account for a large fraction of total LDC nuclear power. GW = gigawatt = 1000 megawatt. Compiled from the world surveys in Nuclear Engineering International (8/88), Nuclear News (2/89), and International Atomic Energy Agency Bulletin Vol. 30, No. 3, 1988.

Table 2 shows the relative amounts in individual LDCs. Although six nations have operating plants and three (China, Mexico, and Cuba) are currently building them, only one new developing country, Brazil, has put nuclear power into service since 1978. Of course, with any large project and nuclear plants in particular, one should not count the chickens until they are hatched, hatched and running. For example, the Philippines' nuclear program was initiated after the first oil shock and completed its first unit in 1986. Now, however, with completely different economic, political, and energy situations than the 1970s, the government made a decision to abandon it, not to put it into operation. Iran was building four reactors at the time of the revolution that toppled the Shah in 1979, but since then their future has been uncertain. It has been reselling the uranium originally bought to fuel these reactors. Indeed, mainly as the result of their severe financial difficulties, neither Brazil nor Mexico is likely to pursue new nuclear projects in the near future and may even abandon the existing programs.

Table 3 shows three other measures of nuclear power. By number of units, the USA dominates with 108, about twice as many as its closest competitors, the USSR (57) and France (55). Next come other western industrial countries including Japan (36), but there are no developing countries until South Korea with 8 followed by India and Taiwan with 6 reactors apiece.

By percentage of all electricity generated by nuclear plants, France and Belgium dominate, with close to 70%. Next come Taiwan and South Korea, however, which are both close to 50%. No other developing country has as much as 15%.

There is also a fourth way of looking at nuclear power: in relation to the size of the economy as also shown in Table 3, i.e. how many nuclear
Table 2
Nuclear Power in LDCs as of Early 1989

<table>
<thead>
<tr>
<th>Country</th>
<th>Year of First Operation</th>
<th>GW</th>
<th>% of Total LDC</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>In Operation and Under Construction</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>India</td>
<td>1969</td>
<td>2.9</td>
<td>13</td>
</tr>
<tr>
<td>Pakistan</td>
<td>1972</td>
<td>0.1</td>
<td>1</td>
</tr>
<tr>
<td>Argentina</td>
<td>1974</td>
<td>1.6</td>
<td>7</td>
</tr>
<tr>
<td>Korea</td>
<td>1978</td>
<td>7.2</td>
<td>31</td>
</tr>
<tr>
<td>Taiwan</td>
<td>1978</td>
<td>4.9</td>
<td>22</td>
</tr>
<tr>
<td>Brazil</td>
<td>1984</td>
<td>1.9</td>
<td>8</td>
</tr>
<tr>
<td>Mexico</td>
<td>(1989)</td>
<td>1.3</td>
<td>6</td>
</tr>
<tr>
<td>Cuba</td>
<td>(1990)</td>
<td>0.8</td>
<td>4</td>
</tr>
<tr>
<td>China</td>
<td>(1990)</td>
<td>2.1</td>
<td>9</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td>22.8</td>
<td>100</td>
</tr>
</tbody>
</table>

**Most Likely by 2000**
Turkey
Egypt

**Possible but unlikely by 2000**
Philippines and Iran, which have partially completed units in place
Bangladesh, Indonesia, Libya, and Thailand

This list does not include any Eastern European countries, most of which have nuclear power programs and several of which might be considered LDCs by some definitions. Nor does it include South Africa, which started the first of its two 920 MW units in 1984. See exclusions and data sources at bottom of Table 1.

megawatts are there per billion dollars of the economy? This indicates the relative financial commitment of the nation. France is high on this list as well, but not the highest. The highest is Taiwan, when both operating and in-construction plants are included. Sweden is also high, as is South Korea.

It is instructive in this regard to compare three countries that are often seen to be similar in economic development patterns (although at different stages)—Japan, South Korea, and Taiwan. Japan has one of the oldest and largest nuclear power program in a non-nuclear-weapons nation and did so while achieving high economic growth. How do the programs in Taiwan and South Korea, which started much later, compare with Japan's at a similar stage? It turns out that in comparable financial terms, South Korea and Taiwan have built nuclear power plants 10-15 times faster.
<table>
<thead>
<tr>
<th>Country</th>
<th>Number of Operating Units&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Percent of Electricity Generated&lt;sup&gt;b&lt;/sup&gt;</th>
<th>Economic Intensity&lt;sup&gt;c&lt;/sup&gt; Under Construction</th>
<th>Operating</th>
</tr>
</thead>
<tbody>
<tr>
<td>Korea</td>
<td>8</td>
<td>53</td>
<td>26</td>
<td>76</td>
</tr>
<tr>
<td>Spain</td>
<td>10</td>
<td>31</td>
<td>15</td>
<td>63</td>
</tr>
<tr>
<td>France</td>
<td>55</td>
<td>70</td>
<td>50</td>
<td>57</td>
</tr>
<tr>
<td>Hungary</td>
<td>4</td>
<td>39</td>
<td>16</td>
<td>53</td>
</tr>
<tr>
<td>Taiwan</td>
<td>6</td>
<td>49</td>
<td>80</td>
<td>47</td>
</tr>
<tr>
<td>Czechoslovakia</td>
<td>8</td>
<td>26</td>
<td>13</td>
<td>32</td>
</tr>
<tr>
<td>Germany, Federal</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Republic</td>
<td>23</td>
<td>31</td>
<td>16</td>
<td>31</td>
</tr>
<tr>
<td>Switzerland</td>
<td>5</td>
<td>38</td>
<td>20</td>
<td>31</td>
</tr>
<tr>
<td>Canada</td>
<td>18</td>
<td>15</td>
<td>24</td>
<td>27</td>
</tr>
<tr>
<td>Belgium</td>
<td>7</td>
<td>66</td>
<td>44</td>
<td>26</td>
</tr>
<tr>
<td>Sweden</td>
<td>12</td>
<td>45</td>
<td>86</td>
<td>25</td>
</tr>
<tr>
<td>South Africa</td>
<td>2</td>
<td>5</td>
<td>0</td>
<td>25</td>
</tr>
<tr>
<td>United States</td>
<td>108</td>
<td>18</td>
<td>20</td>
<td>20</td>
</tr>
<tr>
<td>U.S.S.R.</td>
<td>57</td>
<td>11</td>
<td>17</td>
<td>18</td>
</tr>
<tr>
<td>Japan</td>
<td>36</td>
<td>31</td>
<td>15</td>
<td>14</td>
</tr>
<tr>
<td>Germany, Democratic</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Republic</td>
<td>5</td>
<td>10</td>
<td>9</td>
<td>13</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>40</td>
<td>18</td>
<td>20</td>
<td>12</td>
</tr>
<tr>
<td>Argentina</td>
<td>2</td>
<td>13</td>
<td>14</td>
<td>10</td>
</tr>
<tr>
<td>Brazil</td>
<td>1</td>
<td>1</td>
<td>2</td>
<td>10</td>
</tr>
<tr>
<td>India</td>
<td>6</td>
<td>3</td>
<td>5</td>
<td>8</td>
</tr>
<tr>
<td>Mexico</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>7</td>
</tr>
<tr>
<td>China</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>1</td>
</tr>
<tr>
<td>Italy</td>
<td>2</td>
<td>0.1</td>
<td>4</td>
<td>6</td>
</tr>
<tr>
<td>Finland</td>
<td>4</td>
<td>37</td>
<td>49</td>
<td>0</td>
</tr>
<tr>
<td>Bulgaria</td>
<td>5</td>
<td>29</td>
<td>43</td>
<td>0</td>
</tr>
<tr>
<td>Yugoslavia</td>
<td>1</td>
<td>6</td>
<td>8</td>
<td>0</td>
</tr>
<tr>
<td>Netherlands</td>
<td>2</td>
<td>5</td>
<td>4</td>
<td>0</td>
</tr>
</tbody>
</table>

The countries are ranked according to the economic intensity of nuclear plants under construction.

(inflation-adjusted dollars added to the national income per megawatt built). This means that the financial and related vulnerabilities are also substantially higher.

As is well documented, past projections have nearly always anticipated a much larger swing to nuclear power than has thus far been the case worldwide. This is true both for the developed countries and the less developed countries. It can be argued, in fact, that only two countries that have recently entered the nuclear field have made a real success of nuclear power--South Korea and Taiwan--and that, in some respects, they have done better than the USA. As in Japan, their nuclear plants are built on time, at substantially lower unit cost, and operate at higher capacity; evidence of their shared "energy" resources of good management.

Historically, there have been two approaches to nuclear development: exemplified by India and Taiwan. India's initial approach was to create its own industry, to import as little as possible. This decision, coupled with its refusal to sign the Non-Proliferation Treaty, effectively put it outside most of the world market in nuclear technology. Nevertheless, India was the first developing country to build a nuclear power plant (1969), although it has been proceeding slower than it hoped since. Because of the great need for new electric capacity, however, India has just concluded an agreement by which two 1.1 GW Soviet nuclear plants will be built. This marks a sharp change in policy and an even stronger commitment to nuclear power.

In Taiwan, on the other hand, a decision was made not to develop any of its own nuclear technology, but to import the most advanced and largest reactors possible in order to support its general industrial development. Because of its political situation, it has insisted on safeguards against weapons proliferation being written into bids for construction contracts and kept to U.S. units. It apparently does not want even the remotest suspicion to interfere with development or energy security. As part of this effort, a great deal of technology transfer occurred from U.S. companies. No units have been ordered since 1974, however, and Taipower is now faced with substantial public opposition in spite of following Japan's example of compensating local communities near nuclear power sites.

South Korea followed a similar although not so extreme path by which it has been importing reactor technology in concert with technology transfer, but did so from three countries: Canada, U.S., and France.

China has been pursuing a mixed path. It has an indigenous industry building a 300 MW power reactor that is an upgrade of a submarine reactors and developing prototypes for 400 MWth district heating reactors. In addition, it has ordered two French reactors to be built near Hong Kong, and which will be basically "turn-key" projects. Like its Northeast Asian neighbors, however, China expects a high degree of technology transfer so that it can become capable of providing a larger fraction of future plants on its own.
This generic illustration of the Risk Transition shows the decline of traditional and the rise of modern risks. The patterned area represents the period of risk overlap when there may be important interactions between traditional and modern risks. Point A marks the position of a typical developing country contemplating a nuclear power program, while Point B marks the risk pattern in a developed country.

RISK TRANSITION

Economic development implies transitions of several kinds. Well documented, for example, are the demographic transition (changes in birth and death rates) and the energy transition (changes from non-commercial to commercial fuels), both of which signal profound societal transformations. Less-well understood, but also of considerable importance, is the risk transition, which marks a shift in the relative significance of different hazards for human well being. As shown in Figure 1, at low levels of economic development human health risk is dominated by what might be called "traditional" risks, in which poor rural environmental conditions play an important role. The health outcomes from these risks prominently involve epidemic, infectious, and intestinal diseases. Although this set of risks tends to decrease during economic development, it still dominates overall risk patterns in the developing world.
During development, however, these traditional risks are gradually replaced by "modern" risks, which appear to rise with development. These are characterized by chronic diseases and cancers and are associated with environmental hazards related to agricultural modernization and industrialization.

This framework assists in understanding a number of aspects of societal behavior related to risk, including providing plausible explanations for some of the "paradoxes of risk" that have perplexed many observers. For instance, there is a paradox seen in societies at point B in the figure that is often stated as follows: "Why have the richest and healthiest societies in human history become so worried about what, in global and historical terms, are extremely small individual and societal risks from modern technologies?" With reference to the figure, however, this can be explained by reasoning that people intuitively consider the risk-lowering as well as the risk-raising aspects of new technologies when making judgments. Even though formal risk assessment procedures are only well developed for determining gross risk raising, it may be the perception of net risk (the difference between risk raising and lowering) that actually drives human judgment.

Thus, before reaching the level of economic development represented by point B, societies may have been willing to tolerate incremental increases in modern risk because they perceived that major reductions in traditional risks were still occurring. At point B, however, even relatively small increases in modern risks have become unacceptable because little additional traditional risk lowering seems possible. In contrast, at point A it may be quite rational to undertake what, to some observers, may seem to be large modern risks in order to accelerate economic development with its attendant great risk lowering.

Consider now the development history of the high-technology high-visability system, nuclear power. Being high technology, at levels of development too far to the left in Figure 1, the well-known limitations of grid size, personnel, financing, and regulation constrain nuclear development. At some point in development, however, these barriers can be overcome and nuclear power may appear quite attractive. This can be so even though nuclear power may be perceived to be a significant modern risk. The explanation, as introduced above, is that there is so much obvious risk-lowering occurring with new development and, being high-visability, nuclear power is closely associated with development such that the perceived net risk is negative, i.e. risk lowering.

As development proceeds, however, the slope of reductions in traditional risk tapers off, thus shifting the perceived net risk against nuclear.

The way the risk transition is perceived by nations also depends on culture, history, international circumstances, and politics (e.g. freedom of the press). Nevertheless, the recent public outcries in Taiwan that have combined with other factors to lead to a what is essentially a moratorium on nuclear expansion would seem to fit well with this framework. Appropriately following behind, there are similar signs in South Korea, including demonstrations at low-level waste dumps. (Even Japan is experiencing the growth of serious anti-nuclear-power movements.) Indeed, such perceptions can run so high as to lead nations to undergo fairly large sacrifices rather
than accept nuclear power. This is illustrated by the the island of Luzon in the Philippines, which in early 1989 was undergoing severe power shortages with even worse expected by summer. The discomfort, inconvenience, economic loss, and cost of emergency gas turbines were apparently more acceptable than questioning the decision not to start their completed nuclear power plant.

Nuclear power's high visibility sometimes results in yet another type of risk, political. The Philippines reactor seems to be permanently tainted by its association with the Marcos government, for example. Perhaps an even better example is found in the Pacific Islands, which, at first look, might seem far removed from nuclear power debates. Nuclear questions, however, actually dominate the environmental concerns of people throughout the Pacific. In Fiji and Tonga, for example, these issues have been at the top of the environmental agenda for both government and non-government groups. The reason, of course, is the history of nuclear testing in the Pacific, initially by the USA and UK and presently by France. Outside of Hiroshima and Nagasaki (and now perhaps the Ukraine), the Pacific Islands are the part of the world that has sustained the greatest damage from the nuclear age. In addition, the anti-nuclear and the anti-colonial (independence) movements coalesce in the Pacific. Even civilian nuclear power has created concerns in the Pacific Islands, because of past U.S. proposals to use Palmyra Island as a storage site for spent-fuel and Japanese proposals to dump low-level radioactive waste at sea. Combined with the history of testing, failures at the resulting cleanup operations, and perceived coverups by the French and U.S. governments, any such proposals run into strong political and emotional hostility in the Pacific. Risk analysis is almost meaningless in this context as are attempts to distinguish civilian from military uses.

What about the most currently popular of the many risk issues that bear on nuclear power: global warming from fossil fuel combustion? To a first approximation, the impacts of climate change are similar to an increase in the frequency and severity of climate-related natural disasters, i.e. typhoons, droughts, floods, and storm surges. Such events are traditional risks in the framework described here, i.e. they tend to decrease with development. The hurricane of a size that kills ten people in Florida will kill 10,000 in Bangladesh, for example. In the framework of Figure 1, therefore, climate change amounts to an addition to the traditional risk curve all along the development spectrum. The absolute impact will be greater in developing countries, but the relative impact may well be greater at the developed end of the spectrum. This could well act to shift to the right the point at which nuclear risk raising is no longer balanced by its risk lowering. This might increase the size of nuclear power’s niche along the development spectrum.

CONCLUSION

In summary, except in the special conditions of Taiwan and South Korea, nuclear power has not found a major niche in the economic development of many poor nations. It is not dead, but it does not flourish. The places to watch during the next 15 years are China and India, but it seems safe to say that growth will not be rapid. The extent to which concerns about climate change will affect nuclear power's growth rate in developing countries will largely depend on how much the developed countries are willing to pay to do something about it.
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Canada's installed electrical generating capacity has grown by more than ten-fold since 1950 when 91% of all sources were hydroelectric. Since then, conventional thermal and nuclear stations have been built in growing numbers so that by 1988 hydro plants made up only 57% of the 101 GWe total. The balance was distributed between coal, oil, and gas-fired facilities, 18, 8, and 5% respectively, and nuclear generating stations, 12%.

The growth in Canadian electricity consumption during the past six years averaged 5.0% per year, more than double that predicted by utilities and governments alike. Forecasters continue to be cautious in predicting future growth. Forecasts of increases in installed capacity during the period 1987 through 2005 by Canadian utilities, the National Energy Board, and Energy, Mines, and Resources range from 1.4 to 2.3% per year. Thus, electrical generating capacity is expected to reach 129-151 GWe in 2005. If growth near the high end of the range occurs, three new nuclear stations beyond Darlington, totalling about 10 GWe, would be required by 2005.

Over the longer term, if one were to assume an annual growth rate of 1.8% from 2005-2050, installed electrical capacity in Canada could increase to some 300 GWe. In provinces such as British Columbia, Manitoba, and Newfoundland, undeveloped hydroelectric resources are large enough to accommodate most of the anticipated growth. However, the seven remaining provinces must develop other sources, principally conventional thermal and nuclear generating plants.

Societal pressures in opposition to the burning of fossil fuels can be expected to increase because of escalating concerns over the evolution of acid gases, particulate contamination of the atmosphere, and carbon dioxide emissions which exacerbate the "greenhouse" effect. Nuclear power plants, on the other hand, will become much more acceptable because of their relatively small impact on the environment, and their economic and technical advantages.

A conservative estimate indicates that at least 70 GWe of additional nuclear generating capacity would be required in Canada by 2050. This capacity in the form of uranium-fuelled CANDU reactors would be installed in many of those provinces where hydroelectric resources already are or soon will be fully exploited. The future picture seems clear - a nuclear renaissance is about to begin in Canada.
INTRODUCTION

If I could begin on a personal note, problems of electricity supply first became apparent to me four decades ago when the Hydroelectric Power Commission of Ontario, HEPCO, launched a program of frequency conversion from 25 cycles to 60 across the province. The Hydro plan proceeded smoothly and customers like myself were provided with new motors for washing machines, refrigerators, electric razors, etc. at no charge. Even in those days, electrically-powered appliances were an essential part of life. However, when looking at the situation in retrospect, the electrical generating capacity then was surprisingly small.

For example, the year I graduated from university, 1948, Ontario's December peak primary electricity demand was 2245.2 MW(1). HEPCO's dependable peak resources then included 1319 MW of hydraulic capacity and 718 MW of purchases for a total of 2037 MW. Thus, the demand exceeded dependable supply by about 10% and radio appeals for conservation from HEPCO's Chairman at the time, Mr. Robert H. Saunders, were frequent and compelling. Nonetheless, programmed cuts in electricity supply were scheduled, usually around the dinner hour, and candle-lit suppers became commonplace.

Forty years later, in December 1988, a new record peak demand more than ten times larger, 23000 MW, was reached in Ontario. The growth in Canadian capacity was equally impressive as shown in Figure 1(2)(3). Additions proposed for commissioning by 2013 are included as dotted lines in Figure 1(2).

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Figure 1
Installed Electrical Generating Capacity in Canada(2)(3)
It is to be noted that currently no additions to nuclear capability are planned beyond the Darlington Nuclear Generating Station which comes into service over the period 1989-1992. Decisions on new nuclear plants could be taken within the next one to two years in Ontario, New Brunswick, and Saskatchewan in the face of increased load growth but are very dependent on political processes now underway.

Energy Demand and Supply in Canada

According to a recent study by the National Energy Board, NEB(4), end use energy demand in Canada is projected to grow between 1.4 and 1.8% per year to 2005 to levels of 9.0 to 9.9 PJ as shown in Figure 2.

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**Figure 2**

Canadian End Use Energy Demand by Fuel(4)

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Petajoules

12000

10000

8000

6000

4000

2000

0


Electricity

Natural Gas

Oil

3% NGL
7% Renewables
6% Coal & Coke

3% 7%
3% 8%
3% 8%
3% 8%
3% 7%
3% 4%
3% 3%
3% 5%
3% 4%
3% 4%
3% 4%
3% 4%
3% 4%
Fuel shares are expected to remain relatively stable except for oil which should decrease as consumers continue to switch to other energy sources. Coal's share may double because of an anticipated increase in the use of coal for bitumen recovery. Electricity's share should increase to about 21% of the total, representing a growth from the 1986 level of consumption of from 37 to 55%.^{(4)}

The projections in Figure 1 indicate that about one-third of the expected growth to 2005 could be accommodated by already-proposed additions to hydro and conventional thermal plants. The balance will be derived from additional hydro and thermal sources (see Fig. 3). For example, in the high case illustrated in Figure 3, three new nuclear plants beyond Darlington would be required by 2005 whereas in the low growth case no additional stations would be needed then^{(4)}.
Forecasters in the provincial and federal governments are predicting more modest growth in the future than has been actually experienced in the past. The recorded growth in electricity consumption in Canada during the period 1983 through 1988, for instance, averaged 5.0% per year\(^2\). Moreover, the trend line of per capita electricity consumption versus gross domestic product for the period 1960-1987 has been on the increase\(^2\) (see Figure 4).

The NEB's projections for electricity growth rates between 1990 and 2005 range from 1.7% /year for the low case to 2.3% for the high\(^4\), less than half that experienced in recent years. Utility forecasters are being equally cautious as illustrated by the situation in Ontario.

**Figure 4**

Historical Relationship Between Electricity Demand and Gross Domestic Product, 1960-1987\(^2\)
The average annual increase in Ontario's electricity consumption since 1983 has been 4.8% (see Figure 5), about twice the growth that had been forecast by Ontario Hydro.

Figure 5
Ontario's Year-over-Year Growth in Electricity Demand

Should the growth trend of the past six years persist for the next decade, Ontario's electricity demand could increase from 134.4 TWh to over 200 TWh by 1998. This level is considerably higher than that projected by Ontario Hydro in its recently-issued business plan for 1989-1998 as illustrated in Figure 6.
Ontario Hydro's most likely basic load forecast indicates that demand could reach 174 TWh by 1998. It is hoped to reduce that level to what is defined as primary load by higher efficiency programs, load shifting, and non-utility generation. Even the basic forecast seems conservative, however, if healthy economic conditions continue to prevail.

Future Supply Alternatives

Several of Canada's ten provinces are well-endowed with hydroelectric generating capacity and have large potential for further development. These include British Columbia, Manitoba and Newfoundland. According to a recent report, only eleven years of load growth could be met in Quebec with the remaining economic hydraulic resources assuming electricity growth rates equal to the actual increases averaged over the last five years (see Figure 7).

Alberta and Saskatchewan have some hydraulic potential and substantial resources of fossil fuels. Ontario and the Atlantic Provinces are not so fortunate and will have to rely on supply alternatives other than hydroelectric for meeting the bulk of new load growth.
There are several such alternatives available. In Ontario, they include:

1) purchase of power from Manitoba or Quebec
2) installation of new coal-fired plants
3) fuelling with natural gas in either large central plants or in small combustion turbine units owned by the major utility, Ontario Hydro, or by private corporations
4) construction of new nuclear generating stations

Ontario Hydro officials have been exploring with their counterparts in Manitoba Hydro and in Hydro Quebec the possibility of long-term commitments to purchase large blocks of power from either or both utilities. There are several difficult hurdles to overcome in these negotiations including:

- projected costs up to twice those which would result if indigenous facilities were to be built in Ontario
the construction of long, high voltage transmission lines with a large fraction of them outside the province
- the inability of Ontario Hydro to wind up as owner of the generating facilities even though the capital to build them had been supplied initially.

The installation of more coal-fired capacity, particularly for base load, is not highly favoured because of the additional environmental problems imposed on a system that is already facing severe restraints related to acid gas and particulate emissions, and ash disposal. Furthermore, according to a recent public opinion survey conducted by Goldfarb Associates for Ontario Hydro, there is very little support for the construction of new coal-fired generating facilities.

There is considerably more support among the public for the utilization of natural gas which is perceived as a “clean” fuel. Those charged with the provision of secure, economic electricity, however, are concerned that long-term availability of natural gas is less than certain. The National Energy Board concluded recently that wellhead natural gas prices could increase from two- to threefold by 2005 (see Figure 8), and wholesale gas prices in Ontario could more than double within fifteen years for the high case considered (see Figure 9).

Figure 8
Fieldgate Natural Gas Prices, Low and High Cases

($C 1987/GJ)
Currently, Canada exports about one-third of its total production of natural gas to the United States, but this only amounts to about 6% of total US consumption. Many expect that American demand for Canadian supplies could grow markedly in future since proved United States reserves can sustain the current US consumption rate for little more than a decade\(^{10}\). Given the recent deregulation of Canada's natural gas industry and the passage of the Canada/US Free Trade Agreement, upward pressure on natural gas prices for Canadian consumers in the near-term would seem highly probable.

Another concern associated with the use of natural gas and all other fossil fuels is the inevitable evolution of carbon dioxide and its contribution to the "greenhouse" effect. Already, some politicians around the world are beginning to think seriously about imposing limitations on the discharge of CO\(_2\) into the atmosphere. In late February, 1989, for example, at an Ottawa conference on atmospheric pollution and climatic change, the assembled group urged that international controls should be placed on carbon dioxide emissions and that such limits should be incorporated in a new "law of the atmosphere" modelled on the Law of the Sea. It was suggested that negotiations to achieve these objectives could begin at the World Climate Conference in Geneva in 1990\(^{11}\). Such suggestions give electricity supply planners cause for concern in committing new facilities which need to be operated for at least 30-40 years if the original capital investment is to be recovered.
The one remaining alternative which offers reasonable near-term answers to the problems of electricity supply for provinces such as Ontario, New Brunswick and Saskatchewan, is the nuclear option. Within a decade, Quebec may be looking towards new nuclear commitments as well. The long-standing record of nuclear achievement in Canada provides a sound basis for future decision-making. CANDU reactors have been found to be more economic than other supply option alternatives when indigenous fossil fuel is not available(12)(13). They are being operated safely and at high standards of technical performance according to a recent review by a Royal Commission(14). In addition, because nuclear power stations impose little burden on the environment, public acceptance of the nuclear option is growing(15). All told, the time is ripe in Canada for positive decisions at the political level both federally and provincially which could lead to a renaissance of Canada's nuclear industry.

**Future Prospects**

Current forecasts for electricity demand by both Energy, Mines and Resources and the National Energy Board extend only to 2005(2)(4). This relatively short interval provides little more time than that required to obtain all the necessary approvals and to allow for the construction period necessary to build the next round of nuclear plants. Once this first step has been taken the domestic nuclear industry should flourish.

The prospective growth after 2005 could be illustrated by the following example. Electrical generating capacity in 2005 is expected to reach 129-151 GWe(2). If the average annual growth rate of 1.8% predicted for the next two decades is maintained until 2050, installed electrical capacity in Canada then could reach 300 GWe. Some 60% of the 150 GWe increase is likely to be installed in Ontario, Quebec, and New Brunswick. At least two-thirds should be nuclear plants. Thus, Canada's nuclear capacity could increase by 60 GWe between 2005 and 2050.

Some might say that it is premature to be projecting requirements so far ahead. The young people who are already with us and who will still be consumers of electricity after the middle of the next century would not agree.

The projected large increase in the growth of the domestic nuclear power industry raises the question of uranium supply. Canada's present nuclear capacity of some 12 GWe absorbs only about 15% of the current domestic uranium output(16). Given the country's large resource base and its promising geological potential for new discoveries, self-sufficiency in uranium supply seems assured for many decades into the future.
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LUNCHEON AND BANQUET ADDRESSES

LUNCHEON ADDRESSES

MONDAY

The Honourable Robert C. Wong
Minister of Energy, Province of Ontario

TUESDAY

Her Honour Sylvia Fedoruk, O.C., S.O.M.
Lieutenant Governor of Saskatchewan

BANQUET ADDRESS

TUESDAY

The Honourable Jake Epp
Minister of Energy, Mines and Resources Canada
NOTES FOR REMARKS

BY

The Honourable Robert C. Wong
Minister of Energy
Province of Ontario

Luncheon Address – Monday, June 5, 1989

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Good afternoon, ladies and gentlemen. It’s a pleasure to address you at a time when there are significant developments in energy -- particularly electricity -- that affect all of us.

Today, I’d like to bring you up to date on electricity planning in Ontario. It’s a process that involves many groups, including the public, the government and your industry. Recognizing the audience today, I will focus on the nuclear option and give you my views of your role in the planning process.

The atmosphere in which planning is taking place is important. In today’s milieu, there are fundamental issues that we must deal with in electricity planning. And, the public is both interested and concerned.

The recent publicity about fusion, what some people are calling the "fusion confusion," is a good example of what can happen when something really catches the public imagination.
Perhaps only a precious few people had anything but the vaguest notions about the science of fusion. But for a few weeks, many of us were dreaming about the possibilities that would open up if fusion gave us unlimited supplies of inexpensive electricity.

But the real bonus that attracted us to fusion wasn’t the amount of electricity or the cheapness of it. It was the fact that fusion promised us energy without environmental damage or risk to people. Let’s face it -- no one would have been very excited if we came up with a new fuel that was cheap, but dirty or dangerous.

Thinking about fusion may well have helped all of us get a better grip on the energy problems we face. For those of us who make decisions in energy fields, it reinforces the importance of environmental factors in energy decisions.

And fusion may have helped the public get a better grip on a couple of facts. First, that we don’t have endless electricity resources; and, second, that we have to pay a price -- both environmentally and financially -- for the electricity we do have.
We need to have a better public understanding of these issues, because we do have some challenges in electricity planning. Obviously, one of those challenges or goals is to keep the "prices" that we pay for our energy supply as low as possible. In any event, we must work to achieve these goals with today's technology and resources -- not with ideal solutions.

My own province, Ontario, is at an important point in its electricity planning. You, of course, have a keen interest in nuclear-generated electricity.

So do we in Ontario. We have a $30-billion investment in nuclear-generated electricity. Today, more than half our electricity is generated in nuclear-powered stations. In the mid-1990s, when Darlington is fully on line, more than 60 per cent of our electricity will be nuclear-generated.

That's one of the reasons why Ontario supports continuing research on nuclear-generated electricity.

We have joined with the federal Government, Atomic Energy of Canada and Ontario Hydro in a study of the future of the nuclear industry in this country. To ensure continued safe, reliable and economic operation of Ontario's nuclear plants, the industry requires ongoing technical support and continuing research and development.
We also support research that looks to the future. I am, of course, once again referring to fusion. Canada is playing a significant role in the international effort to harness this form of energy. Ontario participates through the Canadian Fusion Fuels Technology Project.

Now, to come back home again, let's take a closer look at the Ontario situation. Right now, Ontario is at a crossroads. We must make decisions on energy because demand for energy, electricity in particular, is rising.

By the year 2000, Ontario's residents are likely to be consuming 13 per cent more oil, 25 per cent more natural gas, and, in the absence of demand management measures, 39 per cent more electricity.

My mandate is to ensure that Ontario has adequate and secure supplies of energy -- supplies that are at reasonable prices and that avoid unnecessary damage to the environment.

In mid-April, when I spoke to the seminar sponsored by the Joint Industry Task Force on Electricity Supply, I reminded them that Ontario is running on a strong and efficient electricity system, one that has met the needs of our society for many years.

At the same time, I made a commitment to make certain that we continue to meet our needs in the future.
I also told the industry task force that energy choices can no longer be resolved in a back room. This government has made a commitment to involve the public in future energy decisions -- because of the important impact such decisions will have on everyone in the province.

We have already begun a public education and consultation process that will lead to timely decisions on electricity. Our first Energy Choices conference was held in Toronto in April. Later this month, we’ll be holding a second conference in Sudbury and there’s a third in Ottawa in October.

We have laid the basis for decision-making on electricity in several ways:

- we have introduced legislation that redefines the role the government plays in the electricity system.

- we have encouraged our society to be more energy efficient, and

- we have taken a methodical approach to gathering and providing the government and the public with information needed to make informed decisions on electricity.
I'm going to begin with the amendments to the Power Corporation Act, the legislation that defines the relationship between the government and Ontario Hydro.

This government believed that the act needed to be updated to ensure that Ontario Hydro is responsive to government policies and public priorities.

Our amendments will give government the authority to issue policy statements on Hydro's activities, and require Hydro to submit its plans and reports for review. The government will also have early access to Hydro information for review and comment.

Hydro will provide the government with all necessary information on its plans, including plans for meeting environmental goals.

A second area in which the government has been active is in promoting energy efficiency. In fact, I have made energy efficiency my first priority. Improving our efficiency record can contribute to our international economic competitiveness and to environmental improvement and protection.
Our new Energy Efficiency Act requires that a number of major appliances and energy-consuming products sold in Ontario adhere to energy-efficient standards. When the regulations begin taking effect this fall, consumers will begin to benefit from the energy cost savings.

We have also encouraged Ontario Hydro to develop programs to help their customers use electricity more efficiently. Hydro now has time-of-use rates that encourage efficiency by giving a price break to industrial customers when they use electricity during off-peak hours.

And we at the ministry are helping Ontario municipalities, large building owners, householders, consumers, industry and agriculture to improve their energy efficiency.

Demand management is going to be a permanent part of our planning on a long-term basis. In economic terms alone, we must be more energy efficient just to compete in the new, global economy.

The third thing the government has been doing is gathering and providing information to the public that will make it possible for everyone to make informed choices about energy.
We have had thorough reviews of Ontario Hydro's Demand/Supply Planning Strategy. The final version of this strategy has taken account of reviews by an independent panel, a dozen provincial ministries and a Select Committee of the Legislature.

Over the years, there has been widespread public concern over nuclear safety, over the costs of nuclear-generated electricity, the relative cost of generating electricity using other fuels, and over the disposal of radioactive waste.

In response to public concern about nuclear safety, the government commissioned the Hare Report, which concluded, and I quote, "No significant adverse impact has been detected in either the work-force or the public. The risk of accidents serious enough to affect the public adversely can never be zero, but it is very remote."

Public concern over costs has been an issue. The panel that examined the cost of nuclear-generated electricity concluded that Hydro's cost estimates are sound ones.
And we will have more information about the costs of generating electricity from coal and other fuels when the independent reviews commissioned by Ontario Hydro are complete.

Each of these reports is a step in fulfilling the Ontario Government's commitment to gather all the information necessary for the best possible decision about the future of our electricity system.

Other levels of government are providing information as well. On waste disposal, the federal government has referred AECL's work of the past 10 years for public review. The Ontario Government fully supports this review, which will take two years. Earlier this year, I had the opportunity to visit the research facility at Whiteshell in Manitoba. I left there impressed with the work being done and the dedication of everyone involved in this pioneering program on high-level waste disposal.

Most of the facts are now on the table or soon will be. And that's what I have been doing -- getting the facts on the table and not prejudging conclusions.
In April I told the Energy Choices conference that a major new generating station is one of Ontario's energy options. I also specified that a new nuclear generating station is clearly a possibility, reminding people that our existing nuclear stations produce about half our electricity, efficiently and safely.

Ontario is fortunate in having a range of supply options to choose from. Keep in mind that oil-fired, gas-fired and coal-fired generating stations are also among our supply options -- along with non-utility generation, enhanced supply from existing generating stations and purchase of electricity.

Later this year, Hydro will be presenting the government with its preferred plan for meeting electricity needs in the next two decades. The plan will address the timing of need for supplies, what electricity we should plan to purchase, what generating stations we should plan to build, and what demand management measures we should plan to implement.

At that point, public interest in electricity planning will grow rapidly. Forecasts indicate that additional supply will be needed in the 1990s. And all of us have to recognize that the people of Ontario will accept the need for new, major sources of electricity supply only if they are convinced that all other cost-effective and environmentally acceptable options have been explored.
I believe, as does the Select Committee on Energy, that no supply option should be ruled out until the relative benefits and costs of each have been determined. That is precisely what I mean when I say I am keeping an open mind on our options.

We also have a wide variety of demand-side options to consider. When I attend meetings of the Premier’s Round Table on the Environment and the Economy, I promote demand management for the environmental and economic contributions it can make.

Our goal must be sustainable development -- development that seeks to meet the needs and aspirations of the present without compromising the ability to meet the needs of the future.

The questions I will ask about any proposed mix of energy options are:

- Do these choices enhance our industrial competitiveness and economic well-being?

- Are they sensitive to the needs of the environment?

- Do they provide Ontario with adequate and secure supplies of energy at reasonable prices?

- Do they address the merits of demand management?
It is clear that nuclear electricity from Candu systems already plays a major role in our energy mix and will continue to do so in the future. In the next century, our energy mix will include such things as private generation -- in the form of small hydro and cogeneration -- wind energy, solar energy, geothermal energy and other non-nuclear sources.

All of us involved in energy have a challenge facing us. We must see that the public has the facts about our needs and options. We must inform the public that there are costs, there are benefits -- and there are trade-offs to be made when making any energy decision.

That's why I have encouraged other energy industry associates, and why I'm asking the CNA to help the public become better informed about the options and choices facing them. Only when we have complete and accurate information can we make wise energy choices.

By working together, we can create an energy future that will best serve the needs of this generation and generations to come.
NOTES FOR A SPEECH
BY
Her Honour Sylvia Fedoruk, O.C., S.O.M.
Lieutenant Governor of Saskatchewan

LUNCHEON ADDRESS
Tuesday, June 6, 1989

SEE VOLUME I
NOTES FOR A SPEECH

BY

The Honourable Jake Epp

Member of Parliament for Provencher and
Minister of Energy, Mines and Resources Canada
Government of Canada

Banquet Address - Tuesday, June 6, 1989
INTRODUCTION

I want to thank the CNA for inviting me to speak this evening. This is an accomplished and auspicious group, responsible for the achievements that have made Canada a world leader in peaceful nuclear technology.

A mere 50 years after the discovery of nuclear fission, nuclear energy is making a great contribution to the world's energy supply and has brought immense benefits to medicine, science and industry.

All along, Canada has been a front-runner in nuclear research and in putting the fruits of that research to peaceful use across the entire nuclear fuel cycle. I understand there are people in the audience tonight who are pioneers, going back to the days when AECL was first formed in 1952. I wonder who would have believed back then that in three decades or so nuclear energy would produce as much electricity as Canada was then producing from all sources?

When I look ahead, I see more successes for Canada's nuclear industry. I know we face difficulties. Many people have deep concerns about nuclear energy. And in recent years, no one has been selling reactors. But for a variety of reasons, times are changing. I think all of us, including the public and your customers, the power utilities, must take a fresh look at the nuclear option.

TIME FOR A NEW ASSESSMENT

Since becoming energy minister, I have begun to make a new assessment of nuclear energy. I am compelled to do so because we in Canada have some tough decisions to make. But let me repeat to you what I told the energy ministers of the western nations at a meeting of the International Energy Agency in Paris last week: "[Nuclear energy] is an option which I strongly support, which I believe can be safe and environmentally sound and for which there is significant potential for further development in Canada as in many countries of the industrialized and non-industrialized world. We must take whatever steps are necessary to ensure confidence in nuclear power." We must re-examine Canada's nuclear industry precisely because we want to retain the nuclear energy option. When I look at the forces shaping our future it seems very clear that we are going to need nuclear energy.
In the future, demand for energy -- and particularly electricity -- is going to increase. If we assume modest growth in the economy here in Canada and an accompanying increase in electrical demand of two or three per cent a year, we will need to double generating capacity by the year 2020.

Certainly, conservation and increased energy efficiency will offset some of the need for new supply, and that is good. It always makes economic and environmental sense to eliminate wasteful use of energy. But in a cold country with many energy intensive resource industries and vast distances to travel, there are limits to how much we can reduce our use of energy. Renewable energy sources will meet some of our needs. But they are also constrained, for reasons that have to do with limited supply, economics or the environment. Most forecasts call for increases in our energy supplies.

In the newly industrializing countries, electricity use tends to grow even faster than the GNP. For example, Korea's electricity consumption has, on average, grown at about 15 per cent a year for three decades, while its GNP has grown at about nine per cent.

Throughout the world, the mix of types of energy being used is changing. Electricity is expanding its share of the total energy market. Indeed, greater energy efficiency often means more use of electricity, not less.

The question is: how are we to generate the electricity to meet this growing demand?

We in Canada are blessed with a variety of energy sources. We have ample uranium, coal, gas, heavy oil and tar sands, and untapped hydro-electric potential. Today, two-thirds of Canada's electricity comes from hydro-electric generating stations. And some regions, particularly Alberta, enjoy an abundance of low-sulphur coal which can be used to generate electricity economically.

However, some areas of the country have readily turned to nuclear-generated power. Ontario now derives half its electricity from Candu reactors; New Brunswick over a third.

In some regions large-scale hydro-electric projects or easily accessible low-sulphur coal can be the most economic choices for electricity supply. But in many regions these options are not available and nuclear power can then be the preferred option. That is the situation in Canada and in much of the industrialized and developing world.
When we consider such a crucial commodity as energy, I think it is obvious we wish to maintain as many supply options as possible. We want both security of supply and diversity of sources. With our efficient Candu reactor technology and our abundant uranium resources, Canada can count on nuclear power as one of our options.

Of course, the benefits Canadians derive from the nuclear industry go beyond a supply of electricity. The skills and jobs of a very dynamic, high technology industry with export potential have contributed immensely to the Canadian economy. This is the sort of industrial development we want in Canada.

Environmental issues have come to preoccupy us and that has to have a dramatic impact upon the choices we make about energy. Urban smog, acid rain and global warming have become close to household words. In Toronto last year, the Conference on the Changing Environment recommended that, by 2005, we reduce carbon dioxide emissions by 20 per cent from 1988 levels. That is a very tall order — when we consider expected growth it is equivalent to a halving of our expected carbon dioxide emissions in 2005. The issue is being given the most careful study here and abroad. I discussed it with other Ministers at the International Energy Agency meeting in Paris last week. While nuclear energy poses environmental issues of its own in many people's minds, it does not contribute to global warming.

For this and the other reasons I have outlined, Canada needs a viable nuclear industry.

PUBLIC CONCERNS

We do have to take a look at the difficulties confronting the industry, and get at the task of dealing with them.

Many members of the public still have deep concerns about nuclear power. Your industry has taken on the task of informing the public. You may get assistance in that by the unfolding of events. The greenhouse effect and other concerns are going to increase public interest in energy issues, and focus attention on the trade-offs we face regardless of what energy options we choose. Clearly, nuclear power, which can contribute to meeting new demand for electricity without producing acid rain or greenhouse gases, is going to get closer attention.

Opinion polls tell us public concern centres upon the safety of nuclear reactors and the disposal of radioactive wastes.
Canada's track record on reactor safety is very good. The Candu reactor has been subjected to several safety reviews recently and has passed with high marks. At the request of the Ontario government, Professor Kenneth Hare completed a study last year of the safety of Ontario Hydro's Candus. He did suggest some minor improvements but his major findings were that the Candu reactors are operated safely and with high standards of technical competence. An Operational Safety and Review Team from the International Atomic Energy Agency conducted a safety review of the Pickering Station and reached similar, favorable conclusions.

Canada has long had an excellent research and development program directed to reactor safety. Not only do we do our own research in this area, but through routine and ongoing exchanges, we enjoy access to research and information from other countries.

Within Canada, we have developed a very good system for the regulation of nuclear facilities. It places the onus upon plant operators to maintain high standards of safety and they have done well. And we have developed an effective regulatory agency which is devoted to ensuring that workers and the public are protected. The President of the Atomic Energy Control Board was quoted last week as saying he needs more resources to maintain an adequate standard. The Government knows that public safety and public confidence are essential ingredients in a sound nuclear industry. Because of that, we attach as much importance to the question of regulatory resources and capability as we do to any other aspect of current review of Canada's nuclear industry.

When it comes to nuclear waste, the industry knows the importance of good science and responsible management. The approach being taken now is of safe interim storage of spent reactor fuel in pools at reactor sites, a system that provides indefinite storage.

In fact, this kind of storage has been the preferred method of management for many years. It allows the heat and radioactivity produced by the fuel to decay so that ultimate handling and disposal will be much easier. Because of uranium's high energy content, the volumes of spent fuel to be stored and disposed of are small.

For the longer term, plans are being developed for disposing of nuclear fuel wastes by burrying them deep in the stable rock formations of the Canadian Shield. At Whiteshell, AECL is carrying out a very thorough research program on this disposal concept. The nuclear industry in Canada is preparing to meet very high standards of safety and environmental
protection for waste disposal over a very long period, and to ensure that no harm is done to human beings or the environment over that period.

This disposal concept will be subjected to an exhaustive Environmental Assessment Review, with public hearings and extensive critiques by scientific authorities, before any final decisions are taken. This review will begin soon and will be one of the most important environmental assessments ever carried out in Canada, a proof of the government's desire to address fully the public's concerns about this issue.

I have been a frequent visitor to Whiteshell; it is in my riding. And since taking on the duties of the Minister of Energy, Mines and Resources, I have visited most of the other AECL sites. I am very impressed with the skill and dedication of the people at all the sites and the work they are doing. Thanks to these efforts, and those of others in the uranium industry, the provincial utilities, and the private sector nuclear firms, Canada has become recognized as a world leader in so many areas of scientific and technological endeavour and across the nuclear fuel cycle. We have made exciting discoveries in medicine. We have developed new techniques and devices for a wide range of industries. And, of course, we have continued our ongoing support for the Candu reactors, which includes coming up with new and improved designs.

We keep telling everyone that when you look at reliability, safety and economy, the Candu reactor is the most successful in the world. AECL and the entire nuclear industry is a valuable asset to the Canadian economy. This government is not about to allow such a successful venture to languish.

THE NEED FOR RESTRUCTURING

We are taking steps to keep Canada's nuclear program moving. But before I discuss what we are doing, I'd like to review a bit of the background.

Initially, the Candu system was devised by a team drawn from AECL, with its scientific skills; Ontario Hydro, with its operating experience; and Canadian industry, with its manufacturing capabilities. In the early 1970s, there were a lot of orders for new reactors both in Canada and abroad. There was plenty of work for everyone. AECL and Ontario Hydro began to go their separate ways. Ontario Hydro began to do more and more of its own engineering, while AECL specialized in engineering for reactors in other parts of Canada and abroad.
But the work dried up. The nuclear industry has endured a period of drought. AECL's last sale of a Candu reactor was made to Romania in 1981 and Ontario Hydro made its last commitment to a new station, at Darlington, even earlier than that. The drought has hit our international competitors too, though that brings small comfort to our industry.

A year or two ago, AECL was unable to fully fund its engineering work out of operating revenues. It asked the government to help finance the design of a new reactor, the Candu 3. That request prompted us to take a hard look at the larger issues surrounding Canada's nuclear program. Was there a solid future for the Candu business? Could Canada reasonably afford to pay the cost of staying in the business until markets improved?

These questions are being reviewed by the government. We can see that the nuclear program is important to Canada from several perspectives: energy supply, environmental considerations, industrial development and scientific research. We also believe, for the reasons I have already discussed, that Canada and other countries will need new generating capacity and will want to maintain a competitive nuclear option. There will be a market which can sustain an ongoing nuclear program.

However, we do have some concerns. Most fundamental is the need to restructure Canada's nuclear industry to effectively harness its immense capabilities. AECL has become largely marginal to the nuclear program in Ontario, where the bulk of the world's Candu capacity resides. At the same time, Ontario is a significant indirect beneficiary of AECL's major research and development and design efforts -- a fact recognized by Ontario Hydro's contributions to AECL's research effort. So we need to review how these two main actors in our Candu program should relate to one another. We also must consider how the private sector can best participate in a restructured nuclear "team Canada".

**ACTIONS AND DECISIONS**

My colleague in the Ontario government, Robert Wong, spoke to you yesterday. We met earlier this year and I found him most supportive. As you know, Ontario is now going through a process of evaluating its energy supply options for the next decade or so. I understand it may be a year or more before any decisions will be made, but Ontario wants a competitive Candu system to be an available option.
Mr. Wong and I, as well as AECL and Ontario Hydro agree there is a need to rebuild a single team. Thus, the key players will now evaluate options, in an atmosphere of cooperation, for restructuring some parts of the Canadian nuclear industry.

We have sought and received opinions from the private sector on this need for restructuring, which might actively involve some key private players. For this reason and others, we will ensure that close consultation continues.

Currently, we have a consultant working with all the players to sort out the available options for restructuring. He will report back within a month or so. Then we will make some decisions. I expect that our plans will take shape later this summer.

I don't want to speculate further on the details of the restructuring at this time. However, let me be clear about our objectives. We want to retain a nuclear energy option and keep for Canada the economic, scientific and industrial benefits that flow from a nuclear program. The Candu reactor is a proud Canadian accomplishment with a demonstrated record of excellence. We believe markets will open up. And when they do, we want Canada to be prepared to take advantage of them.

AECL has retained its international competitiveness. We are the leaders in surmounting an obstacle in markets at present: the high up-front capital costs of building a reactor. AECL has taken the lead in developing a smaller, less expensive reactor, the Candu 3. Its size is well matched to the needs of many utilities around the world. This new reactor is also easier to construct. It can be built in pieces in factories and then the modules can be shipped to the site and assembled. This permits much better control over the timing, cost and quality of construction. We are now actively pursuing various discussions which we hope will lead to the building of the first Candu 3 within Canada, so that it can serve as a demonstration to attract potential customers.

CONCLUSIONS

In conclusion, let me emphasize that I am a strong supporter of retaining a healthy nuclear industry in Canada. There are very compelling reasons to have faith in the future of your endeavors. The demand for electricity is expanding, both in Canada and worldwide. Environmental considerations will compel a fresh look at meeting that demand through nuclear energy. We in Canada have demonstrated that nuclear power can be safe, clean and economical. We have been a front-runner in the industry for much of the fifty years since the discovery of
fission. We still are front-runners today. Our agenda is intended to keep us a front-runner for the foreseeable future.

Thank you.
PROCEEDINGS

10th Annual Conference

OTTAWA, ONTARIO
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Foreword

As in previous years, the Annual Canadian Nuclear Society Conference was held in conjunction with the CNA Conference. This year, to recognize the Fiftieth Anniversary of the Discovery of Nuclear Fission, a special plenary Symposium, co-sponsored by the Canadian Nuclear Society, Canadian Nuclear Association, and other Canadian Learned Societies, was held as part of the CNS Annual Conference. The proceedings of the Special Symposium are published as Volume 1 of the Proceedings of the 10th Annual Conference of the Canadian Nuclear Society.

Volumes 2 and 3 contain the proceedings of the 17 technical sessions from the 10th Annual Conference of the Canadian Nuclear Society. We are pleased to include several papers which had been accepted but which, because of travel difficulties, were not available for presentation at the Conference.

The papers for these proceedings were prepared on standard forms supplied by the Canadian Nuclear Society and are generally published as submitted by the authors. Responsibility for the content of each paper rests solely with the author.

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The success of the 10th Annual Conference was due not only to the efforts of authors and other participants at the Conference, but also to the assistance of the numerous CNS members and others who gave of their time to participate in reviewing the large number of papers received. The support of the CNA organizing committee under the chairmanship of R. Veilleux also contributed significantly to the success of this conference. The secretarial assistance of Mrs. V. Mussell is also gratefully acknowledged.

P.J. Fehrenbach, T.J. Jamieson
Conference Co-Chairmen
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Session 1:

Advanced Fuel Cycles

Chairman:

R.W. Morrison, Energy, Mines & Resources
ABSTRACT

The pace and focus of the CANDU advanced fuel cycle program will be influenced by a number of world-wide trends and opportunities. These include the status of uranium production and resources, the slower pace of nuclear expansion, the delay in deployment of fast breeder reactors by several decades, and the growing stockpile of light water reactor (LWR) spent fuel. Development of SEU cycles and the CANFLEX fuel bundle is the next logical step for CANDU. The demonstration of high burnup cycles is an essential step for all potential advanced cycles in CANDU and must be executed as soon as possible. Tandem cycles offer the possibility of LWR-CANDU symbiosis using a proven reactor technology and a range of fuel recycle options which, in some cases, utilize modifications of existing or near-developed technology.

INTRODUCTION

The recent history of the CANDU advanced fuel cycle research and development program, conducted by Atomic Energy of Canada Limited (AECL), has more or less paralleled the experience of the Canadian nuclear industry. From proposals and plans for broad-scale, large-scale expansion in the middle 1970's, the perspective changed in the early 1980's to one of deferred expectations, and rapidly contracting markets and prospects. A major internal review of the program in 1985 resulted in significant reductions both in scope and funding, and a refocussing on the near-term prospects for exploiting the development of slightly-enriched uranium (SEU) fuelling.

The future pace and focus of the CANDU advanced fuel cycle program will be influenced by a number of world-wide trends and opportunities, and these are discussed later. However, the underlying future potential is still similar to that described twenty-five years ago by W.B. Lewis in his paper "How much of the rocks and oceans for power? Exploiting the uranium-thorium fission cycle" (Reference 1). Fission energy using advanced fuel cycles can be an economic power source for many centuries, even if the natural uranium price is many times higher than current levels. The current major problem is the maintenance of an adequate pace of development during a period of low demand, to ensure that the technology will be available in a timely fashion to meet the challenges and exploit the opportunities of the future.

TRENDS AND OPPORTUNITIES

The pace and focus of the CANDU advanced fuel cycle program will be influenced by a number of world-wide trends and opportunities. These include the status of uranium production and resources, the slower pace of nuclear expansion, the delay in deployment of fast breeder reactors by several decades, and the growing stockpile of light water reactor (LWR) spent fuel. These are explored in more detail below.

Uranium Resources and Supply

A perspective on uranium resources and needs is given on Table 1 (taken from Reference 2) for the world excluding the centrally-planned economies (CPE). The annual natural uranium requirements for the years 1985 and 2000 are compared with two levels of resources taken from the 1986 "Redbook" (Reference 3). Only uranium resources recoverable at a cost up to $130 US/kgU have been considered. The ratio of the resources to the annual requirement gives the resource lifetime i.e. the length of time the resources would last at that annual consumption rate.

<table>
<thead>
<tr>
<th>YEAR</th>
<th>1985</th>
<th>2000</th>
</tr>
</thead>
<tbody>
<tr>
<td>ANNUAL NEEDS (10^3 Mg/a)</td>
<td>36</td>
<td>50 - 60</td>
</tr>
<tr>
<td>RESOURCES (10^3 Mg)</td>
<td>~ 3600 (1)</td>
<td>~ 2900 (2)</td>
</tr>
<tr>
<td>RAR + EAR-1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RESOURCE LIFETIME (YEARS)</td>
<td>100</td>
<td>50 - 60</td>
</tr>
<tr>
<td>RESOURCES (10^3 Mg)</td>
<td>~ 5300 (1)</td>
<td>~ 4600 (2)</td>
</tr>
<tr>
<td>RAR + EAR-1 + EAR-2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RESOURCE LIFETIME (YEARS)</td>
<td>150</td>
<td>75 - 90</td>
</tr>
</tbody>
</table>

NOTES

1. 1986 REDBOOK, UP TO $US 130/kgU.
2. ASSUMING NO NEW DISCOVERIES.
In 1985, the resource lifetime for "cheap" uranium is estimated to be in the range 100 - 150 years. This is calculated to fall to 50-90 years by the year 2000, assuming no new mineable uranium discoveries are made. Consequently, even under these pessimistic assumptions, there are adequate resources to meet the needs of a power program based on once-through fueling until well into the next century.

It is believed, however, that uranium exploration programs will find additional uranium resources before the year 2000. Currently, there is a relatively low level of exploration as measured by the expenditures given in Table 2. The peak year for uranium exploration was 1979 and expenditures have declined rapidly since that time. The level relative to oil drilling exploration is also significant and indicates that the large uranium resources already defined have been found relatively easily and at a small fraction of the investment needed for other energy sources. Consequently, it seems likely that significant new discoveries will be made when earlier levels of exploration activity are renewed.

<table>
<thead>
<tr>
<th>YEAR</th>
<th>URANIUM (EXCL CPE) 10^6 US$</th>
<th>OIL DRILLING USA (1)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1979</td>
<td>756</td>
<td>16 079</td>
</tr>
<tr>
<td>1982</td>
<td>344</td>
<td>39 428</td>
</tr>
<tr>
<td>1986</td>
<td>144</td>
<td>9 800</td>
</tr>
</tbody>
</table>

**NOTES**

1. **REPRESENTS ~30% OF WORLD ACTIVITY (EXCL. CPE).**

Although the resource picture appears healthy for several decades into the next century, the current supply situation appears more uncertain. The annual non-communist world uranium production (Table 3) indicates current levels at or below annual consumption (Table 1). The spot price for uranium, however, does not reflect this situation, and recently has fallen to a new low of below $10 US/lb of yellowcake. Although there have been recent suggestions of price manipulation which are currently under investigation in the USA (Reference 4), the usually accepted explanation of the current situation is that any supply shortage has been more than counter-balanced by a world-wide reduction in the natural uranium inventories held by the utilities.

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>TOTAL</td>
<td>37.8</td>
<td>37.7</td>
<td>34.7</td>
<td>44.0</td>
</tr>
<tr>
<td>CANADA</td>
<td>12.6</td>
<td>11.7</td>
<td>10.9</td>
<td>7.2</td>
</tr>
</tbody>
</table>

* - ESTIMATED

**SOURCE: NUKEN MARKET REPORT**

How long this current situation can last is uncertain. The steady increase in uranium consumption and some inventory rebuilding could result in some demand/supply imbalance. The demand-stimulated increase in uranium exploration and new mine development will take time before there is an increase in supply. In the interim there may be a consumer panic with sharp increases in price as utilities compete for more secure supplies to cover current spot purchases. Canadian utilities and off-shore CANDU customers should be in a more secure position, however, based on expanding Canadian production (Table 3).

### Face of Nuclear Expansion

The projected nuclear capacity for **OECD countries** in the year 2000 is 300,000 MWe (Reference 5). Most of this capacity is either installed or under construction and the remaining time will not permit any significant increases. This compares with projections of over 1,000,000 MWe made in 1977 and approximately 500,000 MWe in 1982. Obviously, the rate of expansion is slower than originally anticipated and a major driving force underlying the development of advanced fuel cycles has been considerably reduced. The oil shocks of the 1970s led many nations to introduce policies which promoted the replacement of oil by fission energy and resulted in projections of high growth rates for installation of nuclear power stations. In parallel, the extent of natural uranium resources and supply appeared inadequate to fuel this rapidly expanding system and significant demand/supply imbalances were anticipated in the early decades of the next century. Consequently, many nations accelerated their research, development and demonstration (RDD) activities on programs for spent fuel recycling and on Fast Breeder Reactors (FBR) in order to enhance the security of their future nuclear fuel supplies.

The actual growth in installed nuclear power, as illustrated above, has fallen far short of the projections of a decade ago and current projections of future growth are the lowest of the past decade. Consequently, there is now time to consider and develop strategies other than the thermal to fast breeder reactor transition which, in the past, seemed dictated by the demands of rapid expansion. A strategy in which there is evolution from the current thermal burner reactors to advanced converter reactors...
now seems possible with CANDU having the characteristics necessary for both roles (Reference 6).

Fast Breeder Reactor Deployment

As discussed above, one outcome of the slowdown in the pace of nuclear expansion has been a significant delay introduced into national planning for the deployment of fast breeder reactors. In the United Kingdom there have been large cuts in the funding for RD&D, and in France there has been a significant change in the policy regarding use of recovered plutonium. Plans are being implemented to recycle it into thermal reactors in contrast to the previous policy where it was reserved for exclusive use in FBR's. Similarly, in Japan, large scale deployment of FBR's is now planned for the period after 2020 rather than before.

Accumulation of Spent LWR Fuel

The final trend meriting attention is the growing stockpile of spent LWR fuel. From levels of approximately 60,000 Mg in 1988, the stockpile is projected to grow to 165-200,000 Mg in the year 2000 and the range 400,000 - 2,000,000 Mg by the year 2020. Some countries are continuing to reprocess their spent fuel as a matter of national policy, generally for waste management and resource recovery reasons. Others, however, are not doing this and the USA is pursuing a policy and program of spent fuel disposal.

One problem is that there is no economic incentive to reprocess and recycle. Lacking large-scale disposal facilities, this leads to large-scale accumulations of spent fuel in interim storage facilities. A major opportunity exists if a means can be found to economically recover the contained fissile values.

Discussion

Uranium resources appear adequate to sustain worldwide nuclear growth well into the first half of the next century. When the possibility of new discoveries and the potential of higher cost resources are added, then there does not appear to be a major incentive to move from once-through to more advanced fuel cycles. The supply/demand situation, however, is more uncertain and the possibility exists of increasing demand being out of phase with those activities (e.g. exploration, new mine development) necessary to bring increasing supplies to market. Some instability could arise with an associated "price roller-coaster". Utilities would then look to other sources of supply, particularly the potential of fissile values locked in the accumulating stockpiles of spent fuel.

The slower pace of nuclear power growth, the increasing delays in plans for large-scale FBR deployment, and the accumulating stockpiles of spent fuel, all point to an opportunity for an advanced converter reactor to fill the growing niche between the current LWR "burner" reactor and the FBR. CANDU using advanced fuel cycles could fill this niche.

FUEL CYCLE OPTIONS

The CANDU fuel cycle options of interest for the next few decades are illustrated on Figure 1. The existing fuel cycle for all operating CANDU's is the natural uranium once-through cycle. Characteristics of this cycle are the simplicity of the fuel design and fabrication, ease of storage of the spent fuel, and the overall excellent economics (fuel cycle unit energy costs are less than half those for LWR's).
in economic penalties in conversion, enrichment and fabrication. U-236 is a strong neutron absorber, and its presence requires additional enrichment (SVU) for compensation, another economic penalty in terms of both dollars and neutron economy. The softer spectrum in CANDU, however, reduces the relative absorption in U-236 to the extent that it has negligible burnup penalty. Once RU is burned in CANDU, the residual U-235 is below the level of enrichment tails, and hence there is no incentive for further recycle of that residual U-235.

This cycle could well be of interest to LWJ utility owners who reprocess their fuel and are faced with a mounting stockpile of recovered uranium.

TANDEM - Current

The Tandem cycle uses both the plutonium and uranium recovered from reprocessing spent LWR fuel. The combined fissile content of the mixture (approximately 1.5 wt% U235 and fissile plutonium) can be used directly as fuel for CANDU with a resultant average burnup in the range 25-30 MW.d/kg heavy elements.

The advantages and incentives to use the Tandem cycle are:

- twice as much energy can be obtained from use in CANDU compared to use in LWR's. There is no need for on-enrichment of uranium in an enrichment plant or blending with higher enrichment uranium. Consequently, for a growing nuclear electric system, the growth in demand for natural uranium supplies and enrichment services is reduced.

- recovered uranium-plutonium-mixtures can be used directly as a fuel for CANDU without the need to adjust isotopic ratios. Consequently all recovered material can be used directly.

- long-term storage of CANDU spent fuel is a safe, proven, low-cost technology compared to LWR spent fuel storage, and no problems have been identified which should prevent application to CANDU tandem fuel.

- the U-236 poses no problems in tandem fuel cycles, and there is no risk of contaminating uranium enrichment facilities with this or other isotopes (e.g., U-232).

- there is the potential for further simplification of the fuel cycle processes (described below).

- there is good potential for economic competitiveness with natural uranium fuelling in both the near- and long-term, and hence there should be significant economic advantage relative to competing systems.

Again, this cycle could be of interest to LWJ owners faced with system expansion and wishing to diversify and secure the most efficient utilization of their recovered material supplies.

TANDEM - Future

This cycle is very similar to the potential "current" Tandem cycle, but takes advantage of the fact that CANDU can efficiently utilize the mixture of uranium and plutonium recovered from the spent LWR fuel without the need to adjust the uranium/plutonium ratio. This is essential if the material is to be reused in LWR's. If it were at all mechanically feasible, CANDU could burn unprocessed spent LWR fuel and obtain an economic cycle with a burnup of at least 16 MW.d/kg despite the presence of the original fission products. Feasible fabrication of CANDU fuel, however, requires some processing but this can be restricted to removal of fission products and does not demand separation and individual purification of the uranium and plutonium. Consequently "decontamination" i.e. only removal of fission products, can replace conventional reprocessing with the potential for significant simplification of the process and cost reductions.

This cycle offers some possibility that re-use of spent LWR fuel would become economically attractive and competitive with once-through uranium cycles and thus effectively utilize the fissile values accumulating in the spent fuel stock-piles.

Summary

Four potential fuel cycles have been identified for CANDU for possible implementation in the near-term. The SEU cycle is the logical evolution of natural uranium fuelling offering reduced fuelling costs, improved uranium utilization and reduced spent fuel volumes.

The other three cycles are aimed at utilizing the fissile values in spent LWR fuel. Compared to the alternative of recycle into LWJ's, these cycles offer the potential of increased energy yields, lower energy costs and simplicity in implementation.

DISCUSSION AND CONCLUSIONS

The future evolution of CANDU advanced fuel cycles and those for LWJ's are given on Figure 2. The current once-through fuel cycles are given at the top of the figure, and progression down the figure leads to fuel cycles giving more efficient fuel utilization.

This list of options is virtually unchanged from that discussed by W.B. Lewis twenty-five years ago (Reference 1). What has changed in the interim is the perspective on the necessary pace of transition from current to advanced cycles. In the mid-1970's, the projected growth was so rapid that it was believed that, for CANDU, there would be no rapid transition from natural uranium to thorium fuelling, skipping over SEU and mixed-oxide recycle. Similarly a rapid LWR to FBR transition was anticipated. Nowadays, however, it is expected that there will be a more orderly transition. The pace of change will also be affected by the perceived economic competitiveness of the advanced cycles, since there is reduced pressure to secure energy supplies at any cost.

Development of SEU cycles and the CANDU-FLEX fuel bundle is the next logical step for CANDU. The demonstration of high burnup cycles is an essential step for all potential advanced cycles in CANDU and must be executed as soon as possible.

One new factor is the potential symbiotic relationship between CANDU and LWJ's. The driving tie is the superior performance of CANDU in utilizing the LWR spent fuel. The perceived inefficiency of the LWR fuel cycle forced many countries into heavy investment in the development of fast breeder reactors. It was thought that these new reactors would be needed to replace LWJ's before the end of the century and would
be fuelled by material recovered from LWR spent fuel stockpiles. Recently, however, the reduced pace of nuclear power development, the increasing difficulty of developing a totally new reactor concept to the industrialization stage, and the disappointing economic potential of fast breeders, has now caused many of these same nations to defer consideration of fast breeder introduction for several decades. This move has further highlighted the problems which are posed by the growing stockpiles of LWR spent fuel. CANDU tandem cycles offer a solution using a proven reactor technology and a range of fuel recycle options which, in some cases, utilize modifications of existing or near-developed technology. This opportunity, however, will not remain open for ever. The steady pace of commercial demonstration of recycle into LWR will eventually overwhelm any Tandem cycle potential, if it is not also demonstrated in a parallel fashion. Consequently there is an urgent need to investigate this potential application in cooperation with LVR owners and secure a greater international recognition of the benefits and potential role for CANDU and its advanced fuel cycles.

FIGURE 2

THE CANDU STRATEGY

CANDU

NAT. U

O.T.

LEU

O.T.

MOX RECYCLE

HIGH BU THORIUM

U RECYCLE

MOX RECYCLE

ACCELERATOR BREEDER

FUSION BREEDER

LWR-LMFBR

CURRENT O.T.

EXTENDED B.U.

TANDEM

MATERIAL

(POWER FACTORY)

(FISSILE SOURCE)

REFERENCES

(1) LEWIS, W.B., "How much of the rocks and oceans for power? Exploiting the uranium-thorium fission cycle", 1964, April.


ABSTRACT

The characteristics of Advanced Fuel Cycle reactor lattices have been exploited for the optimization of CANDU power plant design. Reductions of 15.8% in specific capital cost and of 18.4% in total unit energy cost are predicted.

INTRODUCTION

Studies on reduction in the capital and operating costs of CANDU power plants is a continuing activity at AECL-CO. For the next generation of CANDU power plants, cost reduction effort has been focussed on shortening of construction time and consequent reduction in the interest on invested capital. This shortening has been achieved mainly by:

a. Modularization of equipment to increase the amount of off-site construction

b. Simplification of design, and

c. Reduction in the number of components such as steam generators, pumps valves etc.

In this paper we describe a study that exploits the Advanced Fuel Cycles to achieve further reduction in capital and operating costs.

In the past, such studies have focussed on the use of AFC to improve fuel utilization and thereby reduce plant fuelling cost. However, the present CANDU plant cost breakdown is such that fueling costs represent only 12% of Total Unit Energy Cost. A major part, 58% of TUEC, is the interest and depreciation of capital. Operating and Maintenance costs account for the rest.

The relatively low fuelling cost of CANDU implies that the reduction effort should focus on capital cost. However it is still possible to reduce fuelling cost up to 30% by using AFC. This represents a 4% reduction in TUEC. In comparison a 30% reduction in plant capital cost (although not easily envisaged) would reduce TUEC by 18%. This paper identifies methods of exploiting AFC to reduce plant capital cost and gives results of an optimization study carried out for a CANDU-6 plant that uses Recovered Enriched Uranium. The methods of cost reduction used, however, apply to a variety of AFC.

SCOPE OF CAPITAL COST REDUCTION

There are two features of AFC that determine the scope of capital cost reduction:

1. The AFC lattice is significantly more reactive than the natural uranium lattice. This permits changes in the reactor and fuel design that hitherto were unacceptable from the viewpoint of neutron economy.

2. The range of fissile content possible with AFC fuel provides novel methods of power flattening, both globally over the reactor and locally across the fuel bundle.

The application of these features has been studied to reduce cost by:

1. Increasing pressure tube thickness (which becomes possible due to the increased reactivity of the AFC lattice) to upgrade Primary Heat Transport conditions and thereby achieve higher thermodynamic efficiency.

2. Increasing the power output of outer channels of the core by increasing their fissile content.

3. Increasing the power output of the inner fuel elements of the fuel bundle throughout the core by increasing their fissile content.

4. Adding burnable poison to the fuel to eliminate refuelling ripple and thereby permit higher power output for a given maximum channel power.

5. Reducing moderator inventory by decreasing the lattice moderator volume and the reflector volume (made possible by the high reactivity of the AFC lattice).

Several other applications with less tangible cost reduction implications have also been studied. These are:

1. The use of neutron absorbers added to the fuel to introduce inherent safety features to the CANDU lattice.

2. The introduction of passive safeguards to reduce safety system requirements and cost.

DESIGN BASIS

The CANDU-6-II Plant design (1) was the starting point of the optimization study. The following design changes were imposed:

1. The CANFLEX fuel bundle design (2) was adopted. This design provides higher bundle power limits. It is also expected to perform better at the high fuel burnups which are to be expected with AFC.
2. Other relevant parameters are given in Table 1. The composition of the REU fuel is shown in Table 2.

POWER UPRATING

THERMODYNAMIC EFFICIENCY

PHT conditions that maximize thermodynamic efficiency were available from optimization studies for CANDU plants. Based on these results it was decided to increase the PHT system pressure at the outlet header to 12 MPa and the temperature to 323.6°C. These conditions provide the maximum increase in thermodynamic efficiency while maintaining the coolant at saturation conditions. The increase of 1.5 percentage points in efficiency raises the electric power output by 5%.

PRESSURE TUBE THICKNESS

The pressure tube was re-designed to accommodate the uprated PHT conditions. Assumptions regarding PT strain limits, corrosion and wear allowance, creep rate increase due to higher temperature and methodology were based on established criteria. The pressure tube thickness necessary for the new PHT conditions is 0.508 cm.

POWER FLATTENING

Increasing the fissile content of the fuel in the outer channels is an effective method of raising the power of the outer channels. This uprates the reactor power level without exceeding channel power limits. Several fissile content distributions over the core were studied to maximize the power of the outer channels of the core. A core configuration having four levels of fissile content was selected. The radial power form factor for this configuration is 0.929.

BURNABLE POISON

The relatively high fissile content of fresh fuel is responsible for the power ripple due to refuelling. The size of the power ripple and consequently the reactor power derating required to stay within channel power limits is high for AFC compared to the natural uranium cycle. The reactivity potential of the AFC lattice permits the addition of burnable neutron absorbers to the fuel and eliminate the power ripple. No derating of reactor power is then required due to power ripple.

Several burnable poisons have been studied for this purpose. They have to satisfy multiple requirements such as:

1. The poison burnout rate should match the reactivity load that builds up on refuelling due to xenon-135 and other saturating fission products.
2. The reactivity rise due to poison burnout should match the reactivity drop due to fuel depletion.
3. The energy dependence of the absorber cross sections should be such as to enhance the inherent safety of the CANDU lattice. This is achievable by increasing the negative feedback reactivity due to fuel temperature rise and also by reducing coolant void reactivity.

Several absorbers have been tested to meet these requirements. The most promising ones belong to the rare earths family. A combination of gadolinium, dysprosium and samarium oxides seem to fulfill all the above requirements.

The reduction of refuelling ripple allows a reactor power increase of between 8 to 12% depending on the bundle shift scheme used.

BUNDLE ENRICHMENT GRADING

Adopting the AFC provides a method of reducing the Maximum Linear Heat Generation Rate by increasing the relative fissile content of the inner elements of the fuel bundle where the neutron flux is lower. Optimization of enrichment grading was carried out to minimize MLHGR over the life of the fuel. The relative fissile content of the inner vs the outer ring of fuel elements was adjusted to maximize the power flattening across the high powered bundles of the core. A maximum to average pin power of 1.04 was achieved compared with 1.13 in the case of fresh fuel. This provided the potential to uprate reactor power by 8.0% if MLHGR was the power limiting parameter. As it turns out, in this study the reactor power was set by the channel power limit of 8.1 MW (th).

UPRATED REACTOR POWER LEVEL

The increase in radial form factor due to the higher fissile content of the outer channels contributes 10% to the power uprating. With refuelling ripple absent due to the use of burnable poison, the thermal power output of 392 channels, limited by a maximum channel power of 8.1 MW(th) is 2950 MW(th). With the increased efficiency this translates to a net electrical output of 972 MWe.

HEAVY WATER INVENTORY

Reduction in the ratio of the moderator-to-fuel volume becomes feasible with a high fissile content fuel. The reduction is limited by the minimum lattice pitch that is required for considerations such as feeder placement. Based on this, the maximum reduction that might be achieved is 0.6 cm.

Further reduction in moderator inventory can be achieved by increasing the size of the calandria tubes. The increase is however limited by the presence of reactivity devices. Based on such an evaluation it was found that the calandria tube O.D. can be increased to 15.7 cm from 13.2 cm.

In addition, the reflector thickness was reduced to 39.7 cm from 65.4 cm. The total reduction in inventory achieved is 40,000 kgm or between 15 to 20% of the inventory in the calandria. Increase in moderator system inventory due to higher heat generation in the moderator will reduce this figure slightly but has been neglected for this stage of the optimization.
EXIT FUEL BURNUP

Fuel Management studies with 3-D models were used to calculate exit fuel burnup.

The exit fuel burnup predicted for the optimized reactor, after including penalties for the increased neutron absorption and leakage, is 12,750 MW.d/te(U). Fuel burnups in this vicinity have been achieved regularly by CANDU fuel in Bruce A.

Table 3 lists the optimized parameters.

COSTS
CAPITAL COST

The reduction in capital cost per installed kilowatt is obtained as follows:

1. The cost of the additional heat removal is based on assigning 15% of the plant capital cost to heat removal systems. This 15% is made up of:
   a) PHTS and Auxiliaries: 3% of plant capital cost
   b) PHTS and Auxiliaries installation: 3% of plant capital cost
   c) T/G and Condenser: 7% of plant capital cost
   d) T/G and Condenser installation: 2% of plant capital cost.

   The 22.7% uprating of power output (from 2404 to 2950 MW) will therefore increase capital cost by 15% of 22.7% or 3.4%.

2. The capital cost is decreased by the saving of 40 te of heavy water inventory. At a cost of $275/kg the saving is 1%.

   Based on the above, the capital cost of the AFC-optimized plant is 15.8% lower. It is assumed in this analysis that the interest on capital in TUEC can therefore be reduced by 15.8%.

FUELLING COST

The reduction in fuelling costs are based on

a) The REU cost being taken to be the same as the natural uranium cost.

b) The fuel fabrication cost being increased to the value estimated for SEU fuel (This is an optimistic assumption) and

c) The fuel utilization being increased according to the burnup increase from 6300 to 12700 MW.d/te(U).

   The increase in fabricatio... of SEU is $20/kgU. This is an increase over the natural uranium fuel fabrication cost and represents an increase of 1.3 m/kWh in TUEC.

   The increase in fuel utilization reduces the fuel bundle usage per kWh by 50%. The net effect of b and c is to reduce fuelling costs in TUEC from 9 m/kWh to 5.1 m/kWh.

SPENT FUEL MANAGEMENT

The reduced volume of fuel per kWh reduces the spent fuel management cost in TUEC by 50% (Table 4).

TUEC

Based on the above, the estimated reduction in TUEC is 13.8 m/kWh.

It is likely that there are several cost components that have been overlooked in this estimation. But it is most likely that they are minor. Reduction in equipment size due to the uprated PHTS conditions is one component that will further reduce TUEC.

TABLE 1

<table>
<thead>
<tr>
<th>PLANT OPTIMIZATION STUDY ASSUMPTIONS</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Fuel: Recovered Uranium</td>
</tr>
<tr>
<td>2. Fuel Design: CANFLEX</td>
</tr>
<tr>
<td>3. No. of Channels: 392</td>
</tr>
<tr>
<td>4. Channel Power Limit: 8.1 MW(th)</td>
</tr>
<tr>
<td>5. MLHGR Limit: 58.5 kW/m</td>
</tr>
<tr>
<td>6. Power Flattening:</td>
</tr>
<tr>
<td>Radial: 4 Burnup Zones</td>
</tr>
<tr>
<td>Axial: Adjuster Rods</td>
</tr>
<tr>
<td>Bundle: Burnable Poison</td>
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<tr>
<td>7. Reactor Power: Maximum Allowable</td>
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</table>

TABLE 2

<table>
<thead>
<tr>
<th>COMPOSITION OF RU</th>
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</thead>
<tbody>
<tr>
<td>wt%</td>
</tr>
<tr>
<td>U234:</td>
</tr>
<tr>
<td>0.020017</td>
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<td>0.000996</td>
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<td>U238:</td>
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<td>98.5712</td>
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</table>

TABLE 3

<table>
<thead>
<tr>
<th>OPTIMIZED PLANT</th>
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</thead>
<tbody>
<tr>
<td>1. Lattice Pitch: 27.975 cm</td>
</tr>
<tr>
<td>2. P.T. Thickness: 0.508 cm</td>
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<tr>
<td>3. C.T. O.D.: 15.715 cm</td>
</tr>
<tr>
<td>4. PHT System Pressure: 12 MPa at O.H.</td>
</tr>
<tr>
<td>5. Coolant Temperature: 323.6°C at O.H.</td>
</tr>
<tr>
<td>6. Reflector Thickness: 39.7 cm</td>
</tr>
</tbody>
</table>
7. Bundle Shift Scheme: 4 B.S. Regular
8. Burnable poison: <1% Natural Rare Earths in Fuel
9. Maximum Channel Power: 8.1 MW(th)
10. Reactor Thermal Power: 2950 MW
11. Increase in Thermal Efficiency: 1.5% Points
12. Net Electrical Output: 972 MWe
13. Average Exit Fuel Burnup: 12.750 \( \text{MWd/te(U)} \)
14. MLHGR: 48.5 kW/m

TABLE 4

COST REDUCTION DUE TO USE OF REU IN CANDU

<table>
<thead>
<tr>
<th></th>
<th>%</th>
</tr>
</thead>
<tbody>
<tr>
<td>Depreciation</td>
<td>22.4</td>
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<tr>
<td>Interest</td>
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<tr>
<td>Operating &amp; Maintenance</td>
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<tr>
<td>Fuelling</td>
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<td>Spent Fuel Mgmt</td>
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<tr>
<td>Decommissioning</td>
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<tr>
<td>TUEC</td>
<td>18.4</td>
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</table>

SUMMARY

In summary, a 16% reduction in specific capital cost and a 18.4% reduction in TUEC seems possible if a plant is optimized for the use of REU.

Furthermore, the world inventory of REU is growing at an increasing rate with no definite proposals of how it could be utilized. The AFC optimized CANDU plant seems to be the most promising method available for the efficient use of REU.

REFERENCES


ACKNOWLEDGEMENTS

The authors gratefully acknowledge the efforts of:

1. Dr. P.S. Narayanan for providing the Optimum PHT System Conditions.

2. Messrs. S. Dua and C.C. Yao for calculating the pressure tube thickness for the uprated PHT conditions.

3. Mr. P.S.W. Chan for the lattice calculations using burnable poison and those for enrichment grading.

4. Dr. K. Hau for lattice pitch reduction estimates.

5. Dr. A.M. Yu for providing cost components for the CANDU-6-II.

* Mr. A.M. Manzer for providing fuel fabrication JStS.
Advanced fuel cycles for CANDU reactors are well on their way to being implemented. The first step is slightly enriched uranium (SEU) which is economical today. A new fuel bundle is seen as the vehicle for all fuels in CANDU. CANDU fuel fabricated from uranium recovered from fuel discharged from light-water reactors (LWR) is also economical today and readily achievable technically. Future fuel cycles would utilize plutonium recovered from light-water reactors or CANDUs and eventually thorium. R&D in support of these cycles focuses on those topics that require a high degree of confidence in their implementation such as fuel fabrication and defect-free performance to high burnup. Reactor physics codes and nuclear data for advanced fuel cycles will be validated against experiments.

CANDU reactors operating with once-through, natural-uranium fueling are now well established. They provide electricity at fuel-cycle costs that are typically half of those arising from other commercial reactor types. The low cost of the CANDU fuel cycle can be attributed to several factors, including heavy-water moderator and coolant; on-power refueling; simple, robust fuel bundles fabricated on a mass-production basis; minimization of neutron absorption in structural materials in the reactor core; and technically straightforward storage and disposal of the irradiated fuel. All of these factors are equally applicable to advanced fuel cycles in CANDU and promise to continue to yield fuel cycle costs well below those of competing reactor concepts.

CANDU advanced fuel cycles cover the range from slightly enriched uranium (SEU) or recovered uranium (RU) through mixed oxide plutonium/uranium cycles (MOX and TANDEM) to thorium cycles with their promise of near-breeding. In this paper MOX refers to the recycle of plutonium retrieved from discharged CANDU fuel, while TANDEM refers to fuel based on the uranium and plutonium recovered from discharged light-water reactor fuel. Conceptual fuel-cycle studies for CANDU using MOX fuel have been completed, as well as the demonstration of the fabrication route, fuel testing in research reactors, and zero-energy reactor physics experiments. For thorium fuels, conceptual fuel cycles have been investigated, fuel has been fabricated and tested in research reactors, laboratory-scale reprocessing has been demonstrated, and zero-energy reactor-physics experiments are being performed using (U233, Th)O2 fuel. The introduction of these cycles depends on such factors as their economics, as well as strategic or political considerations. At this time, only fuel cycles using enriched uranium (SEU or RU) are economically attractive.

The SEU cycle uses uranium enriched to about 1.2%, is characterized by good economics and improves the already excellent uranium utilization of CANDU by approximately 30%. The RU cycle, known as CANFURL, uses the uranium component of fuel discharged from LWRs after re-enrichment, when discharged from CANDU has a U235 content above that of enrichment plant tails, and hence there is no incentive for further recycle of the uranium. RU can be recycled in LWRs after re-enrichment. The uranium component of this second recycle uranium, after discharge from an LWR, can also be used in CANDU; however, because of the associated radioactivity it is unlikely that it could be enriched once again for further LWR recycle.

The development program, currently being pursued and planned leading to implementation of these fuel cycles, is the subject of this paper.

INTRODUCTION

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Slightly Enriched Uranium. The SEU cycle is based on slightly enriched uranium with a U235 content of about 1.2%. Burnups of around 22 Mwd/kg U are expected to be achieved with this fuel cycle, which is about three times the burnup routinely achieved in current LWRs fuelled with natural uranium. The advanced CANFLEX bundle is being developed for the implementation of this cycle in existing or future CANDUs. CANFLEX is an acronym for CANDUn FLEXible, where flexible refers to the versatility of the bundle with respect to fuel cycle and operating conditions. It will be necessary to establish an SEU fuel-fabrication line of commercial scale.

The CANFURL Fuel Cycle. Recovered uranium (RU) from current LWRs can be used to fuel both CANDUs and LWRs. The feasibility of using fuel from extended-burnup LWRs remains to be proved. For RU containing about 0.9% U235, a burnup of about 13 Mwd/kg U is expected to be achieved in CANDU.

Due to the presence of the neutron-absorbing isotope U236 in this material, the enrichment level needed to use it in LWRs is...
greater than that required for fuel derived from natural uranium. An economic penalty is thus incurred. Further, because of the presence of U232, or more precisely, the radioactive decay products of this isotope, enrichment may not be possible in a diffusion plant because of possible contamination. This may be less of a problem in centrifuge enrichment plants, where some cascades of centrifuges could be dedicated to this material. The problem of contamination may, however, result in further economic penalties. After irradiation of the RU in an LWR, the discharged fuel has a uranium enrichment which is higher than that in fuel derived from natural uranium, and there is again incentive to recycle this material. However, the U236 content in this second recycle material is even higher, resulting in the need for higher enrichment levels and greater economic penalty. More important, however, is the increased level of U232, which makes the enrichment step unattractive.

RU can be used directly in CANDU without enrichment, and after irradiation the U235 content is about the level of enrichment plant tails. There is, then, no incentive to further recycle this uranium. Because there is no requirement for enrichment, it is also possible to use second recycle material in CANDU. The U236 present in RU results in a burnup which is less than that obtainable with fuel of the same enrichment derived from natural uranium. For uranium containing about 0.9% U235 and 0.4% U236, the burnup loss is about 0.6 Mwd/kg U.

There may be an economic penalty to the recycle of RU in either CANDU or LWR arising again from the radioactive daughters of U232, but in this case the impact is on the cost of fuel fabrication. This is being addressed in the fuel-fabrication program.

The decision to use RU will rest primarily on economic arguments and the availability of the material to a utility. Because of the burnup penalty in CANDU arising from the presence of U236, a CANDU utility would buy RU for use in its reactors only if the price is attractive relative to the alternatives, i.e., natural uranium or SEU. Utilities owning both CANDUs and LWRs will find the use of RU in CANDU most attractive, primarily due to the omission of the enrichment charges. LWR utilities owning RU may find it attractive to install CANDUs to use this material, provided it results in an economic advantage over the alternatives of re-use in LWRs, continued stockpiling, outright sale, or disposal. Some countries have legislated fuel reprocessing and recycle, almost without regard to cost.

The TANDEM Fuel Cycle. The TANDEM fuel cycle uses fuel which incorporates both the uranium and plutonium components of reprocessed LWR fuel. This material, which is about 1.5% fissile, would be mixed with natural uranium in a 2 to 1 ratio to produce a fuel that, in CANDU, would produce a burnup of about 22 Mwd/kg U.

A potentially attractive feature of the cycle is that it is not necessary during the reprocessing to separate the uranium from the plutonium, as is the current practice. This decontamination process could result in lower reprocessing costs in a new plant designed for that purpose. Existing plants, designed for the separation of plutonium, may not show a cost advantage for the decontamination process. An additional element to the advanced fuel-cycle program would be to determine whether or not such economies would occur. Lower reprocessing costs are not an important feature of the TANDEM cycle because many utilities who already reprocess fuel charge the cost of reprocessing against the electricity cost of the station generating the fuel, and the reprocessed material is then essentially free to the next fuel cycle. In this case, if decontamination was cheaper than reprocessing, it would result in cheaper power from the LWR, but not from CANDU.

While the current reactor-physics methods are expected to be adequate for design and feasibility studies, it will be necessary to perform zero-energy experiments in a facility such as ZED-2 to verify key reactor-physics parameters before full-scale irradiation testing.

DESCRIPTION OF THE R&D PROGRAM

Overview

The program presently in place addresses the SEU fuel cycle in CANDU and includes fuel fabrication, fuel design, fuel testing, reactor physics, safety, fuel disposal, and licensing. In most respects the CANFURL cycle, with its lower enrichment, can be regarded as a less demanding variation of the SEU cycle. Results from the SEU program will be directly applicable to the CANFURL and other advanced fuel cycles. There is, however, one aspect of the CANFURL cycle that requires additional program components, and these are now being put in place. Uranium from LWRs contains a larger amount of U234 than fresh uranium, as well as trace amounts of U232 and residual fission products. Since the radioactivity associated with these isotopes may complicate the fabrication and subsequent handling of this fuel, additional program components are needed to assess these complications. As well, the economics of using recycled uranium in CANDU must be compared with the economics of re-enrichment and subsequent utilization in LWRs.

Fuel Development

Overview. Fuel development directed toward the use of various fuel cycles in CANDU reactors has been in progress for over 20 years. This work has generally been driven by various economic analyses and studies. It has included work on many cycles, including MOX, TANDEM, (U235, Th)O2, (Pu, Th)O2, (U233, Th)O2, SEU and CANFURL. It has included fuel in many forms of oxide, including conventional sintered pellets, spherexcavated pellets, extruded slugs and coated particles, plus some non-oxide forms such as carbides and silicides. The areas of fuel development covered in this work include powder processing, fabrication (particularly for novel formats, and α-toxic or γ-active fuels), irradiation behaviour (particularly extended burnup and power cycling) and behavioural modelling.

The unpredictability of the time scale for the economic use of a particular fuel cycle, the differences in incentives for the use of specific fuel cycles in different parts of the world, plus the many, large, high-grade uranium deposits discovered recently in Canada, have led to a fuel development plan based on the following premises:

- The plan should, where possible, be evolutionary, so that the necessary technologies can be developed in a logical sequence from established and simple through to the more complex, the more technologically difficult and the more expensive.

CNS 10th ANNUAL CONFERENCE, 1989, 1-11
The plan should concentrate on the introduction of the fuel cycles to CANDU reactors through analogous use, since the major factors influencing the operating technologies are the level of fissile content, the burnup, the cost of the fuel and the local economics for its use. For example, the extra fissile content in SEU fuel can establish the fuel-management technologies and demonstrate irradiation behaviour for most extra-fissile-content fuels such as CANFURL, MOX, TANDEM and, to a lesser extent, thoria.

The plan should make available the fuel-bundle technology that will meet the needs of all fuel cycles likely to be attractive within the foreseeable future, in market areas where the CANDU reactor is an attractive option.

The base of existing CANDU reactors should be used to demonstrate key aspects of the technological needs of the evolving fuel-cycle program, but in a manner that offers an economic incentive to the particular reactor operators involved.

Because of its high neutron efficiency, the CANDU reactor is an attractive option for burning low-enriched uranium from reprocessed LWR fuels. The plan should put in place the technologies which will support the implementation of such a cycle.

Therefore, the main elements of the fuel-development plan are as follows:

- Develop and demonstrate a new bundle design which will provide the flexible bundle vehicle for most fuel cycles. This design is known as CANFLEX.
- Demonstrate the use of SEU fuel in a commercial CANDU reactor and establish the necessary fuel-management technologies and licensing requirements.
- Have available the reactor-physics core parameters and codes for all likely fuel cycles to be used in CANDU.
- Have available the requisite fuel-behavioural data bases and modelling codes for the use of likely fuel-cycle fuels within the CANFLEX bundle, covering the anticipated range of operating conditions and burnups.
- Have available the fabrication technologies necessary to support the various fuel cycles in CANDU.

Develop and Demonstrate the CANFLEX Bundle. The CANFLEX bundle design is the central fuel technology for the use of advanced fuel cycles in CANDU reactors. The CANFLEX bundle is more subdivided than either of the currently used 37- or 28-element bundles, as it contains 43 elements in two element sizes. Compared with the current design, CANFLEX will have higher power capability (1250 kW versus 1035 kW in the current 37-element bundle) and higher burnup potential (in excess of 22 MWd/kg U), or some combination of the two. CANFLEX is a logical extension of existing technology: progressive designs have increased the number of elements on a CANDU bundle from 7 to 19 to 28 to 37, and now to 43.

In existing reactors, CANFLEX provides a 10 to 20% reduction in peak linear-heat ratings for a given bundle power (depending on burnup), compared with the 37-element design. Additionally, there is expected to be an 8% increase in critical channel power, and the bundle is compatible with existing fuel-handling systems. Some key points showing the wide application of CANFLEX are:

- At current bundle powers, lower heat ratings are made possible in both current and future reactors;
- In future reactors, power uprating without exceeding current heat ratings is made possible;
- The achievement of extended burnup is facilitated by lower heat ratings and optimization of internal design;
- Both natural and enriched fuel can be used;
- Operating margins can be increased;
- Power manoeuvring is facilitated.

Additionally, at current bundle powers, the lower fuel-operating temperature makes CANFLEX more tolerant in accident scenarios. The greater operating margins are a particular benefit in fuel-management operations required for advanced cycles, where they add a significant degree of additional scope and flexibility. Also, CANFLEX can facilitate a burnup increase by reducing the need for flattening the radial channel power distribution.

The predominant influence on high-burnup behaviour is the fuel-chemistry effect, which increases operating temperature for a given power rating at extended burnups. This is being addressed by membership in the International High Burnup Chemistry Program, coordinated by Belgonucleaire, and by a program of inactive testing of the effects of fission products on the conductivity and gas diffusion in UO	extsubscript{2}, using SIMFUEL technology. Power-ramp tests are planned at extended burnups to fill in gaps in the current data base. Six CANFLEX bundles for extended burnup testing at a maximum linear power of 65 kW/m (equivalent to 1250 kW bundle power) are being fabricated, for insertion in NRU during 1989. This first proof test will contain fuel pellets optimized with respect to geometry and fuel density. A further six-bundle irradiation, scheduled for 1991/92, will address further possible CANFLEX design modifications.

The critical channel power (CCP) can be increased significantly by the arrangement of boundary-layer tripers and structural appendages on the CANFLEX bundle, which will enhance the critical heat flux. A design has been conceived which maximizes enhancement while minimizing the associated penalties in pressure drop and bundle construction cost. These anticipated advantages are in the process of being confirmed by testing in a Freon loop, which permits an assessment of the effect on critical channel power. Freon tests are more convenient than water tests and the results can be extended to water. An 8% improvement in CCP has been identified through analysis, and will be confirmed by the testing program. This program is designed to also produce thermohydraulic data for mixed channels containing both CANFLEX and 37-element bundles, to provide information for the licensing of the reactor transition from 37-element to CANFLEX bundles. The tests also cover a range of bundle flux depressions, thus providing data for a range of fissile contents in the CANFLEX bundle.
The CANFLEX bundle geometry was optimized by minimizing the peak linear element ratings for both natural and enriched fuels. Currently, fuel-management studies are either complete or in progress for several fuel-management options appropriate for 1.2% SEU, such as regular bundle shifting, variants of the checkerboard fuel-management scheme and axial fuel shuffling. The optimal fuel-management strategy depends on details of reactor design, such as the presence and location of adjuster rods, and whether fueling is with or against the coolant flow. The current fuel-management studies provide power envelopes for assessing fuel performance. Future physics effort will include benchmarking and validating the calculational methods being used for SEU fuel, including comparisons against low-power lattice experiments. Safety studies will cover all aspects of CANFLEX performance, with either SEU or natural uranium fuel.

As the CANFLEX bundle geometry differs from current bundles, and there are smaller-diameter elements in the outer periphery, the bundle must be subjected to the same range of vibration and endurance tests as has become standard for the 37-element bundle. The new bundle must also interface with both the current latch and side-stop fuel-handling systems, and be capable of resisting the hydraulic and fuelling-machine-induced loads during such operations. Representative bundles will be fabricated and used in the flow-testing program. Confirmatory fuel handling and strength tests are also scheduled.

Demonstrate SEU in a Commercial CANDU Reactor.

A commercial demonstration is required in a CANDU power reactor for the following reasons:

- To demonstrate the feasibility of operating a CANDU reactor with SEU fuel.
- To establish the licensing procedures required for the introduction and steady-state use of SEU fuel in CANDU reactors.
- To demonstrate the adequacy and appropriateness of fuel-management procedures and spent-fuel handling developed for both the introduction and steady-state use of SEU in CANDU reactors.
- To serve as a focus for the establishment of a SEU fuel capability in the CANDU fuel-fabrication industry.

The CANFLEX bundle must also be demonstrated on a commercial scale in a CANDU reactor. Since the CANFLEX bundle is expected to be the major vehicle for the use of SEU and other fuel cycles in CANDU reactors, it would be desirable to combine both of these demonstrations into one, though it would be possible to demonstrate CANFLEX with natural uranium. The SEU demonstration has the added advantage of providing the maximum financial benefit to the utility undertaking it.

The necessary data base, small-scale demonstrations, and fuel-modelling codes will be available to support the commercial demonstration of SEU in a CANDU reactor by 1992/93. The actual time at which such a demonstration would start will depend upon the agreement reached with a utility partner, and this has not yet been finalized. It is also possible that the CANFLEX bundle, without enrichment, may be demonstrated first, depending upon the interest of the operator(s) involved. Studies for the transition of CANDU reactor cores from natural uranium to SEU indicate that a radial replacement pattern, starting at the outside, is the most appropriate way to introduce some SEU bundles into a power reactor.

Fuel Fabrication.

Overview. A significant fraction of the CANDU support effort on fuel, over the past 20 years, has been devoted to fabrication development work specific to CANDU fuel requirements for a number of fuel cycles. This work has focused principally on variants of the U-Pu and thorium cycles, since the fabrication of enriched conventional UO2 and UC fuel was done routinely for the production of fuel experiments supporting Candu! fuel. There has been a three-fold need for this general fabrication-development work:

- To fabricate sufficient quantities of \((U, Pu)O_2\), \((Pu, Th)O_2\) and \((U233, Th)O_2\) elements and bundles to assemble a core loading of each in the ZED-2 reactor for reactor-physics measurements.
- To sufficiently develop those fabrication technologies that are particularly appropriate to the CANDU reactor so that a realistic assessment of relative commercial feasibility and cost can be made, and to clearly identify those areas where further technology development is necessary. This work has investigated conventional pelleting of \((U, Pu)O_2\) and both \((Th, U)O_2\) and \((Pu, Th)O_2\) plus a number of novel fabrication techniques based on sol-gel technologies (spherap, extruded clays and pelletized microspheres) and the impregnation of green or partially sintered pellets, most of which were developed specifically for thorium-based fuels.
- The fabrication for irradiation testing of conventional pelletized fuel, plus samples of the various novel fuels to generate a behavioural data base.

Most of the objectives of this general fuel-fabrication development program have now been met, and fabrication is focused now on the near-term introduction of both the CANFLEX bundle and SEU fuel into power reactors. But as noted above, there is little development work needed for the fabrication of conventional UO2 pellet fuel using SEU, since all of the conventional CANDU fuel fabricated for irradiation testing in experimental reactors was and is enriched. Instead, the focus is more on the fuel element and bundle integrity to high burnups and with significant power cycling, and this is a major focus on the CANFLEX program.

CANFLEX Fabrication. The CANFLEX bundle is intended to provide the structural integrity required for most fuel cycles by accommodating burnups which are increased by a factor of three or more, and providing additional operating margins, and flexibility so as to accommodate more complex (and possibly more demanding) fuel-management schemes.

This program of work includes producing the reference design drawings for the CANFLEX bundle, the development of fabrication technologies required for the bundle, the production of bundles required for testing (in conjunction with commercial fabricators), the assembly of a CANFLEX bundle specification for use in future orders, and the control of key CANFLEX design features. The basic CANFLEX design drawing has been finalized, and Zircaloy tubes for development and fabrication received. Closure and assembly weld development, specific to CANFLEX requirements, is currently in progress and will be completed in
time for the fabrication of irradiation test bundles in 1989. The CANFLEX irradianes will be used to qualify the various specifications and fabrication features specific to the CANFLEX bundle.

SEU Fabrication. While the fabrication technology for conventional enriched UO2 fuel pellets for use in CANDU is well known, it has been established only on a relatively small-scale basis. The Canadian fuel fabrication industry will need to set up for large-scale commercial production, handling and accountability for SEU fuel.

RU Fabrication. RU is essentially SEU with an enrichment of 0.9% U235. However, it is complicated by the different concentrations of some uranium isotopes, notably U232, U234 and U236, plus trace quantities of some fission products (particularly those that are volatile at elevated temperatures, such as Cs or Sr). The major difference posed by RU from a fabrication viewpoint is thus centred on the radiological hazards that its use may pose in a normal CANDU-fuel-fabrication plant, and the impact that any changes required to eliminate any radiological hazards would have on the cost of producing RU fuel. Significant laboratory work and analyses have and are being undertaken on this problem, and the conclusion is that the magnitude of any such problems are a function of the concentration of fission-product contaminants in the RU as received from the reprocessing, and the chemical form of the uranium supplied. This is because most fission products partition to some extent during any changes in chemical form (i.e., during conversion from nitrate to oxide), and those that are volatile can be driven off during calcination and sintering operations, where they could be concentrated in the furnaces involved.

A realistic specification is needed for all important isotopes in RU to be used for the fabrication of CANDU fuel. To generate such a specification, representative samples of RU product from probable suppliers must be analyzed and characterized. Also, the relative economics of recycling RU in CANDUs and LWRs need to be assessed. AECL and COGEMA are presently discussing a joint study of these topics.

Therefore, the strategy for RU fuel development is to identify a specification for acceptable RU material as both U3O8 and UO2, and to identify what changes would be required to a standard CANDU-fuel production plant to accommodate RU meeting this specification.

Uranium-Plutonium Fabrication. Considerable experience has been gained in the fabrication of MOX fuels and in the operation of a glove-box-based, pilot-plant, fabrication facility. Studies of the commercial application of this technology have been completed to serve as a basis for determining relative cost.

However, these studies have shown a large cost increase in fabrication over SEU, and new technologies based on an a-cavern concept rather than glove boxes will probably have to be used to reduce these costs. Design studies of such facilities have been undertaken by engineering consulting firms and indicate the potential for significant savings over glove-box-based facilities. Cost analyses are available for both types of facility.

Thorium Fabrication. The fabrication of thorium fuel with recycled U233 has both an alpha-toxicity hazard plus a potentially high-level alpha activity hazard, and is thus more complex and expensive to fabricate than U-Pu fuels. The work on thorium fuels has thus investigated a wide range of novel fuel formats that could reduce these problems and therefore the cost. The development work on thorium fabrication has looked at most of these novel formats on a laboratory scale, and produced fuel for irradiation testing to establish its viability. However, little has been done in the line of engineering cost studies on potential fabrication plants using these various technologies.

Reactor Physics

Experimental Reactor Physics[8] The major experimental facility is the zero-energy reactor ZED-2 at the Chalk River Nuclear Laboratories. This facility is used to generate flux and fission activation distributions in and around a fuel channel and to obtain reactivity information, such as the material buckling, for different fuels and coolants. The buckling information can be converted into information on reactivity coefficients.

In the past, a series of measurements on natural uranium fuel in 7-, 19-, 28- and 37-element bundles was completed. More recently, experiments using 36-element MOX fuel and 36-element (Pu, Th)O2 were completed, and are present a program of measurements on 36-element (U233, Th)O2 fuel is in progress.

Most of the early experiments used a full core of the experimental fuel and the geometric buckling (equal to the material buckling for a critical core) was obtained from analysis of neutron-flux distributions. Latterly, due to the high cost of experimental fuel, only sufficient fuel for a few channels has been available. The buckling in this case is inferred from the critical height and neutron-flux distributions in a series of experiments in which the experimental fuel replaces progressively the fuel in a reference core. Buckling measurements of this type are inherently less accurate than those obtained with a full core of fuel. Uncertainties are also introduced into the comparison between calculated and measured detailed neutron-flux and reaction-rate distributions due to the possibility that the neutron spectrum in the experimental fuel is not truly representative of that which would be present in a whole fuel core.

Before advanced fuels are introduced into power reactors, it will be necessary to perform ZED-2 experiments to confirm the computer-code predictions. It would be preferable if these experiments were done with a whole core of fuel. Therefore, these experiments will not be performed until enough fuel from the large-scale demonstration is available to provide a core for ZED-2. The schedule is therefore dependent on other parts of the overall program.

Code Development and Nuclear Data[8] A supporting activity is the provision of codes and associated nuclear data. The goals are to improve calculation methods and nuclear data and so reduce conservatism in reactor design by:

- Maintaining, developing and acquiring improved calculation methods for the design, costing and prediction of the behaviour of CANDU,
- Providing improved nuclear data.
Validating the improved codes and data against experimental results.

Uncertainty in nuclear data translates directly into conservatism in design ratings, trip settings, amounts of shielding and estimates of radiation dose to structural materials. Continued association with the United States data evaluation group gives access to the most up-to-date evaluated nuclear data and processing methods. To give confidence in the code predictions and to identify and quantify code inaccuracies, a continuing process of code validation against experimental data has been established. This uses a database of experimental information from experiments in ZED-2 and worldwide. Validation of codes and data contributes to the quality assurance (QA) requirements of the nuclear industry.

CONCLUSION

Advanced fuel cycles for CANDU reactors are well on their way to being implemented. The first step in this progression is slightly enriched uranium (SEU), which is economical today. A new fuel bundle, CANFLEX, is the vehicle for SEU and other fuels in CANDU. CANDU fuel fabricated from uranium recovered during the reprocessing of fuel discharged from light-water reactor discharged fuel is also economical today and thought to be readily achievable technically. Future fuel cycles would utilize plutonium recovered from light-water reactors or CANDUs and eventually thorium. R&D in support of these cycles is driven by the marketplace and focuses on those topics that require a high degree of confidence in their implementation, such as fuel fabrication, defect-free fuel performance to high burnup under realistic fuel-management conditions, and licensability.

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THE ROLE OF THE NEW CANFLEX FUEL BUNDLE IN ADVANCED FUEL CYCLES FOR CANDU REACTORS

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ABSTRACT

The fuel cycles of principal interest for use in CANDU reactors are: slightly enriched uranium (up to approximately 1.2 wt% U-235), various cycles using the U-235 or Pu recovered from spent light-water reactor (LWR) fuel, and the high-burnup thorium fuel cycles. In all of these cycles the fuel bundles contain more fissile material than with natural uranium fuel in order to achieve the higher burnups required for optimum economics. This higher burnup results in the fuel being subjected to power ramps and high fuel ratings at burnups higher than are adequately covered by current CANDU fuel experience, and the extra fissile loading generally requires more complex fuel-management procedures, with the associated higher risk of greater localized power peaks than are currently acceptable. The new CANFLEX fuel bundle is the most appropriate vehicle for the optimal use of these fuel cycles in CANDU reactors because of the lower fuel ratings and enhanced operational flexibility which it provides. The peak fuel-element ratings are lowered by approximately 20% (dependent upon burnup and fissile content) through the use of smaller-diameter fuel elements than in current CANDU bundles. The flexibility is further enhanced by the thermalhydraulic optimization of the CANFLEX bundle to increase its critical heat flux (CHF) capability by approximately 15% above that of current CANDU bundles. The CANFLEX bundle is compatible with current CANDU reactors, so that it can be used to support advanced fuel cycles in either new or existing CANDU reactors.

INTRODUCTION

The CANDU* reactor was originally developed to allow the economical use of natural uranium fuel, and in order to achieve this, neutron economy had to be built into all aspects of the reactor's design and supporting technologies. While the heavy-water-moderated, pressure-tube design concept of the CANDU reactor has been extensively documented (1), it is important to note that the following features of the reactor are important to the subject of this paper:

(1) inherent neutron economy;

(2) on-power fuelling;

(3) short fuel bundles that allow axial fuel management; and

(4) low cost, simple fuel bundles with an excellent performance record.

These features give rise to fuelling costs which are typically half of those for other commercial power reactors, very high capacity factors, compact, easily stored fuel bundles, and low-discharge burnups compared to other reactors. In fact, seven out of ten power reactors with the best lifetime capacity factors are CANDU reactors*.

However, it is specifically these features that also make CANDU reactors such an attractive vehicle for the use of various advanced fuel cycles. The neutron economy of the CANDU reactor ensures that it will be the most efficient, and therefore potentially the most economical thermal reactor for operating advanced fuel cycles (2,3). The on-power fuelling and short fuel assemblies will allow more flexible and efficient fuel management than is possible in other reactor types. The simple, low-cost fuel bundles will allow lower fuel-fabrication costs for the various cycles based on recycled fuel (particularly those based on U-233, where gamma radiation is an important factor). The relatively low discharge burnup of natural uranium fuel from CANDU reactors allows those fuel cycles that operate at higher burnups to have significantly enhanced economics in a CANDU reactor. The following fuel cycles are of particular interest for use in the CANDU reactor and are being pursued for that reason (4-10).

The slightly enriched uranium (SEU) fuel cycle has an enrichment level ranging from 0.9 to 1.5 wt% U-235. This provides the potential for increasing the burnup (11-13) by as much as a factor of four over that in natural-uranium (NU) fuelled CANDU reactors.

The recovered uranium fuel cycle (RU) is a sub-set of the SEU cycle, and uses the enriched uranium recovered from the reprocessing of various light-water reactor (LWR) spent fuels. This fuel (14) would generally have an enrichment level of approximately 0.9 wt% U-235, but would be contaminated with other isotopes of uranium, and possibly trace quantities of some fission products.

*CANada Deuterium Uranium, registered trademark.

* Source: Nuclear Engineering International

1-16 CNS 10th ANNUAL CONFERENCE, 1989
Cycles based on recycling recovered plutonium with natural or depleted uranium are of wide interest for use in LWR reactors as well as CANDUs, and are normally referred to as mixed oxide fuels (MOX) (5,15,16). These cycles could use either CANDU- or LWR-generated plutonium. A variant which has some attraction as a means of facilitating the safeguards control of pure plutonium is co-processing, in which a mixture of less than 20% plutonium in recovered uranium is supplied from the reprocessing plant, either for direct use or for blending with natural or depleted uranium. The use of Pu and enriched uranium reprocessed from LWR fuel is usually referred to as a TANDEM cycle.

Finally, there is a series of thorium cycles which are of particular interest for use with the CANDU reactor, since they offer the potential for near breeding or fissile self-sufficiency. These cycles (6,17-21) make use of the U-233 generated when thorium is irradiated. U-233 is the most attractive fissile isotope for use in well-moderated reactors such as CANDU, because of its high neutron yield during fission.

All of the above cycles are typified by a higher fissile content in the fuel than is the case with natural uranium fuel, and most are intended to operate to significantly higher fuel burnups. However, these higher fissile contents and higher burnups result in greater demands on CANDU fuel than are covered by current CANDU technology. Although clearly some increase in fissile content and burnup can be accommodated using the 37-element bundle currently used in the CANDU-6, Bruce and Darlington reactors, optimum economics result from fissile loadings and burnups which are higher than could be achieved conveniently with this bundle.

It is the need for higher burnup and greater operational flexibility that has resulted in the identification of the new CANFLEX bundle as the best vehicle for using advanced fuel cycles in CANDU reactors (22,23).

In comparison to the current 37-element bundle, the CANFLEX bundle has peak fuel-element ratings that are lowered by approximately 20% (dependent upon burnup and fissile content) through the use of more fuel elements than in current CANDU bundles, coupled with the use of two diameters of fuel elements; the smaller of which are in the outer, high-flux rings of the bundle. The flexibility is further enhanced by the thermalhydraulic optimization of the CANFLEX bundle to increase its critical heat flux (CHF) capability 15% above that of current CANDU bundles. The CANFLEX bundle is being developed to be compatible with the fuel-handling equipment in current CANDU reactors, so that it can be used to support advanced fuel cycles in either new or existing CANDU reactors, but is also compatible with NU.

FUEL CYCLE DESCRIPTIONS

The various fuel cycles that are of interest for use in the CANDU reactor are depicted schematically in Figures 1 through 4.

Figure 1 depicts the once-through, natural-uranium fuel cycle, which is the basis for the design of the CANDU reactor, and has been used exclusively in these reactors up to now. The fuel-fabrication step shown in Figure 1 includes the conversion of the uranium concentrate (or yellowcake) into nuclear-grade sinterable UO2 powder, its fabrication into high-density UO2 pellets, the incorporation of these pellets into 50 cm-long Zircalooy-clad fuel elements, and the assembly of the fuel elements into CANDU fuel bundles. These bundles can have differing numbers of elements, depending on the reactor into which they are loaded, typically 28 or 37. Natural uranium fuel achieves a burnup between approximately 7 and 8 MWd/kgU, dependent on the specific CANDU reactor in which it is used.

Figure 2 shows the once-through, SEU fuel cycle. This is the cycle commonly used in most power reactors other than the CANDU type, but when used in other reactors the enrichment level is higher, and it is normally referred to as an enriched uranium cycle. This cycle involves an enrichment step prior to fabrication of the fuel assemblies or bundles. For this, the uranium concentrate is converted into the gaseous compound uranium hexafluoride (UF6) to enable enrichment in the concentration of the isotope U-235 by means of either the gaseous diffusion or centrifuge process. Following enrichment to the desired level, the UF6 is converted to ceramic-grade UO2 powder for the normal fuel-fabrication process. Typical burnups for SEU fuel could be as high as 28 MWd/kgU for an enrichment of 1.5 wt%, but the economic optimum for current component costs is 21-22 MWd/kgU with an enrichment of 1.2 wt%.

Figure 3 shows the major steps in the uranium- plutonium fuel cycle as it would be used in either the LWR or CANDU reactors. The fuel fabricated for this fuel cycle is normally referred to as mixed-oxide (or MOX) fuel because it is a mixture of uranium and plutonium oxides in the form of UO2 and PuO2. Plutonium is created in all power-reactor fuel during irradiation by neutron capture in the U-238 that it contains, and so spent power-reactor fuel can be reprocessed to chemically separate this plutonium. The fabrication of fuel for the uranium-plutonium cycle thus starts...
with the reprocessing of previously irradiated spent fuel to separate the plutonium. In the case of the co-processing variant, the reprocessing separates all of the fission products, but not all of the uranium from the plutonium, leaving at least 80% of the uranium in solution with the plutonium. This Pu or U-Pu mixture can then be blended with natural or depleted U, either in the form of liquid nitrate solutions, or in the form of oxide powders. If liquid nitrate solutions are blended, homogeneous (U,Pu)\(_2\)\(_2\) can be obtained, whereas when dry powders are blended the degree of homogeneity is dependent on the particle size of the powders involved. The resultant (U,Pu)\(_2\)\(_2\) powder is then fabricated into fuel assemblies as described previously, except that because Pu is an alpha-toxic carcinogen, all of the fabrication operations involving powders or exposed pellets must be undertaken inside sealed, sub-atmospheric glove boxes or alpha caverns. Glove boxes are normally used for small-scale production, whereas large alpha caverns are necessary to accommodate the large-scale equipment required for mass production. The potentially higher costs for Pu and alpha-active fuel fabrication tend to push the economic optimum burnup for MOX cycles to higher values than those for SEU. However, practical burnups for MOX cycles in CANDU reactors will probably be defined by experience with SEU cycles, which in most cases will precede the use of MOX cycles.

Although some of this U-233 is fissioned in the original fuel as it is irradiated, there is a considerable quantity of U-233 left in the fuel when it is discharged. Therefore, to optimize the cycle, the U-233 in the spent fuel must be recycled with fresh thorium and some Pu or U-235 "topping". This requires special facilities for reprocessing and fabricating thorium fuel when it has had the U-233 added to it, because of the high gamma fields that are associated with the U-233. These additional special facilities tend to make the reprocessing and fabrication portions of this cycle very expensive, and thus make the cycle less economic. However, the cycle does offer the opportunity for near-breeding or uranium self-sufficiency. The fuel burnups associated with thorium fuel cycles are expected to vary over a wide range. The high front-end costs of this cycle tend to drive burnups to high levels, whereas optimization for fissile resource self-sufficiency tends to favour low to intermediate burnups in the range 10-15 MWd/kg HE.

**FUEL CYCLE REQUIREMENTS**

The major incentives for the use of advanced fuel cycles in CANDU reactors are:

- to reduce uranium consumption per kW of electricity generated, and
- to reduce the cost of each kW of electricity generated.

Because of the cost of using some fuel cycles, the second incentive cannot be met with current electricity and uranium prices, although this balance varies from country to country. Therefore, the order in which fuel cycles are introduced will be dependent on their cost of use, which generally reflects technical complexity and difficulty. A significant fraction of these costs is associated with the cost of producing the fuel, and can be reduced through the use of simple, easily fabricated, fuel bundles. These costs can also be balanced to some degree by maximizing the amount of energy extracted from each fuel bundle, i.e., by increasing the burnup. Therefore, the availability of a simple, inexpensive fuel bundle design that allows greater flexibility of operation and higher burnups without increased incidence of fuel failures, can assist in making economically marginal fuel cycles viable.
TABLE 1: COMPARATIVE URANIUM SAVINGS AND BURNUP FOR CANDU FUEL CYCLES (AT EQUILIBRIUM)

<table>
<thead>
<tr>
<th>CYCLE</th>
<th>U SAVINGS (%)</th>
<th>BURNUP (RELATIVE)</th>
</tr>
</thead>
<tbody>
<tr>
<td>NU Once-Through</td>
<td>0</td>
<td>1</td>
</tr>
<tr>
<td>SEU Once-Through (1-2%)</td>
<td>30</td>
<td>3</td>
</tr>
<tr>
<td>U-Pu Recycle</td>
<td>50-60</td>
<td>4*</td>
</tr>
<tr>
<td>U-Th Recycle (Pu Topped)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- economic bias</td>
<td>70-75</td>
<td>4*</td>
</tr>
<tr>
<td>- conservation bias</td>
<td>75-100</td>
<td>1.5-2</td>
</tr>
</tbody>
</table>

As shown in Table 1, the uranium savings possible range from 30%, by the addition of slight enrichment to the present once-through cycle, to approaching 100% (self-sufficiency) within a conserving, plutonium-topped thorium cycle. The relative burnup to which the fuel bundles must be able to operate increases with uranium savings, with the exception of the most-conserving thorium cycles, where the burnup is limited by reactor physics considerations. The order in which the cycles are listed also corresponds to the order of technical difficulties, their probable cost, and thus the probable sequence for their introduction into Canada (21). It should be noted that this is the inverse of the order for uranium conservation.

FUEL PERFORMANCE REQUIREMENTS

CANDU fuel technology has a number of requirements common to most fuel cycles. These are:

1. To be able to operate to burnups significantly higher than twice current natural uranium experience, probably in the range of three to four times as high. The fuel-management procedures and higher fissile content associated with fuel for various extended-burnup fuel cycles result in these bundles having to operate at higher ratings at any given burnup above 8 MWd/kg U than would be the case for natural uranium fuel, as shown in Figure 5.

2. All fuel cycles require specialized fuel-management procedures. Some of these may subject fuel bundles to large changes in power during shuffling or the movement of absorber rods. The fuel must be able to withstand these changes, but also the bundle design must be able to accommodate the changes in heat flux without causing reactor trips. This probably requires bundles with better thermalhydraulic characteristics than the reference 37-element bundle currently used in most CANDU reactors, if derating channel powers is to be avoided.

3. As nuclear power accounts for an ever increasing fraction of the power on distribution grid systems, nuclear plants are increasingly having to accommodate daily or weekly fluctuations in load. This load following will subject individual fuel bundles to over 200 cycles of between 50 and 100% of full power.

4. There are a number of other potentially life-limiting factors associated with extended-burnup fuel for advanced fuel cycles. These include:
   - corrosion, due to the longer residence time in the reactor;
   - enhanced fission-gas release, due to chemistry effects within the fuel at high burnups; and
   - environmentally assisted sheath cracking, due to stress corrosion, hydriding and increases in local sheath stress.

It should also be noted that LWR fuels routinely operate to burnups beyond the target for advanced fuel cycles in CANDU, although at lower ratings than NU CANDU fuel. They use the same materials as CANDU fuel, and although the sheaths are free-standing, pellet-to-sheath contact ultimately occurs due to pellet expansion. Currently, LWR fuel designers are seeking to extend the core-average discharge burnup to 60 MWd/kg U. Thus, much of the generic LWR data base is CANDU-relevant, and helps confirm the viability of CANDU fuel-cycle requirements.

THE CANFLEX BUNDLE

The CANFLEX bundle (22,23) is the next logical step in the evolutionary development of CANDU fuel-bundle technology from the original 7-element bundle in NPD, through the 19, 28, 37 to the 43 elements in the CANFLEX bundle. Each of these steps has been to increase the power-producing capability of the bundle from less than 200 kW for the NPD 7-element bundle to 1250 kW for the CANFLEX bundle.

CANDU FLEXIBLE FUELING
As shown in Figure 6, the CANFLEX bundle has two sizes of element: two rings of smaller diameter elements on the outside, plus one ring, and a central pin of larger elements. The larger number of small diameter elements allow the CANFLEX bundle to operate up to 1250 kW without exceeding the 65 kW/m peak linear rating currently established as the maximum for CANDU fuel.

If the CANFLEX bundle is used at the current power limit of 1035 kW for 37-element bundles, the peak element rating can be reduced by approximately 20% dependent on enrichment level. This provides increased flexibility to operate to higher burnups, or to withstand greater power ramping than would be possible with a 37-element bundle. To further augment this greater operational flexibility, the thermalhydraulic performance of the CANFLEX bundle has been optimized to provide an 8% increase in critical channel power (CCP). The new bundle has also been designed to be compatible with the fuel handling systems in current CANDU reactors, so that the CANFLEX bundle can be both demonstrated and used in such reactors. The use of the CANFLEX bundle at current bundle powers also provides enhanced safety. For example, the 20% lower fuel rating results in the fission-gas release being substantially lowered because of lower fuel temperatures. It also lowers peak element temperatures, clad temperatures and pressure-tube temperatures during loss of coolant accidents, and lowers post-dryout clad temperatures.

APPLICATION TO ADVANCED FUEL CYCLES

The major advantage that the CANFLEX bundle provides to all fuel cycles from SEU through to thorium cycles, is the additional flexibility provided by the lower fuel ratings to operate to the higher burnups which these cycles require in order to optimize their economics. However, the additional operating flexibility provided by the enhanced thermalhydraulic capabilities of the CANFLEX bundle also increases confidence that the more-complex fuel-management procedures, often required by such cycles, can be implemented more easily. The higher CCP, and the more benign behaviour of the CANFLEX bundle under accident conditions, also eases the licensing problems associated with new fuel cycles.

FIGURE 7: RELATIVE COST SAVINGS WITH SEU

BENEFITS TO ADVANCED FUEL CYCLES

The benefits the CANFLEX bundle brings to fuel cycles in CANDU reactors are best illustrated by the SEU fuel cycle. As shown in Figure 7, the relative savings associated with using SEU peak at an enrichment of 1.2% U-235 or higher, dependent on the relative costs for natural uranium and for enrichment.

The cost of a separative work unit (SWU) is used as a measure of the cost of enrichment. An enrichment of 1.2% corresponds to a core-average burnup of approximately 21 MWd/kg U in most CANDU reactors, and the judgement is that for acceptable confidence on fuel defect rates and accident behaviour at this burnup, the CANFLEX bundle should be optimized to reduce.

Thus, to optimize the benefits from the fuel cycle, the CANFLEX bundle is needed. Similarly, the cost for the ultimate disposal of the spent fuel is reduced with greater burnup, because of the decrease in the volume of fuel that must be accommodated, as shown in Figure 8. This is particularly true if the storage time prior to disposal is as long as 50 years.

FIGURE 8: EFFECT OF BURNUP ON DISPOSAL COST

Although the CANFLEX bundle has more elements than the 37-element bundle, and thus can be expected to have a slightly higher fuel-fabrication cost, this is not a significant factor in its use in the SEU fuel cycle because the fabrication cost is a small fraction of the total fuel cost. In a mature industry, one would not expect a substantial difference in the fabrication cost of CANFLEX compared with that for a 37-element bundle. This stems from our current experience, where there is only a small differential in the fabrication costs of 37- and 28-element bundles.

For the various MOX fuel cycles, the picture with respect to the benefits provided by the CANFLEX bundle does not alter significantly. It is likely that the fuel-power changes and flux ripple during fuel shuffling will be higher than with SEU, and thus provide an even greater incentive to use the CANFLEX bundle. Also, the significantly higher cost of the plutonium cycles in some areas, such as Canada, will drive the most economical operation toward very high burnups (6,8,21). However, the more complex and expensive fuel fabrication necessary for alpha-toxic fuel may accentuate the effect of the small fabrication cost difference between the CANFLEX and 37-element bundles, but that is expected to be insignificant in the overall cost of the cycle.

The same advantages also apply to the use of the CANFLEX for the various variants of the thorium cycle,
because the high costs of that cycle will drive its economics toward high burnups. However, if the fuel-cycle cost structure in the future is heavily influenced by the need to conserve uranium, then the lower-burnup self-sufficient thorium cycles could be economical placing a lower incentive for high-burnup behaviour.

INTRODUCTION STRATEGY

Although all of the testing and demonstration irradiations of the CANFLEX bundle undertaken during its development program have been, and will continue to be, with SEU fuel, the first commercial demonstration in a power reactor is likely to use natural uranium for simplicity. However, once this has been achieved, clearly the first fuel-cycle application will be with SEU or its RU variant, because of the strong economic pressures, and relative simplicity compared to the other fuel cycles.

Following this, the CANFLEX bundle could be used with MOX fuel cycles in those areas where the economic or other strategic factors make it attractive, and where the appropriate technological infrastructure exists for such a cycle. The supply and application of the CANFLEX technology and experience to that infrastructure would be a relatively simple matter.

Since it is clear that the economical application of the thorium cycle would not occur before the MOX cycles, the experience gained in those cycles could be applied to the eventual use of the CANFLEX bundle in the thorium cycles.

CONCLUSION

Although natural uranium is currently used in all CANDU reactors, slight enrichment has already become significantly more economical, and will establish the operating technologies necessary for the use of other cycles which are expected to become economical early in the next century. The CANFLEX bundle will move that date forward.

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ABSTRACT

Reactor physics data and calculational methods can be checked against integral measurements performed in "zero-energy" critical facilities such as the ZED-2 reactor at CRNL. This has been done for several advanced fuels --{(Pu,U)O₂, (Pu,Th)O₂ and (U²³³,Th)O₂}-- that are of interest in proposed fuel cycles in the CANDU reactor. Since insufficient quantities of these fuels were available to assemble critical cores containing only the fuel of interest, methods of measurement and analysis for mixed fuel cores had to be devised. The measurements reported here were made with the (Pu,U)O₂ fuel and include both reactivity worths and reaction-rate distributions. Comparisons with calculations using the cell code WIMS-AECL and the reactor-core code CONIFERS are generally satisfactory.

INTRODUCTION

The CANDU reactor is naturally well suited to development for use with advanced fuel cycles because "neutron economy" was a major criterion in the development of the current commercial version. Neutron economy means that the number of neutrons captured in the uranium fuel is maximized while those lost to parasitic absorption in structural materials, moderator, coolant, various absorbers and shields, etc., are correspondingly minimized. This results in a reactor which achieves the highest uranium utilization of all current commercial reactor types by burning natural uranium oxide fuel in a once-through fuel cycle characterized by low fissile inventory and high conversion ratio. The use of pressure tubes and the consequent heterogeneous nature of the reactor lattice also allows easy use of fuels other than natural uranium, including slightly enriched uranium (SEU), plutonium-uranium (Pu-U) and thorium fuels.¹

There is then, a wide range of fuel cycles that can be implemented in CANDU. These include cycles that are symbiotic with those of light-water reactors (LWRs); for instance, a possible source of SEU is the recovered uranium (RU) obtained when LWR fuel is reprocessed. This material typically has a U²³⁵ content of about 0.9%.

While the major advantages that these advanced fuel cycles maintain for CANDU are those of superior resource utilization and superior economics, enriched fuel cycles also allow the possibility of further reductions in capital costs (e.g., reduced heavy-water inventory, increased average channel power). However, these reductions would be at the expense of the resource utilization.

An important part of the capability needed to assess the economics and resource utilization of advanced fuel cycles, and to design and operate the reactors required for their exploitation, is the ability to properly perform the necessary reactor physics calculations. This ability is based on having available the appropriate computer codes and data bases of differential neutron-interaction-cross-sections validated for the necessary range of materials, geometries and physical conditions. In general, the calculations become more difficult for enriched fuels because of the increased blackness to neutrons of the fuel and because of the greater change in fuel composition during the irradiation.

For the CANDU reactor an important part of the validation of the codes and data is the comparison of their predictions with careful measurements performed in the ZED-2 critical facility at Chalk River Nuclear Laboratories. During the past several years three fuel types, relevant to the exploitation of advanced fuels in the CANDU reactor, have been studied in ZED-2. These were: (Pu,U)O₂ containing 0.45% fissile Pu in U depleted to 0.34% U²³⁵, (Pu,Th)O₂ containing 1.8% fissile Pu in heavy elements, and (U²³³,Th)O₂ containing 1.4% U²³³ in heavy elements.

¹CANadian Deuterium Uranium.

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EXPERIMENTS ON (Pu,U)O₂ FUEL

General Description of ZED-2

The ZED-2 research reactor is a tank-type critical facility with a cylindrical aluminum tank 3.3 m in diameter by 3.3 m deep. Fuel assemblies are hung from stainless-steel beams located above the tank. The reactor is moderated by heavy water (inventory about 27 tonnes) and is controlled by varying the level of the heavy water in the tank. The depth of heavy water in the tank is measured when required by a probe lowered from above until it just makes electrical contact with the water. The reproducibility of measurements with this probe is better than ±0.1 mm. Changes in reactivity in ZED-2 are measured as changes in critical height of the heavy water.

The maximum operating power of the reactor is 200 W nominal, which corresponds to neutron fluxes of about \(10^5\) cm\(^{-2}\) s\(^{-1}\). This is adequate to activate neutron measuring foils that can be placed in the core, but also allows ready accessibility to the core for modifications.

The Fuel and Lattices Used

The (Pu,U)O₂ fuel consisted of mixed dioxides of Pu and U, depleted to 0.34% of U\(^{235}\), such that the fissile Pu content was 0.45 wt% of the heavy elements. Fuel elements were assembled into 36-element bundles having the cross-section shown in Fig. 1. Twenty-two such bundles were available, and for the experiments in ZED-2, they were assembled on central-support-tubes into five rods, one of which contained five bundles of the (Pu,U)O₂, the other four each being topped by a filler bundle of almost identical geometry, but containing natural uranium. Because the critical heights of the ZED-2 cores used were low enough, the natural uranium bundles were never in the core and so have negligible effect on the measurements.

These fuel rods were placed in the ZED-2 hot sites, which are designed to enable the fuel and coolant contained in them to be heated to 300°C. The hot sites are illustrated in Fig. 2 which shows that they consist of a pressure tube surrounded by a concentric calandria tube, both tubes being capped at the bottom with hemispherical caps. At the top a breech block provides a pressure seal so that in operation a helium gas over-pressure can be maintained to suppress boiling. Heat is provided by a 2 kW electric heater coil at the bottom of the central-support-tube, and to minimise heat losses, the gap between the pressure and calandria tubes is evacuated.

Five rods of the (Pu,U)O₂ fuel were not enough to achieve a critical core, therefore experiments were performed in which the (Pu,U)O₂ rods were substituted into reference cores containing natural uranium fuel. Two reference cores were used, both having a hexagonal array of fuel rods, one with a pitch of 31 cm and the other with 24.5 cm. A plan view of the 31 cm pitch core is shown in Fig. 3, it consisted of 55 rods containing five bundles each of 28-element fuel in an...
aluminum mock-up of a pressure/calandria-tube assembly. The 24.5 cm pitch core was similar except that to provide sufficient reactivity it was necessary to have an extra ring of 30 natural-uranium-metal "Zeep" rods. Both the 28-element and the Zeep rod fuel have previously been described.\(^3\),\(^4\).

**Substitution Experiments**

In these measurements the reactivity effect, in terms of critical height of the moderator, was measured as 1, then 3, and finally all 5 rods of the (Pu,U)\(_2\O_2\) fuel were substituted into the reference core. The three arrangements of the substituted rods are shown in Fig. 4.

To aid in the interpretation of the critical height measurements, the relative neutron density distribution was measured throughout the core by activating copper foils. In particular, foils were placed at the cell boundary between fuel rods across the core at the elevation of the maximum flux and vertically on either side of the central rod.

Only measurements made in the lattice of 31 cm pitch are reported here. The (Pu,U)\(_2\O_2\) was "cooled" by both heavy water and air as this allows a check to be made on the ability of computer simulations to calculate the reactivity associated with voiding the coolant.

**Measurement of Detailed Reaction-Rate Distributions**

These measurements were made by activating thin foils containing various materials of interest that were placed in and around a special demountable (Pu,U)\(_2\O_2\) fuel bundle. To simplify analysis of the results it is desirable that the neutron spectrum and distribution at the position of the bundle be as close as possible to that which would obtain in an infinite lattice of the fuel. For this reason, 21 of the bundles available were redistributed amongst seven hot sites so that all the nearest neighbours of the demountable bundle, both radially in adjacent channels and above and below, were also (Pu,U)\(_2\O_2\) bundles. The remaining positions in the hot sites, above and below the (Pu,U)\(_2\O_2\) bundles, were filled with natural uranium bundles of the same geometry.

The demountable bundle contained six special elements into which the foils could be loaded. The location of the six special elements (denoted A, B, C, D, E, and F) is shown in Fig. 1. Because of the radiological toxicity of the Pu in the fuel, loading and unloading were performed in a special laboratory and the sheaths were welded closed for the irradiation.

Circular foils of the following materials were placed between the pellets of the demountable elements: Pu-Al alloy, natural U, Lu-Mn-Al alloy, Au-Al alloy, Cu, and critical at a constant power and the critical height of the moderator was measured when the contents of the hot sites had equilibrated at various temperatures.

Since only five rods are heated out of a core of at least 55, the reactivity effect is small (less than 1 milli-k), corresponding typically to a change in critical height of only about 1 cm. Consequently, great care has to be taken to correct for other effects that affect the reactivity of the reactor over the approximately 8 hours it takes to raise the temperature to 300°C. The most important of these is heating of the heavy-water moderator. The temperature of the moderator, as a function of depth, was measured with a thermistor located on one side of the reactor, typically the temperature changed by less than 0.5°C during the experiment, which corresponds to a height correction of about 0.06 cm.

Measurements were made at both pitches, but only with heavy-water coolant, since the equipment does not allow a gas "coolant" to be heated.
Al as shown in Fig. 5 (note that the Cu and Au-Al foils were in the same gap). Each foil was wrapped in at least one layer of 0.03 mm thick Al-foil to protect the foil material from contamination by Pu or fission products from the fuel. This wrapper was discarded when the foils were removed from the elements for counting after the irradiation. The foils containing fissile and fertile material were permanently wrapped in another layer of the same Al-foil to prevent loss of the foil material.

In addition to the foils in the fuel elements, Cu, Au-Al alloy and Lu-Mn-Al alloy foils were attached to the outer surface of the calandria tube and to a stringer located on the cell boundary between the central fuel rod and those immediately next to it. Copper foils were also mounted on the stringers at intervals of 10 cm to measure the axial thermal neutron flux shape at those locations.

Examples of all types of foils were also irradiated on a rotating aluminum wheel placed in the heavy water near the ZED-2 vessel wall. At this location, far from the nearest fuel rod, the neutrons are well thermalized to the physical temperature of the moderator and only a small component of epithermal neutrons is present. This component was found by measuring the cadmium ratio for a resonance absorber, such as Au-197, and interpreting this in terms of the Westcott epithermal index, \( r \). The average value of \( r \), (approximately the fraction of epithermal neutrons) obtained from all these measurements, was \((10 \pm 3) \times 10^{-5}\).

Froils irradiated in this reference flux provide activities corresponding to a simple, well-understood neutron spectrum for which activation cross-sections are well known.

A typical irradiation was for 1.75 h at a nominal power of 150 W, followed by a cooling period of about 10 hours, after which the foils were removed and prepared for counting. The gamma-ray activity of the foils was counted with pairs of NaI detectors of various sizes coupled to amplifiers, discriminators and scalers. An automatic sample changer positioned the foils between the crystals and a computer-controlled counting system recorded the following: the number of counts in each channel above a fixed discriminator level, the time of day and the elapsed time of each count. The foils were mounted in Lucite trays before being stacked in the sample changer. The stack, which included some empty trays for background measurements, was counted many times. Data analysis consisted of: correction for counter dead time and activity decay; subtraction of room background, rejection of counts more than three standard deviations from the mean for a foil, and normalization by foil mass or sensitivity.

Measurements were made at both lattice pitches with the heavy-water coolant at ambient temperature (typically 21°C) and at 300°C; also, measurements were made with air in the coolant space at ambient temperature.

**COMPUTER CODES USED FOR SIMULATION OF THE MEASUREMENTS**

**General Description of the Codes Used**

The cell or lattice code WIMS-AECL (formally known as WIMS-CRNL(5)) was used to model individual cells of the reactor. WIMS (Winfrith Improved Multigroup Scheme) is a multigroup transport code originally created at the United Kingdom Atomic Energy Establishment, Winfrith, that has been widely distributed, existing in various versions and in many countries. WIMS-AECL solves the neutron transport equation in one or two dimensions and many energy groups, and provides as output few-group homogenized cross-sections suitable for use in reactor models based on diffusion theory.

The code used to model the reactor was CONIFERS(6). CONIFERS is a neutronics code for nuclear reactors whose fuel is in channels that are separated from each other by several neutron mean-free-path lengths of moderator. It can treat accurately situations in which the usual homogenized-cell diffusion equation becomes inaccurate, but is more economical than other advanced methods such as response-matrix and source-sink formalisms. CONIFERS uses exact solutions of the neutron diffusion equation within each cell. It allows for the breakdown of this equation near a channel by means of
data that almost any cell code can supply. It uses the results of these cell analyses in a reactor equations set that is as readily solvable as the familiar finite-difference set. CONIFERS can model almost any configuration of channels and other structures in two or three dimensions and with any number of energy groups.

**Calculations for Comparison with Reaction-Rate Measurements**

WIMS-AECL alone was used to calculate reaction rates for comparison with the measured values. The nuclear data library used had 89 energy groups and was generated from the ENDF/B-V(7) nuclear data file. A two-dimensional calculation (PIJ), in 18 energy groups, was used for the fuelled cell. To obtain thermal reference spectrum reaction rates, a homogeneous calculation was performed with the Westcott spectrum option using the measured epithermal index and the moderator temperature. This calculation was done using the full 89 energy groups available in the data library.

One of the main questions to be answered on the basis of the comparison of calculated and measured reaction rates is whether or not WIMS-AECL calculates the correct neutron spectrum in the cell. The basic WIMS-AECL calculation assumes that the cell of interest is surrounded by an extensive lattice of identical cells; this was not the case in the experimental setup. To see if this was likely to be a problem some special WIMS-AECL calculations, in which the true environment was simulated, were made. These calculations confirmed that the neutron spectrum at the measurement site was very similar to that in an infinite lattice of the test fuel.

**Calculations for Comparison with Substitution and Fuel/Coolant Heating Measurements**

In these instances it is necessary to model the reactor using CONIFERS with homogenized few-group cross-sections and other data derived from WIMS-AECL. An automated system of codes, illustrated in Fig. 6, is used to do this. The other codes in this package serve to massage and pass the required data from the user to WIMS-AECL and CONIFERS and back again to the user. The main transport calculation in WIMS-AECL was again run with 18 energy groups, but a one-dimensional solution (DSN) was used. WIMS-AECL was run for each fuelled region of the reactor and the code WINCON (Fig. 6) condensed the WIMS output data into four energy group cross-sections for CONIFERS. The CONIFERS model was three-dimensional; a horizontal slice is illustrated in Fig. 7, which shows that the radial boundary of the model was at the inside edge of the ZED-2 calandria tank. Axially the model included all materials between the moderator upper surface and the bottom of the calandria. The CONIFERS model has the moderator level set at that measured in the experiment and iterates on the value of k-eff which, if the model were perfect, would be calculated to be unity. To allow the calculated deviations of k-eff from unity to be converted to corrections to the measured height, level coefficients of reactivity (LCRs) were also produced by CONIFERS. In addition, CONIFERS calculates cell boundary fluxes which are converted to Cu-64 activities for comparison with measured values.

**Figure 6. The Automated System of Computer Codes Used for The WIMS-AECL/CONIFERS Calculations**

**Figure 7. Plan View of the Mesh Used in the CONIFERS Calculation. Two-Fold Reflectional Symmetry was Used. The Hexagonal Cells Contain Fuel, the Rest Heavy Water**

**COMPARISON OF MEASUREMENT AND CALCULATION**

**Detailed Reaction Rates**

Comparison of measured and calculated reaction rates provides a check that the measured and calculated neutron spectra are in agreement. To compare spectra, ratios of activities are formed, in particular the
ratio of an activity arising from a reaction having a prominent resonance with that of one from a reaction having a one-upon-v cross-section. The problem of normalization between measured and calculated ratios is handled by measuring the ratio of the same activities on the reference wheel where the spectrum is well defined. Thus one actually compares measured and calculated ratios of ratios such as:

\[
\frac{(A_1/A_2)_R}{(A_1/A_2)_C}
\]

where the subscripts 1 and 2 refer to the two activity types and the subscripts C and R refer to locations in the cell and in the reference spectrum, respectively. The use of this kind of "relative to thermal" ratio has the great advantage that many hard-to-measure parameters, such as counter efficiency for different activities and absolute quantities of materials in foils, cancel.

Of the reactions measured, neutron capture by Cu\(^{63}\) and Mn\(^{54}\) have one-upon-v cross-sections, at least in the thermal region, and only small resonance integrals. \(\text{U}^{235}\) fission is also close to one-upon-v, but \(\text{Pu}^{239}\) fission and \(\text{Lu}^{176}\) capture both have prominent resonances in the thermal region at 0.296 eV and 0.141 eV, respectively. The latter two reactions are therefore sensitive to the shape, or temperature, of the thermal neutron spectrum. The capture cross-section of \(\text{U}^{238}\) and \(\text{Au}^{197}\) both have prominent resonances in the epithermal region at 6.67 eV and 4.91 eV, respectively, so that the measured activity depends on the ratio of epithermal to thermal neutron flux.

In Figs. 8 to 15 the percentage differences between the experimental (E) and calculated (C) reaction-rate ratios \(\frac{(C/E-1)\times100}{1}\) are given for the following average activity ratios: (Pu\(^{239}\)/U-nat fission), (U\(^{238}\) capture/U-nat fission), (Lu\(^{176}\)/Mn\(^{55}\) capture), and (Au\(^{197}\)/Cu\(^{63}\) capture), for each of the experimental conditions.

Before discussing the quality of the agreement between the measurements and calculations, a few comments should be made about the uncertainties associated with the measurements. Two types of uncertainty, random and systematic, exist. The random errors associated with the activity measurements can be estimated approximately by examination of the results because the activity at a given location, in the cell or on the reference wheel, was always measured with two foils. Thus the difference between the two results gives an estimate of the random error in the measurement. Averaging over all the irradiations and all positions shows that for all the activities measured the random error was about ±0.3%. The largest difference between two measured activities that were nominally the same corresponded to an approximate random error of ±1.0%.

The estimated random error contains contributions from many sources: the reproducibility of counting (typically ±0.1%), variation in positioning of foils during counting and in the reactor for irradiation, small differences in thickness between foils, errors in measured masses or sensitivities of foils, etc.

An important systematic error to consider is the perturbation of the neutron field by the presence of the measuring foil. This will occur to some extent whenever the neutron interaction cross-sections of the foil material are different from those of the surrounding medium (which is usually the case). As far as possible, foil materials and dimensions are chosen to minimize such systematic errors. In the comparisons reported here no corrections have been made for these perturbations. This may not cause problems in many cases because, in the ratios formed, the errors tend to cancel. A remaining cause for concern is, however, self-shielding in foil materials, where a large fraction of the activation is due to resonance capture and the calculation does not account for this explicitly, for instance, \(\text{Au}^{197}\) activation.

In Figs. 8 and 9 Pu/U fission ratios are compared and the agreement is good, especially for the 31 cm pitch case. This indicates, in particular, that the change in Pu fission rate with neutron temperature is being properly calculated.

The agreement is not quite so good for the U\(^{238}\) capture to U-nat fission ratios in Figs 10 and 11, especially at the 24.5 cm pitch. The discrepancies may be due to problems with the calculation, or to systematic errors arising from the perturbation of the neutron field by the aluminum-wrapped uranium-metal-foil placed between the mixed oxide pellets of the fuel.
For the Lu$^{176}$/Mn$^{54}$ capture ratios the agreement is not too bad at 31 cm pitch, but is less good at 24.5 cm. There is also a marked tendency for the calculation to underestimate the change in ratio with increasing temperature. Since in the ratios involving Pu fission this was not observed, it must be supposed that the code calculates the thermal neutron spectrum correctly and that therefore there may be a problem with the Lu$^{176}$ cross-sections in the data base. The observed trends with pitch and penetration into the bundle may be explained by lack of correction for resonance self-shielding in the Mn activation.

The poor agreement observed in the comparison of Au$^{197}$/Cu$^{63}$ capture ratios shown in Figs. 14 and 15 is believed to be due largely to lack of correction for self-shielding in resonance activation, particularly of gold.

Substitution Measurements

The comparison of measurement with the WIMS-AECL/CONIFERS calculations is summarized in Fig. 16, where the CONIFERS k-eff is plotted against the number of (Pu,U)O$_2$ fuel rods substituted, with both heavy water and air "coolant". A perfect simulation would correspond to every k-eff being unity; in fact, the actual values lie between +0.2 and +2.4 milli-k of that value. The uncertainty in the measured critical height is estimated to be ±0.3 mm, which corresponds to ±0.022 milli-k. Agreement is satisfactory, although there is a divergence between the heavy water and void curves as the number of rods substituted increases, which may indicate a problem with calculation of the void reactivity.

A further comparison of calculation and measurement is illustrated in Figs. 17 and 18, where the ratio of calculated to measured copper activity is plotted against axial and radial position in the reactor.

These plots are for one rod of (Pu,U)O$_2$ substituted and, in general, agreement is good with most points lying in the range ±1%.

Reactivity Effect of Heating Fuel and Coolant

The measured variation of critical height with fuel/coolant temperature of five rods of (Pu,U)O$_2$ fuel in the 31 cm pitch core is shown in Fig. 19. The data have been corrected for variation of the moderator temperature during the measurement period using a
measured coefficient. The estimated uncertainty of the measured heights is ±0.3 mm or ±4.8% of the maximum height change observed. The WIMS-AECL/CONIFERS calculation, for which differences in k-eff have been converted to height differences using calculated LCR’s, is also shown in Fig. 19. Clearly, there is a significant difference between measurement and calculation which amounts to a maximum of 9.5 mm in height or 0.72 milli-k in reactivity. This is more significant than it may at first appear since it must be remembered that only 5 out of 55 rods were heated.

CONCLUSIONS

With regard to calculation of reaction rates, in particular those occurring in the materials of the fuel studied, such as fission of Pu and U and capture in U²³⁸, the cell code WIMS-AECL gives satisfactory agreement with measured values. For the spectral indicating ratios Lu¹⁷⁶/Mn⁵⁴ capture and Au¹⁹⁷/Cu⁶³ capture, the agreement between calculation and measurement is less good, but could probably be improved by applying corrections for self-shielding for resonance capture in the foils.

For the analysis of substitution measurements of 1, 3 and 5 rods of the (Pu,U)O₂ fuel, the calculational scheme of WIMS-AECL combined with the reactor-core code CONIFERS shows promise of providing an accurate analysis. In future, the sophistication of the analysis will be increased to emphasize the accuracy of prediction of quantities of interest such as the void reactivity.

The reactivity effect of heating the heavy-water coolant and (Pu,U)O₂ fuel from 20 to 300°C is not well modelled by WIMS-AECL/CONIFERS although a similar measurement with natural UO₂ is. This suggests that WIMS-AECL has problems with the calculation of temperature effects for fuels containing plutonium.

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D.C. McElroy, who made the system of computer codes work and produced the calculational results reported here.

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ABSTRACT

Fuel-management simulations have been performed for 1.2% SEU fuel in a CANDU 6 reactor at equilibrium, for three fuel-management options:

1) axial shuffling,
2) a regular 2-bundle shift with the adjuster rods removed from the core,
3) a regular 2-bundle shift with the adjuster rods present.

Both time-average and time-dependent simulations were performed, from which the physics characteristics of the cores at equilibrium were estimated. Power and power-boost envelopes were derived for both 37-element fuel, and the advanced CANFLEX bundle.

INTRODUCTION

Background

The use of slightly enriched uranium (SEU) fuel in CANDU offers many benefits. At the optimal enrichment of about 1.2%, fuel-cycle costs are reduced by about 25% compared to natural uranium fuel; the amount of energy derived from the mined uranium increases by 30%; burnup increases by a factor of three over natural uranium, resulting in a three-fold reduction in the amount of spent fuel produced per unit of energy; and in new reactors, enrichment offers greater flexibility in design.

The use of enrichment in CANDU does pose some challenges. The achievement of the same outstanding level of fuel performance at extended burnups as at natural uranium burnups is more demanding, especially if power maneuvering is required. Fuel-management strategies are key to ensuring acceptable fuel performance, as well as in maintaining bundle and channel powers within acceptable limits. Enrichment also affects reactor control and the response of the reactor and the fuel in postulated accident scenarios.

In order to meet these challenges with the highest degree of assurance, Atomic Energy of Canada Limited has developed an advanced fuel bundle, called CANFLEX (CANDU flexible fuelling). This advanced bundle continues the trend of increased subdivision in successive CANDU bundles, featuring 43 elements arranged in four rings, with two pin-sizes (the pins in the inner two rings are larger than in the outer two rings). The increase in the number of pins and the grading of pin sizes results in a 20% reduction in peak linear element ratings over the 37-element bundle (for a given bundle power). This reduction in ratings, along with changes to the pellet design, facilitate the achievement of higher burnups. The lower ratings are also in line with world-wide trends towards greater safety margins, and result in lower fuel temperatures during postulated large-break LOCAs. Additionally, boundary-layer flow-disturbers increase the critical channel power (CCP) by about 8%, providing increased operating and safety margins. The bundle will be compatible with fuel-handling equipment in existing and future reactors. Hence, the CANFLEX bundle is the optimal vehicle for introducing enrichment in CANDU.

Fuel Management with Enriched Fuels

Enriched fuels necessitate different fuel-management strategies than for natural uranium fuel, because the initial reactivity is greater, and there is a greater variation of reactivity with burnup. The central location of the adjuster rods also complicates fuel-management with enriched fuels, because the flux and power tend to be depressed near the adjusters. Several fuel-management options have been identified for 1.2% SEU fuel, including the checkerboard fuel-management scheme, axial shuffling, a regular 2-bundle shift, and two-pass refuelling. In new reactors, reactivity devices, and in particular, adjuster rods, can be relocated to facilitate the use of a variety of fuel types.

This paper looks in detail at three fuel-management options for 1.2% SEU in a CANDU 6 reactor:

1) axial shuffling,
2) a regular 2-bundle shift with the adjuster rods removed from the core,
3) a regular 2-bundle shift with the adjuster rods present.

Axial Shuffling. This fuel-management option provides the greatest flexibility in shaping the axial power distribution. Axial shuffling involves removing some or all of the bundles from the channel, rearranging the bundles in pairs, and reinserting some of the fuel-bundle pairs back into the channel in a different order, along with fresh fuel. Axial shuffling is the reference fuel-management strategy for the CANDU 3(9), since all fuelling is done from one end of the reactor. Preliminary studies indicate that axial shuffling is feasible in other CANDU reactors as well, although in some cases hardware changes may be needed.

The ease of application, and the number of bundles which can be removed from the channel, depend on several factors. For instance, the fuelling machine magazine capacity must be sufficient to hold all of the bundles discharged from the channel, the fresh bundles which are to be inserted into the channel, and other components, such as the channel closure and shield plug. The coolant flow must be sufficient to discharge all of the bundles.

The particular axial shuffling scheme chosen for this study was the best of several studied. The shuffling scheme is bi-directional, although in a reactor in which the fuelling is from one face of the reactor, such as in the CANDU 3, bi-directional fuelling can be "mirrored". If bundles are numbered from 1 to 12 starting at the "refuelling" end of the channel, then bundles are first loaded into positions 1 and 2. Upon subsequent fuelling operations, the bundles in positions 1 and 2 are shifted as follows:

1 - 5 - 3 - 9 - 7 - 11 - discharged
2 - 6 - 4 - 10 - 8 - 12 - discharged

The axial shuffling scheme was restricted to the central channels, in the vicinity of the adjuster rods. In the peripheral channels, a regular 2-bundle shift fuelling scheme was employed (Figure 1).
**Reactor Model**

The calculations were done for a CANDU 6 Mk 1 reactor, having 380 fuel channels, using the fuel-management code FMDP(11). To some extent, the results will be applicable to other reactor types. Figure 2 shows a top view of the reactor.

Cell-averaged cross sections for the FMDP model were calculated using the cell-code WIMS-AECL(15) for the CANFLEX bundle with 1.2% SEU. The transport calculation in WIMS was done in 20 energy groups, which were subsequently condensed to two energy groups using a critical spectrum. The burnup calculations were performed at the core-average fuel ratings, with critical bucklings.

Incremental cross sections for the reactivity devices were not specifically calculated for CANFLEX fuel: for the adjuster rods and zone controllers, values calculated in an earlier study for 1.2% MOX (mixed uranium/plutonium oxide) fuel were used; for the structural material, values appropriate for natural uranium were used. In general, the incremental cross sections have been found to be rather insensitive to fuel type.

**Time-average Calculations**

In the time-average calculation, properties of the lattice cell were averaged over the dwell-time of the fuel, at each position in the core. The resultant flux and powers are indicative of what would be seen "on average". In reality there would be perturbations about the time-average distributions, due to refuelling, control rod action and so on. The time-average channel power distribution serves as the reference power distribution for actual refuelling.

In setting up the time-average model, the core is divided into irradiation zones, over which the average fuel discharge irradiation is constant. These irradiation zones are chosen to make the reactor critical (or to maintain some excess reactivity to account for parasitic absorption in the core which is not explicitly modelled). They also enhance the flattening of the radial channel power distribution provided by the adjuster rods (if they are present).

The water level in the zone control compartments was set to 45% full, representative of the normal operating conditions.

**METHODOLOGY**

**Outline of Method**

For each of the three equilibrium fuel-management strategies, time-average calculations were performed to determine the average core properties such as burnup and refuelling rate, and the reference bundle and channel power distributions. The time-average calculation does not take into account refuelling ripples. To do this, an "instantaneous" calculation was first performed, which is meant to approximate a "snapshot" of the burnup and power distributions in the core at an arbitrary point in time. Then a time-dependent fuel-management simulation was carried out for each fuelling strategy, using the instantaneous calculation as the starting point. The refuelling simulations were carried out for 100 days, and from the simulations estimates of reactor core characteristics such as maximum powers were made, and the power and power-boost envelopes were derived.

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**FIGURE 1: FACE VIEW OF REACTOR**

Regular 2-bundle shift without adjusters. A regular 2-bundle shift fuelling scheme is near-optimal with 1.2% SEU, either in a core without adjuster rods, or in the peripheral channels of a core with adjuster rods. The axial power distribution peaks at the refuelling end of the channel, and decreases along the length of the channel. The axial form factor is good (i.e., the peak bundle power is relatively low for a given channel power). The margin to dryout for the equilibrium power distribution is increased if fuelling is in the direction of coolant flow, since the peak power then occurs at the inlet end of the channel, where the coolant enthalpy is lowest. Although these calculations are done for a CANDU 6 with the adjuster rods removed, it is expected that the general physics characteristics would apply also to Bruce A, in which there are no adjuster rods.

Regular 2-bundle shift with adjusters. When 1.2% SEU is employed with a regular 2-bundle shift fuel-management scheme in current reactors, the adjuster rods depress the power in the center of the channel. This results in a lower axial form factor (i.e., a higher peak bundle power for a given channel power), and also in some power boosting at extended burnups.

**FIGURE 2: TOP VIEW OF REACTOR**
**Instantaneous Calculations**

The instantaneous calculation is meant to be a "snapshot" of the core power and burnup distributions at an arbitrary point in time. Every channel in the core is assigned an "age" -- a number between 0 and 1 which indicates the fraction of dwell-time between visits of the fueling machine to the channel. A channel with an age of 0 has just been refueled; a channel with an age of 1.0 has reached its target burnup and is about to be refueled. The instantaneous irradiation of a bundle can be determined from its time-average irradiation, the fuelling scheme, and the channel age. In the instantaneous calculation, the channel ages range uniformly from 0 to 1, with an increment of 1/380 (380 channels in the core). In the past, a series of instantaneous calculations with a semi-random distribution of ages has sometimes been used to estimate core properties such as actual peak powers and refuelling ripples. However, any clustering of channel ages near 0 or 1 can result in large global tilts in flux and power, with largescale channel and bundle powers in one region of the core. With SEU fuel, the tilts in power resulting from a completely random age distribution are usually very large, and unrealistic. In this study, rather than "correct" for these global tilts in some rather arbitrary way, channel ages were assigned in such a way as to minimize global tilting, on the grounds that these tilts would be avoided in reality through careful selection of the channels for refuelling, and through the spatial control function of the zone controllers. The resultant instantaneous calculation was then used only as the starting point of the short time-dependent fueling simulation.

The following procedure was followed to arrive at a distribution of channel ages for the instantaneous calculation. The core was divided into 5x5 arrays of channels (Figure 3). In each 5x5 block, the channels were ordered, from 0 to 25 (shown in the lower left of Figure 3). This order was identical in each block. (Note that the peripheral channels did not occupy full blocks.) Next, each block was ordered (the large numbers shown in outline in Figure 3). Each channel was then assigned a number, starting from 0 and ending with 379, according to the order within the 5x5 blocks, and the order of the blocks. For instance, blocks 1 and 2 do not contain channel #1 (within the blocks). Hence, channel D-7 (channel #1 in block 3) was assigned the number 0; channel T-12 was assigned the number 1 (channel #1, block 4); channel C-17 was assigned the number 2 (channel #1, block 5); and so on until channel J-7, which was assigned the number 15 (channel #1, block 24); then the ordering continued, starting with channel #2 in block 1 (or the first block containing channel #2). Hence, channel D-10 was assigned the number 16 (channel #2 in block 3), and so on until all the channels in the core were assigned a number, ending with channel G-9 being assigned the number 379 (channel #25 in block 24). The channel age for each channel was then simply the number assigned to the channel, divided by the number of channels (380).

The resultant age map is shown in Figure 3. It is seen that the same location in each block has a similar age. From this age map, the irradiation and burnup of each bundle in the core was determined.

This method has several advantages. First, clustering of channels is avoided, at any age. The standard method of choosing a random distribution of channel ages, then removing clusters of channels with low age can still result in clusters of channels at higher ages. Although these channels may not produce a global tilt in power initially, if that distribution of channel ages is allowed to burn, and is refueled, then at a later time, the cluster of channels having similar ages will need to be refueled close together in time, and at that time a global tilt in power may result. Hence, if the instantaneous distribution is to be used as the starting point of a time-dependent fuel-management simulation, it is necessary to avoid clustering of channel ages about any age, not just low age. This method avoids this clustering. One can view this method as an attempt to define the order in which channels are to be refueled. In practice, one attempts to spread out the refuelling of channels in a similar way, so as to avoid power peaking. Hence, this method aims for the smallest deviation from the time-average distribution, just as the refuelling engineer attempts.

The ripple, or deviation from time-average, resulting from this method is in fact smaller than what could be achieved in reality, and the time-dependent simulation is an effort at estimating more realistically the actual refuelling ripples and peak powers.

It has to be kept in mind that the results of the time-dependent refuelling simulation are influenced by the starting point, especially for a relatively short simulation such as this. In order to "forget" the starting point, the simulation would have to be continued for a longer period of time. Also, the burnup resulting from a short simulation starting from an instantaneous calculation will be somewhat low, since the exit burnups in the instantaneous calculation at the start of the simulation do not exceed the average exit burnup, while in reality there would be a distribution of burnups extending beyond the average exit burnup.

<table>
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**Figure 3: Age Map**

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Time-dependent Refuelling Simulation

In order to estimate core characteristics such as peak powers and refuelling ripple, a time-dependent refuelling simulation was performed for 100 full-power days for each of the fueling schemes, using the SIMULATE module in FMDP. Individual channels were selected for refuelling, and the FMDP powers were calculated at 10-day intervals. The same instantaneous calculation was used as the starting-point in each of the simulations. By the end of the 100 day period, a total of 582 fuel bundles had been discharged from the core, corresponding to the refuelling of 291 channels. In selecting channels for refuelling, the following guidelines were followed:

1) The channel should have reached or exceeded its target burnup (more precisely, those bundles which would be discharged during refuelling should have reached their target burnup). Note that the channels with the highest age also tend to provide the highest reactivity increase on refuelling.

2) Candidate channels and their neighbours should not have high power or overpower before refuelling. This criterion minimizes power peaking resulting from refuelling, and generally complements the first one, since channel powers generally decrease with age.

3) Clustering of highly reactive channels or high power channels must be avoided. This guideline reduces global and local power peaking.

4) Fueling should occur in symmetric parts of the core (both radially and axially). This criterion again aims at avoiding global tilts in the power distribution. Because of the asymmetric axial power profile with SEU fuel, it is much more important to maintain a balance in the number of channels refuelled in each direction than it is with natural uranium fuel.

5) The number of channels refuelled is determined by the need to maintain criticality.

The refuelling simulation was performed first for the axial shuffling scheme. Exactly the same channels were then refuelled for the other two cases (regular 2-bundle shift with and without adjuster rods). There are several reasons why the same channels could be refuelled in all three cases. Each case started from exactly the same age distribution (i.e., the fractional burnup of a particular channel at the starting point of the simulation was the same in each case). Hence, the choice of channels for refuelling at the start of the simulation would be the same (or very similar). The cases had similar power and flux distributions (in particular, the cases with adjuster rods present had nearly the same channel power distributions), and so would tend to "age" at the same rate. With a 2-bundle shift fueling scheme, bundles have accumulated about 90% of their burnup by the time they reach the final position in the channel. Hence, refuelling a channel before it has reached its target burnup has a relatively minor effect on the average discharge burnup and peak powers.

The action of the zone controllers was not modelled during the simulation: the zone control fill was kept at 45% during the entire simulation. This approximation is conservative, since in practice the zone control fill changes in order to minimize the deviations in zonal power from the reference values.

Power and power-boost envelopes were derived from the results of the time-dependent refuelling simulation. Envelopes were calculated for both 37-element and CANFLEX fuel, in the following manner. FMDP stores the power and burnup of every bundle in the core (in the 'STORE' file), at each time-step in the simulation at which the flux is calculated. From the WIMS cell code, the relative element power and burnup distributions within the bundle are known as a function of average bundle burnup. Hence, the element power and burnup can be determined for each ring of fuel, for every bundle in the core, at each time-step in the simulation. The program PHISTRY(13) was used to do this, along with the FMDP 'STORE' file (containing the bundle power and burnup information) and the WIMS 'TAPE 16' (which contains the relative element power and burnup distribution as a function of average bundle burnup). PHISTRY produces a "scatter plot" of element power and corresponding element burnup for every bundle in the core, for each ring of fuel, for selected times in the simulation. From this "scatter plot" the power envelope is drawn as a smooth curve through the power/burnup points, such that no points lie above this envelope. Thus, the power envelope shows the maximum element power (or linear element rating) in the core, as a function of element burnup, for each ring of fuel.

Although the simulation was done using the CANFLEX bundle, power envelopes were deduced for the 37-element bundle by simply substituting the WIMS 'TAPE 16' corresponding to the 37-element bundle. PHISTRY then used the bundle powers and burnups in FMDP which were calculated using CANFLEX fuel, with the intra-bundle element power and burnup distributions appropriate for the 37-element bundle. This method assumes that the axial bundle power and burnup distributions would be similar for the two bundle types, which is a good approximation.

For the 37-element bundle, scatter plots of linear element rating (element power divided by element length) vs element burnup were calculated only for the outer ring of fuel, since the maximum power always occurs in this ring. The CANFLEX bundle has two pin sizes, and the location of the peak linear element rating shifts from the outer ring (ring 4) to low burnup, to ring 2 at bundle average burnups greater than about 7 MWD/kg (burnups are given in terms of heavy element mass). Hence, for the CANFLEX bundle, linear element ratings were calculated for both rings 2 and 4.

Power boost envelopes were derived in a similar manner to the power envelopes, using PHISTRY with input from FMDP and WIMS. PHISTRY computes the change in power for each bundle in the core between successive flux calculations. If a channel was refuelled during the time-step, the program uses the bundle powers at the new bundle position after refuelling, and at its old location before refuelling, to determine the change in power. WIMS is used to convert the average bundle power and average bundle burnup into element powers and burnups. PHISTRY produces a scatter plot of power boosts over the specified time interval.

RESULTS

Time-average Calculations

The time-average radial channel power distributions are illustrated in Figure 4 for a row of channels along the horizontal mid-plane of the core. With SEU fuel, there is greater opportunity for flattening the radial channel power distribution through "burnup flattening" (i.e., through the use of different irradiations, or refuelling rates, in different parts of the core) than with natural uranium fuel, because the resultant fractional burnup penalty is smaller. As a result, the peak channel powers in Figure 4 are lower than with natural uranium fuel. The channel power distributions for the cases with adjuster rods present are nearly identical. The flattening effect of the adjuster rods on the radial power distribution is clear. The radial form factor (ratio of average to peak channel power) is 0.88 with the adjusters present, and 0.86 with the adjuster rods removed from the core.

![Figure 4: Radial Channel Power Profiles](image-url)
In operating reactors, channel flows and/or coolant inlet temperatures are chosen to match the channel power distribution with natural uranium fuel. Hence, in practice it may not be possible to use SEU in operating reactors to realize the degree of flattening achieved here. SEU provides great flexibility in shaping the channel power distribution, and a channel power distribution corresponding to natural uranium fuel could as easily be achieved, if so desired. The powers and power envelopes calculated in this study can be renormalized to another value of peak time-average channel power, by multiplying by the ratio of the desired peak time-average channel power to the peak time-average channel power in this study.

Figure 5 illustrates the axial time-average bundle power profiles for a central channel for all three cases, normalized to the same channel power (6500 kW). The greatest degree of axial flattening is achieved with the axial shuffling fuel-management scheme. In other words, this fuel-management option yields the lowest bundle power for a given channel power. The regular 2-bundle shift scheme with no adjusters in the core also results in a good axial form factor. In this case, the power distribution peaks at the third bundle from the refuelling end of the channel, and decreases monotonically along the length of the channel. With a regular 2-bundle shift scheme, the adjuster rods depress the power in the center of the channel, resulting in a larger peak bundle power for a given channel power, and an asymmetrical double hump in the axial power distribution.

The effect of the axial power distribution on the critical channel power (CCP) has been studied for the time-average equilibrium axial power distribution corresponding to a regular 2-bundle shift fuelling scheme with no adjuster rods. In this case, the CCP is about 4% greater if the direction of fuelling and the direction of coolant flow are the same, than if they are in opposite directions.

From the time-average power and burnup distributions, power histories can be constructed, which show the variation of bundle power and bundle burnup as a bundle gets shifted along the channel. These are illustrated in Figures 6 through 8 for a typical central channel for the three fuel-management options. These are particularly enlightening in this study, since the time-dependent fuel-management simulations were not carried out for a long enough period of time to follow any bundles completely through the channel. While the time-average histories do not show the variations in power due to refuelling, they do indicate the general features that can be expected in typical power histories. The power histories for axial shuffling and a regular 2-bundle shift without adjusters are remarkably similar (Figures 6 and 7), and are excellent from the point of view of fuel performance. Power boosting is either not present (bundle 2 for axial shuffling), or occurs only for relatively fresh fuel, which is resilient to power boosts. For the case of a regular 2-bundle shift with the adjuster rods present, power boosts of at least 100 kW will occur at higher burnups in shifting from positions 7 to 9, and 8 to 10 (Figure 8).
Further time-average results are given in Table 1. In each case, $k_{eff}$ is close to 1.000. The axial shuffling fuel-management scheme results in the lowest peak bundle and channel powers. For a regular 2-bundle shift fuelling scheme, while the radial channel power flattening is greater with adjuster rods present, the axial flattening is better when the adjuster rods are removed. In fact, the greater axial flattening for the case without adjusters more than compensates for the lack of radial flattening, and the peak bundle power is lower with adjuster rods removed. Note too that greater radial channel power flattening could be achieved without adjuster rods, through adjusting the burnup distributions of the core. It is also interesting to note that the burnup for the case without adjuster rods is not much higher than for the other cases.

The reactivity worths of some of the control devices are also given in Table 1. The worth of the adjuster rods is 11.1 mk with axial shuffling, and 8.2 mk for a regular 2-bundle shift. The reactivity required to compensate the increase in xenon during 30 minutes following shut-down from full power (i.e., 30 minutes xenon over-ride time) has not been calculated in this study, but has been calculated in earlier studies\(^8\). In particular, it has been found that WIMS provides a good estimate of the xenon reactivity transient, as long as normalization is to the flux-squared average thermal flux, obtained from a reactor calculation. The reactivity required for 30-minute xenon over-ride time has been calculated for a core with repositioned reactivity devices, using both WIMS, and a 3-dimensional spatially-dependent model. With 1.2% SEU, the reactivity required for 30 minutes xenon over-ride time is about 9 mk. Assuming the xenon reactivity transient during a shutdown is linear over the first half hour or so, then in the case of axial shuffling, the adjuster rods provide about 37 minutes xenon over-ride time, while in the case of a regular 2-bundle shift, the adjuster rods provide about 27 minutes xenon over-ride time. These estimates would have to be verified for the current core configurations by more detailed calculations, and with incremental cross sections for the adjuster rods calculated explicitly for 1.2% SEU.

The worth of the shut-off rods is about 80 mk for all three cases, which is judged to be ample safe reactivity for shutdown purposes. The zone controller worth is also judged to be sufficient for both bulk and spatial control, in all three cases.

### Table 1: Time-Average Results

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<th>Adjusters out</th>
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<td>$k_{eff}$</td>
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<td>0.99775</td>
<td>0.99942</td>
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<tr>
<td>Maximum Bundle Power (kW)</td>
<td>732</td>
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<td>735</td>
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<tr>
<td>Maximum Channel Power (kW)</td>
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</tr>
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<td>Average Burnup (MWd/kg)</td>
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</tr>
<tr>
<td>Feed Rate (Bundles/day)</td>
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<tr>
<td>(Channels/day)</td>
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<td>Adjuster rod worth (mk)</td>
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<tr>
<td>Zone-control worth (mk)</td>
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<tr>
<td>Shut-off rod worth (mk)</td>
<td>79</td>
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</table>
Instantaneous Calculations

Figure 9 illustrates the channel over-power map for part of the core in the instantaneous calculation for the case with axial shuffling. The over-power (or ripple) of a channel is simply the ratio of the channel power to the time-average power of that channel (multiplied by 1000 in Figure 9). If a channel has an over-power equal to 1.0, then its instantaneous power is equal to its map in Figure 3, it is seen that high over-powers correspond to low ages (i.e., power of that channel (multiplied by 1000 in Figure 9). If a channel has an over-power greater than 1.0, then its instantaneous power is equal to its map in Figure 3, it is seen that high over-powers correspond to low ages (i.e., power of that channel (multiplied by 1000 in Figure 9).

Time-dependent Fuelling Simulations

The key results of the simulation are presented in Figure 10. For the cases with the adjuster rods present in the core (axial shuffling and regular 2-bundle shift), \(k_{\text{eff}}\) shows a variation of about 0.5 m/s about its average value, with a slight tendency to increase over the last half of the simulation. With adjusters out, \(k_{\text{eff}}\) increases more, by about 2 m/s over the last half of the simulation. The reason for this is that the equilibrium refuelling rate for the case without adjusters is about 3% lower than for the other cases (Table 1). However, the same refuelling rate was used in the simulation for each case, since exactly the same channels were refuelled during each time-step between flux calculations. The refuelling rate could have been lowered by 3% in the case without adjusters, while refuelling the same channels as in the other cases, by shifting the enrichment and the number of bundles shifted, and the core configuration should have but a small effect on the CPPF. The value of CPPF for natural uranium using an 8-bundle shift fuelling scheme is about 1.10.

It is usual to restrict the calculation of the CPPF to either the channels having an instantaneous power greater than 0.9 of the maximum channel power at that time, or to the high-power region of the core. The peak over-power for the whole core often occurs in a peripheral channel, which may have a larger margin to dryout than the other channels. For instance, at 40 days in this simulation, with the axial shuffling fuel-management scheme, the largest over-power for the whole core was 1.244, and occurred in the peripheral channel E-3. The location of the largest over-power remained at this channel for the rest of the simulation.

The average CPPF over the simulation is 1.11 for all three cases (calculated for channels having an instantaneous power greater than 0.9 of the instantaneous maximum channel power). The CPPF is determined mainly by the enrichment and the number of bundles shifted, and the core configuration should have but a small effect on the CPPF. The value of CPPF for natural uranium using an 8-bundle shift fuelling scheme is about 1.10.

The variation of maximum channel power is almost identical for the two cases with adjuster rods present, yielding an average maximum of 6479 kW for the axial shuffling scheme, and 6503 kW for the regular 2-bundle shift. For the case without adjuster rods, the average maximum channel power is 6816 kW, 5% greater than the other two cases. The average peak bundle powers over the simulation are 777 kW for the axial shuffling scheme, 868 kW with adjusters present, and 829 kW with adjusters absent. The peak powers are well within limits, even allowing for uncertainty in the modelling. The axial shuffling scheme has very low peak channel and bundle powers.

![Figure 9: Channel Overpower Map, Axial Shuffling](image-url)
FIGURE 10: RESULTS OF REFUELLING SIMULATION

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The pin power envelopes are shown in Figure 11 for the case with axial shuffling at 70 full-power-days in the simulation, a time at which the peak powers were near their maximum values. Envelopes are shown for ring 4 of the 37-element bundle, and for rings 2 and 4 of the CANFLEX bundle. With axial shuffling, the power envelope for ring 4 of the CANFLEX bundle is always greater than for ring 2, regardless of burnup. The power envelopes peak at about 3 MWd/kg, and thereafter decline monotonically with increasing burnup. The power envelope for the outer ring of the CANFLEX bundle is about 20% lower than for the 37-element bundle. The peak linear element rating is 41 kW/m in the CANFLEX bundle, and 52 kW/m in the 37-element bundle. One would expect very little fission gas release at these low CANFLEX ratings. Recent calculations of CANFLEX fuel performance during a postulated LOCA have verified the expected very low strains at these low power levels, and the consequent benefits in terms of safety.

The power envelopes for a regular 2-bundle shift fuelling scheme in a core without adjuster rods are similar to those for axial shuffling, and are not shown here. The peak ratings are slightly greater than for axial shuffling -- 45 kW/m for the CANFLEX bundle, and 56 kW/m for the 37-element bundle. With the CANFLEX bundle, most of the fuel in the core is below 40 kW/m.

With a regular 2-bundle shift fuelling scheme with adjusters in the core, the power envelopes show a slight increase at high burnup, due to the asymmetric double hump in the axial power distribution caused by the adjuster rods (Figure 12). The peak fuel ratings are 47 kW/m for the CANFLEX bundle, and 58 kW/m for the 37-element bundle. The power envelope for ring 2 of the CANFLEX bundle is greater than for ring 4 at burnups between 12 and 17 MWd/kg.

The power-boost envelopes are very similar for the case of axial shuffling, and a regular 2-bundle shift fuelling scheme in a core without adjuster rods (Figure 13 shows the power-boost envelope for the 2-bundle shift without adjusters). Sizable power boosts occur only at small burnups, when the fuel is resilient to power boosting. With the 37-element bundle, the largest power boosts are 36 kW/m at an element burnup of 3 MWd/kg. The power boosts decline monotonically with increasing burnup, having values of 9 kW/m at a burnup of 11 MWd/kg, and 3 kW/m at 20 MWd/kg. With CANFLEX fuel, the power boosts decline from a peak of 29 kW/m at an element burnup of 3 MWd/kg, to 7 kW/m at 11 MWd/kg, and 2 kW/m at 20 MWd/kg.

The power-boost envelopes with the adjusters in the core show power-boosting at higher burnups (Figure 14). With the 37-element bundle, power-boosts of 12 kW/m occur at an element burnup of 18 MWd/kg, while with CANFLEX fuel, these boosts are reduced to 10 kW/m. While this is not optimal from the fuel performance point of view, it is judged that the CANFLEX bundle will be able to withstand these power-boosts, due to optimization of the pellet design.

There are other sources of power boosting. One arises from the transient changes in power during the actual refuelling operation, as the bundles move along the channel. These transients could be particularly important with axial shuffling, since the entire channel is emptied during refuelling. Power-boosting would also occur with load-following. The power-boost envelope arising from load-following would simply be the steady-state power envelope, scaled down by some fraction depending on the extent of loading-following. Each of these situations differ in the time over which the power ramp occurs, and the time at the ramped power.

In assessing fuel performance at extended burnup under these operating conditions, we are faced with the fact that our current data-base is insufficient to predict with certainty the performance of either the 37-element or CANFLEX bundles. However, our experimental data-base and our fuel models give us confidence that the lower ratings, and internal design modifications in the CANFLEX bundle, will lead to a defect rate which is equal to or lower than that currently achieved with natural uranium fuel. This will be confirmed through in-reactor testing under the CANFLEX program(3,4,5).
SUMMARY

Axial Shuffling

In the central channels near the adjuster rods, axial shuffling is an excellent fuel-management option with 1.2% SEU, from the reactor physics and fuel performance perspectives. Axial shuffling provides good flattening of the axial power distribution, with low peak powers. With the CANFLEX bundle, the ratings of almost all of the fuel elements in the core can be reduced to below 40 kWm, with significant operational and safety benefits. The average peak bundle and channel powers during the time-dependent fuel-management simulation were very low: 777 kW and 6479 kW respectively, while the average CPPF was 1.11. Axial shuffling results in a declining power history, with very little power boosting at extended burnup as a result of refuelling. The reactivity worths of control devices with axial shuffling is ample. Preliminary assessments indicate that axial shuffling is feasible in all CANDU reactors. Further studies are needed to address the details of fuel handling in CANFLEX and Bruce-type reactors, and to determine the transient powers during refuelling, and their impact on fuel performance and flux detector readings.

Regular 2-Bundle Shift with Adjuster Rods Present

This fuel-management option results in an asymmetric axial power distribution, with the adjuster rods pulling down the power in the center of the channel. As a result, there is some power boosting at extended burnup: at a burnup of 20 MWd/kg, there are increases in linear element ratings of about 11 kW/m. However, it is judged that the CANFLEX bundle will be able to handle this power history without excessive fuel failures, due to the optimization of the pellet design. The worths of the reactivity devices are adequate, although the xenon override time is slightly reduced. The average peak bundle and channel powers during the refuelling simulation were 829 kW and 6816 kW, respectively. The average peak channel power was greater than in the other two cases, due to less radial flattening of the channel power distribution with no adjuster rods. The CPPF was 1.11.

Regular 2-Bundle Shift with Adjuster Rods Absent

This option provides an indication of what might be expected in using a regular 2-bundle shift fuelling scheme with 1.2% SEU in the Bruce A reactors, in which there are no adjuster rods. This is a very attractive fuel-management option, especially if the direction of fuelling and coolant flow are the same, which would increase the margin to dryout. While the axial flattening is not as great as with axial shuffling, it is about the same as that provided by the adjuster rods with natural uranium fuel. The axial power distribution is excellent from the viewpoint of fuel performance, with the peak peaking at the refuelling end of the channel and decreasing along the length of the channel. Hence, significant power boosting during refuelling occurs only for relatively high burnup. In the refuelling simulation with the CANFLEX bundle most of the fuel in the core was at ratings below 40 kWm. Average peak bundle and channel powers during the simulation were low, 829 kW and 6816 kW, respectively. The average peak channel power was greater than in the other two cases, due to less radial flattening of the channel power distribution with no adjuster rods. The CPPF was 1.11.

CANFLEX vs 37-element

With any of these fuel-management options, the fuel performance database at extended burnups is not sufficient to predict with certainty the performance of either the 37-element or CANFLEX bundle, especially if load-following capability is required. The CANFLEX bundle features a 20% reduction in peak element ratings, and optimization of the pellet design for extended burnup. These features provide confidence that the CANFLEX bundle will be able to meet the duty-cycle required for 1.2% SEU with low probability of fuel defect. The CANFLEX program now in progress will help to confirm this.

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Session 2:

Small Reactor Development I

Chairman:

J.W. Hilborn, AECL-CRNL
THE PRESSURIZED LIGHT-WATER REACTOR - A WELL-OPTIMIZED MARINE-PROPULSION POWER SOURCE

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ABSTRACT

Pressurized Water Reactors (PWRs) originally developed for the U.S. submarine program have achieved a high level of maturity and proven capability both as a marine power source and in civilian power-generation applications. Compact design, excellent inherent load following and core lifetimes that challenge conventional machinery refit/refurbishment cycles are features that combine to make the PWR system so attractive.

Requirements for a marine-propulsion power source are identified and some of the relevant characteristics of a typical marine PWR system are reviewed in this paper.

INTRODUCTION

This paper is intended to give an overview of some of the special features that have been considered in the design and development of pressurized water reactors in a marine environment.

HISTORICAL

The first practical demonstration of the use of nuclear power for submarine propulsion occurred in late 1954 with the commissioning of the USS NAUTILUS. NAUTILUS was powered by a Westinghouse pressurized water reactor (PWR) design. This same basic reactor design was subsequently developed as the first U.S. civilian nuclear power-generating station at Shippingport. The NAUTILUS was followed in early 1957 by the USS SEAWOLF, which was powered by a liquid-metal (sodium) cooled reactor designed by General Electric. The success of NAUTILUS and SEAWOLF clearly demonstrated that nuclear propulsion could work, and could work well. Although the liquid-metal plant in SEAWOLF successfully propelled the submarine for almost two years (during which time she completed a record-breaking 60 day submerged run, travelling a distance of 13,761 miles submerged), it became clear that the PWR design was preferable for naval application. In December, 1958, SEAWOLF shut down her sodium-cooled plant for the last time, to be refitted with a PWR.

The NAUTILUS design was developed to power the SKATE class submarines, first commissioned in 1957. Further advances in PWR technology led to the development of the power plant for the SKIPJACK class submarines. Known as the S5W core, this reactor design was used for all the major submarine classes in the U.S. Navy, to the commissioning of the LOS ANGELES class (S6G reactor) in 1976 and the OHIO class (S8G reactor) in 1981. The S5W reactor design alone has powered over one hundred vessels in the U.S. fleet. Altogether, over 180 nuclear propelled submarines have been authorized by the U.S. Congress, with 132 currently in service. In addition, 14 nuclear-powered surface ships are in operation, with 2 more authorized.

The S5W reactor design was also the power unit of the first UK nuclear submarine, DREADNOUGHT, under the terms of the U.K. – U.S. Agreement in 1958. The U.K. nuclear submarine fleet now numbers 22 vessels, using pressurized water reactors developed from the S5W design. Denied access to the US reactor technology of the late fifties, France developed its own versions of the pressurized water reactor. The design of their AMETHYSTE class vessels was being offered for the proposed Canadian submarine fleet, as was the current TRAFALGAR class by the U.K. There are now eleven nuclear-powered submarines in the French fleet.

The USSR submarine fleet is now estimated at over one hundred and ninety vessels. Most of these vessels are powered by two pressurized water reactors developing up to 30,000 shp.

The newer USSR submarine classes have higher core-power ratings. Their TYPHOON class nuclear ballistic missile submarine has 2 reactors developing 120,000 shp. These vessels are all thought to have PWR reactors; however, their latest nuclear-powered fleet submarine class, the ALFA/AKULA class (40,000 shp) is rumoured to use a liquid-metal coolant to achieve the high core-power density.

It is conservatively estimated that marine pressurized water reactors have been used in over four hundred submarines and surface vessels throughout the world, making the PWR reactor a well-proven concept for marine nuclear propulsion.

MEETING THE SPECIAL REQUIREMENTS OF A NUCLEAR MARINE-PROPULSION POWER SOURCE

Biological Shielding

The proximity of the nuclear reactor to the crew within the confines of a submarine or a surface nuclear-propelled vessel places special requirements on reactor shielding to cater to both normal and accident situations. The ICRP regulations are generally the basis for all shielding requirements and the limiting doses under these regulations are a. 5 rem/year for the crew; and...
These regulations are the same regulations followed in the design of the CANDU reactor shielding for commercial power stations.

The primary shielding for marine pressurized water reactors is usually a water-filled shield tank which surrounds the lower portion of the reactor pressure vessel. The outside of this shield tank may also be lead lined. Shielding of the upper portion of the reactor pressure vessel, including the control-rod drive mechanism, is usually achieved by the use of removable blocks.

Secondary shielding is provided by a lead coating on the bulkheads of the reactor compartment and around the access personnel passages provided through the reactor compartment. Additional shielding is often provided by placing liquid-storage tanks adjacent to the bulkheads of the reactor compartment.

Access to the reactor compartment is usually prohibited or very restricted with the reactor at power. With these measures the annual doses usually recorded by the crew based on two crews is well within the ICRP regulations.

**Load-Following Capability**

The ability to respond quickly and reliably to changes in power demand is a prime requirement for ship safety under emergency manoeuvring conditions. One of the more demanding applications for a marine power source is in icebreakers, where many power-reversing cycles full ahead/stop/full astern often need to break through ice ridges. Pressurized water reactors can be designed to meet even these demanding requirements for an unlimited number of power cycles without having to engineer modifications to attenuate the power-demand changes seen by the reactor.

**Shock Loadings**

The equipment for marine reactor systems is required to continue to operate under shock loadings that are, in some applications, orders of magnitude greater than the "designed for" conditions of the land-based commercial reactor programs. To cater for seismic activity, a typical "designed for" shock loading for CANDU reactors is 0.3g. The typical accelerations expected for a marine reactor system for commercial shipping (based on SAVANNAH information) is 0.6g. This value was also considered to be the shock loading that could be experienced in a collision involving considerable hull-structure deformation. Testimony of the crew of the ANDREA DORIA, following her collision with the STOCKHOLM, was that no impact was felt in the collision (indicating that the upsetting force was extremely low).

The design acceleration for the nuclear systems of the Canadian Nuclear Icebreaker project in 1978/79 and based on a POLAR 10 capability (150 000 shp) was 3g in any direction. Full-ahead, full-astern power cycling and a ramming capability to break through ridges of ice up to 30 feet in thickness were features of this vessel design.

Nuclear submarines are required to withstand hostile acts of war such as depth charging in battle conditions. Although the design shock loading for nuclear submarines is classified information, it is known to be well in excess of the 3g design value for icebreaker systems.

The requirement for submarine nuclear systems to withstand high shock loading is reflected in all aspects of the NPS design.

**Safe and Stable Operation under all Sea Conditions**

Safe reactor operation is demanded for an operating environment that has to include all sea conditions and submarine manoeuvres. The nuclear propulsion system has to be designed to operate without power restrictions over a range of boat attitudes and dynamic responses. The instrumentation to measure and control water levels in the primary and secondary loops has to be designed to allow for level variations due to boat motion and at the same time provide indications of genuine level changes for early control and safety responses. For example, the pressurizer level provides one of the early signals for loss of primary coolant accidents (LOCA's), and instrumentation channel redundancy of at least 2 out of 3 and often 3 out of 4 measurements is demanded, to avoid spurious signal initiation even under normal operating conditions. With adverse sea conditions additional precautions need to be taken to prevent an inadvertent shutdown of the reactor. Sensors placed to give coverage for all boat attitudes and use of median signal selection are the techniques often used in these circumstances.

The steam generators for PWR marine reactor systems can be of the recirculating or once-through design systems, the once-through version being usually favoured for "integrated" designs where the steam generator and the reactor are combined within the same reactor pressure vessel. In either case, close control of the secondary steam-generator water levels is required. Poor steam-generator level control can result in poor control of steam quality to the turbines. More importantly however, poor control of the steam-generator feedwater directly affects the stability of the reactor power.

The control rods of pressurized water reactors are provided to bring the reactor to a critical condition. Other than controlling the primary circuit average temperature and accommodating changes in core poison concentrations and through life burnup, the control rods are not usually required to move quickly in response to core-power load changes. Control-rod movement, affected by adverse sea conditions, is not a factor affecting stable operation.

Variations in primary-circuit flow rate are potentially another source of unstable power operation. These variations in flow rate can be induced by adverse sea conditions with the reactor operating under natural-circulation conditions (significant power operation under natural circulation in the primary circuit is often a design feature of some PWR reactor configurations.) Under pumped primary-circuit coolant conditions this effect is negligible.

The negative-temperature coefficient inherent in the PWR reactor design can itself give rise to power instabilities. These instabilities usually arise
where the primary-circuit recirculation times are in excess of the power demand responses. For marine reactor applications, where fast power changes may be expected, primary-circuit recirculation times comparable to the demand response time are usually a necessary feature of the design. A poor transient response will result in large-volume surges in the primary circuit, which have to be accommodated in the pressuriser.

A rapid shutdown of the reactor for safety reasons is achieved by releasing all or some of the control rods. These then fall into the core under gravity, with possibly some spring assistance. This safety demand could arise at any time, and it is therefore customary to allow for the worst sea conditions by defining a maximum core inclination which would not be exceeded.

By these means, both safe and stable power operation can be optimized for the nuclear propulsion system under all sea conditions.

**Reactor Control**

Changes in reactor power are affected by changes in reactivity, where reactivity is a measure of the tendency for the number of neutrons taking part in the fission process to increase or decrease. Changes in reactivity of the core can be made by insertion or withdrawal of the control rods and also by simply changing the temperature of the reactor coolant and, to a lesser extent, the temperature of the reactor fuel.

The magnitudes of the coefficients of reactivity for the fuel and coolant as well as the sign of these coefficients, either positive or negative, are features that can, to a certain extent, be designed in. The multiplication factor $k$ for a chain reaction is defined as the ratio of the number of neutrons in any one generation to the number of neutrons in the immediately preceding generation. An example of how a typical value of $k$ varies as a function of the ratio of hydrogen atoms (in the water coolant) to U235 atoms in the fuel is shown in Figure 1. A core designed with H/U235 atom ratio, say, of 250 would show a strong positive-temperature coefficient; that is, raising the temperature of the coolant would decrease the hydrogen atom concentration and hence the value of H/U235 ratio. This would result in an increase of the multiplication constant and hence core reactivity. Similarly, a reactor with a H/U235 ratio of 100 would have a strong negative-temperature coefficient. It is the negative coolant-temperature coefficient of reactivity that accounts for the inherent load-following capability of PWR's.

Another temperature coefficient influencing changes in the overall core reactivity is the fuel-temperature coefficient. A strong fuel-temperature coefficient may arise from the so-called Doppler effect, and the magnitude of this Doppler coefficient depends on the type of fuel, metallic or oxide, and the enrichment. Neutrons can be lost to the fission process by being absorbed by the U238, and the probability of this occurring is directly related to the temperature of the fuel. Again, the Doppler coefficient depends upon the materials used. Low-enriched U02 fuel elements having poor thermal conductivity and high operating temperatures show substantial Doppler changes in reactivity and hence in core-power levels. Metallic fuel, on the other hand, with good thermal characteristics between the fuel and cladding, will only exhibit small fuel Doppler-coefficient effects. Changes in reactivity caused by the Doppler effect in going from self-sustaining powers up to 100% full-power operation for oxide-fuelled submarine reactors might typically introduce a decrease in the average coolant temperature of up to 6°C, and some control-rod movement would therefore be required to restore the average primary temperature to its nominal value. A metallic fuel having good thermal conductivity, on the other hand, would require little or no corrections to core reactivity (and hence the average primary temperature) following load changes.

It is a common misconception that the reactor control rods alone are responsible for affecting core load changes. As explained above, pressurized water reactors for submarines are designed to have a large negative coolant-temperature coefficient of reactivity and it is this "designed in" negative-temperature coefficient that accounts for the inherent load-following capability of PWR's. With the reactor critical and operating at power, increasing the steam demand causes cooler primary coolant to leave the steam generators. When this cooler water reaches the reactor core, the resulting reactivity addition causes the core power to increase to the new demanded steam power. Control-rod movement is generally unnecessary in the load change, except to maintain the average coolant temperatures by compensating for any fuel-temperature Doppler reactivity changes in the power excursion.

Typical transient responses to changes in steam demand are shown in Figure 2. The variations of the primary system temperatures and volume surges and the reactor power with time are illustrated for a range of expected values of the temperature coefficient. It is a usual design requirement for submarine reactors that there are no restrictions on the number and frequency of power-range maneuvers over the life of the plant, and that there are no requirements for periods of steady-power running to "condition" the reactor fuel following severe and rapid load changes.

The relatively small size of marine PWRs means that the core-power distribution is always close-coupled. The spatial core-power distributions giving rise to core-power flux tilts in the larger reactors are not a

![Figure 1: Effect of H/U235 Ratio on Multiplication Factor](image-url)
Compact Designs

The general concept of the pressurized water reactor allows for some design and layout flexibility in the configuration of its major components. Most commercial power-production reactors are of the distributed or loop-type, where the principal components such as the reactor, steam generator and main coolant pumps are interconnected by pipework. Such an arrangement in a marine application gives good accessibility and weight distribution. With these loop-type reactors it is usual to have two or more loops connected to the reactor pressure vessel, with each loop having its own steam generator and main coolant pump capability. A single pressurizer connected to the reactor pressure vessel maintains the primary circuit pressure at its desired operating value. The steam generators of loop-type reactors are generally of the recirculating type, where the primary circuit is contained within the inverted U-tube nest arrangement. The loop-type distributed arrangement used in the TRAFALGAR class submarines is similar to that found in commercial PWRs.

There are some design incentives to mount the major primary-circuit components in close proximity to reduce pumping powers and achieve a more compact general arrangement. Shown in Figure 3 is an example of the close coupling that has been achieved by the general arrangement of AMETHYSTE class submarine propulsion units.

MARINE PWR CHARACTERISTICS

PWR Mission Profiles and System Availability

The mission profiles expected of a nuclear marine reactor power source vary considerably according to their application. Commercial shipping interests were investigated by a number of countries during the 1960's and early 1970's. One such vessel was the OTTO HAHN. Built as a bulk ore carrier, the nuclear propulsion system operated very successfully. However, the concept was never economically viable, due mainly to the difficulties experienced in obtaining permission to load/unload in foreign ports. In general, permission had to be requested for every docking, making any commercial value in these operations meaningless even though the performance from the reactor system met all expectations.

Pressurized water reactors have been operated in nuclear-powered icebreakers by the USSR for many years starting with the LENIN launched in 1959. The main advantages of these nuclear-powered vessels is their ability to stay on station, breaking ice without the need to return regularly to base for refuelling. The USSR now operates four nuclear-powered icebreakers. Extensive studies of the Canadian Coastguard during the late 70's indicated significant advantages of a nuclear-powered icebreaking option based on one annual inspection and overhaul.

The operation of nuclear-powered submarines places even more demanding availability requirements on the nuclear propulsion system. Long periods of submerged operation lasting many months are not uncommon, and safe operation under the Arctic ice has been demonstrated regularly since the first under-ice expedition to the North Pole was made by the NAUTILUS in 1958.

Since there is always sustained full-power capability available, even while submerged, the concept of economic stationing speeds is no longer a constraint.

FIGURE 2: SYSTEM RESPONSE TO RAMP STEAM DEMAND

FIGURE 3: CAP REACTOR GENERAL ARRANGEMENT
circuit components. In the once-through design, the feedwater is raised to saturation and even to some superheat conditions within the steam-generator tubes themselves. The tube bundles can also be manufactured to particular vessel designs. This gave rise to the "integrated" concept where the core, steam generator, and in some cases even the means of system pressurization were contained within the reactor pressure vessel. This integral design of PWR was developed and successfully used on the OTTO I1AHN. Good natural circulation was a feature of this primary-circuit design.

**Safety Features**

The safety features provided are designed to ensure that the fission fragments remain contained for all credible core operating and accident conditions. Three barriers are considered in making the safety justification. These barriers are the fuel cladding, the primary circuit and the containment boundary. Where metal oxide fuels are used, the sintered matrix exhibits good retention properties for many of the fission fragments.

For commercial power reactors, particularly those where on-load refuelling is a feature, a limited leakage of fission products may be tolerated into the primary circuit, where defected-fuel-detection equipment and clean-up facilities are also a feature of the reactor design.

For marine pressurized water reactors, core lifetimes extending over many years are usually a design requirement and any significant fission-product burden in the primary circuit becomes a major factor affecting system accessibility in the close confines of a submarine environment. For these reasons, rigorous pretesting of the reactor fuel assemblies before installation and the adherence to conservative design and operating margins are requirements for marine reactors.

The final barrier to the fission products is the containment pressure boundary; within a submarine this is usually defined by pressure retaining bulkheads and part of the pressure hull of the submarine.

To maintain the integrity of these barriers to the release of fission products from the fuel under accident conditions, safety systems are provided. These systems ensure a rapid insertion of the reactivity control devices, enable continued core cooling and cause the containment boundary to be isolated on receipt of the appropriate signals. In a marine environment, and particularly a submarine environment, the safety of the ship demands a reliable and highly available power source. For these reasons, responsible reactor protection sensors and devices are a necessary design requirement. It is also a practice to provide a rapid reactor shutdown, but of only a limited reactivity depth in some instances, to cater to transient conditions where the abnormal conditions demanding a reactor shutdown can be immediately dealt with and the reactor quickly brought back to power operation.

**Core Lifetime**

The high availability of CANDU reactors is attributed largely to the on-load refuelling capability provided. For pressurized water reactors, a refuelling requires the reactor to be shutdown and the reactor pressure vessel to be opened. Land-based civilian PWRs have a lifetime between each core refuelling of approximately eighteen months and a fresh charge of fuel usually takes a minimum period of 2 months before the reactor can be returned to operation.

The "ash" of the fission process consists of the fission products created in the fuel. Some of these fission products have large cross sections for the absorption of neutrons and therefore act as poisons affecting the overall multiplication of the reactor. Some of these poisons continue to be formed long after the reactor is shut down and, unless sufficient reactivity is available to override the peak of the poison transient, the reactor is prevented from being restarted until the poison has decayed sufficiently away. The most important of these poison fission products is Xenon (Xe135). This fission product, having a half life of 9.2 hours, is produced from the decay of 1135, having itself a half life of 6.7 hours. With the reactor shutdown after a period of power operation the xenon poisoning therefore builds up in the core, reaching a peak value approximately eleven hours after shutdown.

The relative xenon poisoning reactivity as a function of time after shutdown is shown in Figure 4. Changes in core reactivity due to xenon poison effects are compensated for by the control rods.

For marine reactors, sufficient excess reactivity has to be always available to override any xenon reactivity transients by control-rod movement alone, to ensure the ship's safety. For marine reactors, therefore, the end of the core lifetime is taken to mean the time in core life when such xenon transients can no longer be overridden for normal operating conditions.

Core lifetimes can be extended by designing in excess reactivity and suppressing it by the careful placement of burnable poisons. One such burnable poison is boron. The effect of incorporating burnable poisons on core lifetime is shown in Figure 5. This figure illustrates how the effective reactivity control vested in the control rods can be substantially complemented to permit these longer core lifetimes.
Positive Reactivity

Range of reactivity change to be controlled due to fuel burnout

Net reactivity change

Burnable poison

Most reactive time in core lifetime

Time of Operation

FIGURE 5: EFFECT OF BURNABLE POISONS

Typically, extension of core lifetimes up to between five and seven years or more, depending on the core utilization, can be realized. These core lifetimes are comparable to the projected periods between major refits for modern submarines.

Core Power

Because the placement of the fuel in the PWR reactor core is fixed between refuellings, it is possible to incorporate measures to flatten the power-generation profiles across the reactor by suppressing local power peaks within the fuel assemblies. This can be achieved by varying the local fuel enrichment. In commercial PWR power reactors it is now customary to design the core with typically three fuel enrichments, with the higher enrichment bundles being placed in the lower flux areas of the core. A refinement of this technique is also used on PWR marine reactors to increase the core power that can safely be taken from the core at all times in core lifetime. In Candu reactors, where on-load refuelling is a design feature, only standard fuel bundles are used. However, some flattening of power-generation flux profiles is achieved by varying the bundle residence times in different channels of the reactor.

Decommissioning of Nuclear-Powered Submarines

Extensive studies addressing the decommissioning of civilian nuclear plants in Canada have been completed in recent years. The overall conclusions show that this activity does not pose undue technical challenge, nor does it lead to an unreasonable financial burden on the users and beneficiaries. In Canada there are currently three Candu reactors in various stages of decommissioning:

(1) Douglas Point;
(2) Gentilly 1; and
(3) NPD.

In simplest terms, the approach in land-based reactors is to remove the fuel from the reactor core and then to place the reactor in such a state that the residual activity in the reactor can be allowed to naturally decay for some 25 to 50 years before beginning to dismantle the core.

In the case of nuclear submarines, a detailed review of possible decommissioning approaches has been performed by the US Navy (including public hearings and review) in which the following options were investigated:

Following docking at Bremerton, Washington and removal of the fuel:

a. perpetual care at the dockside, floating in water, with periodic examination of the submarine hull;

b. following an appropriate activity-decay period at the dockside, filling the submarine hull with concrete and sinking it in the deep ocean; and

c. following an appropriate activity-decay period at the dockside, dismantling the submarine and removing the components and hull sections to the Hanford reservation in Washington state.

Disposal option "b" has been used. However, following extensive review, option "c" is now selected as the reference approach in the U.S.

CONCLUSIONS

Pressurized water reactor designs for nuclear submarine propulsion have achieved a high level of maturity and proven capability.

It is estimated that well over three hundred pressurized water reactors are operating in nuclear submarines throughout the world.

Excellent inherent load following can be expected and core lifetimes of up to seven years between refuellings are achievable.

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THE AMPS 1000: AN ADVANCED REACTOR DESIGN FOR MARINE PROPULSION

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ABSTRACT

It has long been recognized that a key determinant of the worth of a submarine is the extent to which it can undertake submerged operations with long endurance. Considerable progress has been made in the development of the family of low power nuclear reactor-based systems known as the Autonomous Marine Power Source (AMPS). The AMPS is being developed for installation in a wide range of submarines, both civilian and military, to provide them with very long endurance atmosphere-independent power.

The AMPS technology adapts advanced design principles of process-inherent reactor safety to the submarine environment. This permits significant plant simplification and minimum reliance on operator action for safe and effective operation. The AMPS 1000 version is designed expressly for integration in a range of modern 2000-tonne class diesel-electric naval submarines. The plant is rated to supply in excess of 1000 kWe to the submarine batteries, at seawater temperatures of up to 30°C, in order to propel the submarine at speeds of up to 12 knots while supplying all ship service loads. The submarine would possess virtually unlimited submerged endurance at these speeds and, by maintaining its batteries fully charged, would retain the high-speed sprint capability of the baseline diesel-electric submarine.

INTRODUCTION

Current generation marine nuclear propulsion technology is primarily based on the pressurized water reactor (PWR), having its genesis in the early 1950s. Because such plants were sized for maximum vessel speed, the technology evolved along the lines of high power density reactor cores at high temperature and pressure. This led to a reliance on engineered safety systems and relatively large numbers of highly trained, on-board operating specialists to assure safe and effective plant operation.

Recent trends in submarine propulsion technology have recognized the utility of atmosphere-independent power (AIP) sources at acceptable power levels much lower than those provided by the PWR. This has stimulated the development of lower cost, lower power AIP technologies including the AMPS.

In common with many of the new-generation reactor technologies, the AMPS was conceived [1] as a nuclear reactor featuring inherent safety, low operating complexity and high reliability. The top-level design requirements for the AMPS 1000 version were established as follows:

1. The plant design should facilitate ready integration within current and new generation diesel-electric submarine designs, with minimal adverse impact on the basic submarine performance characteristics.

2. To the extent practical, the plant should not compromise the noise and infrared stealth characteristics of the basic submarine.

3. The reactor design should possess levels of inherent safety, reliability and simplicity sufficient to overcome the prohibitive fiscal, technical and infrastructural constraints experienced heretofore by many navies seeking nuclear propulsion.

4. The acquisition cost of an AMPS-equipped submarine should not exceed the cost of the baseline diesel-electric submarine by more than about 20%.

5. The crew requirements for the AMPS-equipped submarine should ideally be no higher than that for the baseline diesel-electric submarine.

6. The shore support infrastructure needed to maintain, refuel and refit a fleet of AMPS-equipped submarines should be much less burdensome than that associated with a "full nuclear" PWR-equipped submarine fleet.

Specific design requirements for the AMPS 1000 plant are summarized in Table 1. The requirement of providing sufficient electrical power to propel the candidate submarine at 12 knots arises from a number of submarine operating considerations [2] leading to a net power output requirement which varies with submarine design. In practice, for ship applications targeted in the design of the AMPS 1000, a range of power levels from 700 kWe to 1700 kWe emerges.

This paper will describe the 1700 kWe plant on which the majority of the AMPS 1000 engineering definition effort to date has focused. Operating and safety characteristics of the plant, predicted from a series of detailed plant analyses, are also provided.

PLANT DESIGN

General

The AMPS 1000 plant employs a low-pressure, water-cooled reactor as the heat source supplying a low-temperature steam-based Rankine cycle engine which generates the electrical power used to charge the submarine main batteries. All submarine electrical
The AMPS 1000 design is based on components for which the technologies are either proven through years of service under conditions similar to those of the AMPS 1000, or demonstrated as being in an advanced state of development. While the AMPS 1000 represents a distinct new-generation reactor technology, the design nonetheless derives significant benefit from the technological and engineering infrastructure of water-cooled reactors which has been built up over more than 30 years.

The AMPS 1000 represents a version of the generic AMPS design concept outlined in Reference 1. While primary system temperatures and pressures for the AMPS 1000 are slightly higher than those specified for the original lower power level AMPS applications, they remain well below those encountered in current generation marine PWR technology. It will be evident in the remainder of this paper that the basic AMPS design concepts have been implemented in the AMPS 1000 version.

### Reactor

Core The reactor core consists of an array of 51 shrouded fuel assemblies of the type shown in Figure 1, each consisting of 25 fuel rods held in a square array by spacers. The rectangular shrouds provide structural support and protection and a well-defined coolant flow path. As indicated in the core plan of Figure 2, reflector blocks surround the core radially. The beryllium reflector tends to flatten the power distribution. In the current design, eight lattice sites are reserved for regulating rods and ten sites for shutoff rods.

Each fuel rod contains uranium-zirconium-hydride (U-ZrH₁₆) eutectic alloy fuel. Incoloy 800 clad has been shown to provide good strength, ductility and corrosion resistance throughout the fuel lifetime at AMPS 1000 design coolant temperatures. Each fuel rod has a diameter of 1.27 cm and an active length of 56 cm. The fuel matrix contains 45 wt% uranium, enriched to 19.7 % in Uranium-235. Erbium burnable poison is homogeneously interspersed in the fuel matrix to achieve the design core burnup lifetime of 1400 full power days while avoiding the need for more than an operating margin of excess reactivity.

The fuel and control rod assemblies are positioned in the core region by stainless steel grid plates located above and below the core region. The grid plates are in turn fastened to grid plate supports.
within the reactor vessel. Under shock loading conditions, the reactor vessel and its internals are designed to move as a unit, to minimize the likelihood of control rod misalignment. Figure 3 depicts the reactor heat source conceptual arrangement, showing the reactor vessel, core, internal supports, primary shielding, and control rod drive mechanisms located above the core.

Table 2 summarizes the key design parameters of the AMPS 1000 core sized for a 1700 kWe plant output. The low core power density of the AMPS 1000 provides generous margins to fuel design and safety limits. For example, the peak operating fuel centreline temperature of 391°C is well below the rated long term fuel steady state operating limit of 750°C. This rating has been determined based on consideration of irradiation and fission-product-induced fuel growth and deformation. The steady state minimum departure from nucleate boiling ratio (DNBR) of 5.4 is indicative of the large margins to fuel element dryout in the AMPS design. This feature helps ensure that fuel cooling remains adequate even under degraded core cooling conditions.

Primary Heat Transport System The Primary Heat Transport (PHT) system employs three centrifugal canned pumps (one of which is on standby) to circulate the light water coolant to the core inlet plenum. The three pumps are installed in parallel and have common discharge and supply headers. The coolant flows upward through the core lattice into the outlet plenum to remove heat produced in the fuel elements. The coolant leaves the outlet plenum via core outlet piping to two headers, each feeding one steam generator, where the heat is transferred to the secondary side working fluid. The coolant then returns to the inlet of the pumps via the common supply header. The nominal values of the primary system parameters are provided in Table 3.

Table 3: AMPS 1000 Core Design Parameters

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<tr>
<th>Parameter</th>
<th>Value</th>
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<tr>
<td>Fuel Type</td>
<td>U-ZrH.e</td>
</tr>
<tr>
<td>Uranium Loading</td>
<td>45 wt%</td>
</tr>
<tr>
<td>Enrichment</td>
<td>19.7 wt%</td>
</tr>
<tr>
<td>Erbium Loading</td>
<td>2.0 wt%</td>
</tr>
<tr>
<td>U-235 Mass</td>
<td>69 kg</td>
</tr>
<tr>
<td>Fuel Rod O.D.</td>
<td>1.27 cm</td>
</tr>
<tr>
<td>Cladding</td>
<td>Incoloy 800</td>
</tr>
</tbody>
</table>

Core Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal Power</td>
<td>10.8 MW</td>
</tr>
<tr>
<td>Burnup Lifetime</td>
<td>1400 FPD</td>
</tr>
<tr>
<td>Fuelled Region Effective Dia.</td>
<td>83 cm</td>
</tr>
<tr>
<td>Fuelled Region Height</td>
<td>56 cm</td>
</tr>
<tr>
<td>Heat Flux – Average</td>
<td>35 W/cm²</td>
</tr>
<tr>
<td>– Peak</td>
<td>60 W/cm²</td>
</tr>
<tr>
<td>Minimum DNBR</td>
<td>5.4</td>
</tr>
<tr>
<td>Fuel Temperature – Average</td>
<td>280 °C</td>
</tr>
<tr>
<td>– Peak C/L</td>
<td>391 °C</td>
</tr>
<tr>
<td>Core Power Density</td>
<td>44 kW/1</td>
</tr>
<tr>
<td>Reactivity Coefficients</td>
<td></td>
</tr>
<tr>
<td>– Coolant Void (density)</td>
<td>-2.2 mk/kg void</td>
</tr>
<tr>
<td>– Coolant Temperature</td>
<td>-0.07 mk/°C</td>
</tr>
<tr>
<td>– Fuel Temperature</td>
<td>-0.07 mk/°C</td>
</tr>
</tbody>
</table>

System pressure is maintained by a single pressurizer connected to the pump supply header. In order to simplify the management of primary coolant inventory, the pressurizer is sized to accommodate all system volumetric expansion occurring during the transition from zero power cold to full power hot operating conditions. The vapour space contains steam at saturated conditions. System pressure is controlled...

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by a system of pressurizer heaters to counteract pressure reductions and a pressurizer spray to counteract pressure increases. A simplified flow diagram is provided in Figure 4. Typical of most water-cooled reactors, the auxiliary systems associated with the PHT system include a water chemistry and sampling system, an overpressure protection system, a primary coolant inventory makeup system and an off-gas management system.

Passive Auxiliary Cooling As indicated in Figure 3 the reactor core is supported within a reactor vessel which in turn is mounted inside a reserve coolant tank (RCT) containing a large mass of cool water normally at a temperature of 60°C. All reserve coolant tank penetrations are located near the top of the vessel to guard against the draining of reserve coolant and uncovering the core following a pipe break. The segregation of hot circulating primary coolant, at an average temperature of 198°C, from the cool reserve water (except under shutdown or accident conditions) is a key element of the AMPS 1000 design. This segregation is accomplished by means of hydrodynamic ports (HDPs) located in the core inlet and outlet piping, above and below the reactor core and open to the reserve tank. These devices are entirely passive, employ no moving parts and their operation requires no electrical power or operator intervention. In a PHT system designed for constant flow over the entire load range, they can maintain effective segregation (i.e., minimize flow exchange and consequential thermal losses) during normal operation over the entire range of design conditions.

Under conditions of flow reduction (loss of flow accident) or system temperature increase (loss of heat sink, loss of reactor power regulation), cool reserve water enters the PHT system via the lower HDPs, is passively driven through the core by net thermosyphon head and exits into the RCT via the upper HDPs. In the event of a loss of coolant accident, the HDPs permit the core to be reflooded with cool reserve water. The HDPs are also used for normal post-shutdown decay heat removal.

Reserve Tank Cooling During normal operation at power, heat is deposited in the reserve coolant due to small amounts of inadvertent exchange flow through the hydrodynamic ports. In addition, heat is conducted into the reserve coolant from the reactor vessel and piping inside the RCT. Following reactor shutdown, the PHT pumps are switched off and the decay heat is removed by natural convection through the HDPs into the RCT. In both cases, the ultimate dissipation of this heat to the environment is brought about by the RCT cooling system operating in either an active or passive mode.

In the active mode, the RCT cooling system is designed to remove up to 300 kW and is intended for use mainly while the reactor is operating. It maintains the reserve coolant at a temperature of 60°C and removes heat by means of a heat exchanger immersed in the reserve coolant and with coolers located at the outlet of each HDP. The RCT heat exchanger conveys heat to a fresh water cooling loop which is in turn connected to a seawater heat exchanger and an associated seawater loop independent of the secondary side. The HDP coolers are connected to the secondary side and are fed by the boiler feed pumps.

In the passive mode, the RCT cooling system is sized to provide continuous heat removal under normal post-shutdown or accident conditions. It will operate whether or not the PHT system is pressurized. The RCT heat exchanger is also used in this cooling mode. Although the fresh water loop is equipped with pumps to operate in the active mode as described above, this loop can operate completely passively. For this reason, the seawater heat exchanger has been positioned to provide sufficient static head to permit natural circulation in both the fresh water and seawater loops.

The factors that can limit the heat removal capability of passive cooling systems in the submarine environment and which have been carefully considered in the AMPS 1000 design include:

(i) limited availability of thermosyphon head within the restricted pressure hull diameters (typically 7.3 to 8.4 m) of candidate submarines.

(ii) dynamic boat motions which lead to variations in amplitude and direction of the g-force which drives natural convection loops.

(iii) the effect of design basis boat orientations on the allowable positioning of seawater inlet and outlet nozzles while the submarine is surfaced.

(iv) heat sink (seawater) temperatures for submarine operations in warmer climates which approach and can even exceed 30°C.
For conditions of extreme boat motion or high seawater temperature, where the passive mode of operation may experience a limited degradation in effectiveness, the active mode of RCT cooling may be preferred.

Shielding Primary shielding, to attenuate direct core radiation, is provided by an annulus of lead and concentric annuli of borated stainless steel plates, surrounding the reactor vessel within the reserve coolant tank, as shown in Figure 3. These, together with the reserve coolant water, provide sufficient shielding to permit access to the reactor compartment for inspection and maintenance when the reactor is shutdown, and for a limited time with the reactor at power. Secondary shielding takes the form of lead on the outside of the reactor compartment bulkheads and in the access tunnel traversing the reactor compartment. The secondary shielding further reduces radiation dose rates to acceptable levels outside the reactor compartment and inside the access tunnel. As indicated in Table 4, total shielding weight for the AMPS 1000 plant represents the major component of the total plant weight. Measures that are taken to optimize the design include: (i) positioning the reactor compartment within the submarine to maximize any collateral shielding benefit from fuel tanks and battery compartments, and (ii) permitting higher dose rates in adjacent unmanned or infrequently manned spaces based on projected occupancy factors.

**TABLE 4: TYPICAL WEIGHT FRACTIONS OF AMPS 1000 PLANT SYSTEMS**

<table>
<thead>
<tr>
<th>Component</th>
<th>Weight Fraction</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Heat Source</td>
<td>20%</td>
</tr>
<tr>
<td>Energy Conversion Unit</td>
<td>14%</td>
</tr>
<tr>
<td>Shielding</td>
<td>61%</td>
</tr>
<tr>
<td>Electrical and Control</td>
<td>2%</td>
</tr>
<tr>
<td>AMPS Outfit</td>
<td>3%</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td><strong>100%</strong></td>
</tr>
</tbody>
</table>

*Exclusive of structure, supports, hull extension and submarine modifications, all of which are dependent on the specific submarine under consideration.

**Energy Conversion Plant**

The conversion of the relatively low grade heat to electrical energy is achieved in the AMPS 1000 by means of a subcritical saturated steam Rankine cycle. While AMPS plant designs with lower primary coolant temperatures employ organic working fluids to achieve reasonable thermal efficiency, component sizes and cost, the primary system temperatures in the AMPS 1000 plant make viable the use of water as the working fluid.

The AMPS plant secondary circuit consists of two parallel loops, each consisting of one recirculating steam generator, two turbine alternators, one seawater cooled condenser and a boiler feed system. The two loops are cross-connected to enhance production reliability in the event of component failures. Every for certain combinations of multiple component failures, part load operation will be possible. The boiler feed system provides additional process functions such as alternator cooling, turbine lubrication, feed pump lubrication and other minor cooling functions. Figure 4 shows the key components of the cycle.

Contrasting with conventional steam plants, the secondary circuit in the AMPS 1000 has been designed as a completely closed system. In particular, design measures have been adopted to prevent ingress of air or release of water from the plant. These measures include elimination of dynamic seals at the low pressure end of the plant, the provision of high-integrity static seals throughout the plant, the use of canned boiler feed pumps and virtually all-welded construction. Moreover, the use of integral, totally-enclosed direct drive turbine/alternators with hydrodynamic bearings and process fluid cooling have eliminated the need for auxiliary lubrication and cooling systems, external shaft seal systems and gear-boxes. It is envisaged that, during a mission, there will be no need to purge the condenser of non-condensible gases, or provide makeup.

As indicated in Table 5, a net plant output of 1700 kWe is provided at seawater temperatures between 0 and 15°C. This represents an overall plant net conversion efficiency of approximately 16%.

**TABLE 5: AMPS 1000 PLANT RATING**

<table>
<thead>
<tr>
<th>Component</th>
<th>Rating</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Thermal Power</td>
<td>10.8 MW</td>
</tr>
<tr>
<td>Thermal Losses to RCT (Average)</td>
<td>0.2 MW</td>
</tr>
<tr>
<td>Plant Aux. Loads &amp; Conversion Losses</td>
<td>156 kWe</td>
</tr>
<tr>
<td>Net Power to Batteries</td>
<td></td>
</tr>
<tr>
<td>• 0 - 15°C Seawater Temperature</td>
<td>1700 kWe</td>
</tr>
<tr>
<td>• 30°C Seawater Temperature</td>
<td>1432 kWe</td>
</tr>
<tr>
<td>Net Efficiency @ Nominal Design Cond.</td>
<td>15.7%</td>
</tr>
</tbody>
</table>

**Configuration**

Considerable attention has been paid to the compactness, weight and configuration of components in order to ensure that they can be adequately supported and suitably isolated for shock and noise propagation. A typical integrated arrangement for the reactor compartment of a naval AMPS 1000 plant in a typical 2000-tonne class submarine is depicted in Figure 5. The key features of this arrangement include a primary containment boundary enclosing all AMPS primary systems and a 1-metre wide shielded tunnel which permits crew transits through the reactor compartment. The primary containment boundary comprises the two bulkheads located at either end of the reactor compartment and the walls of the access tunnel. The containment boundary, together with the PHT system pressure boundary and the fuel cladding, comprise the three physical barriers to radioactivity release. In order to provide shock protection, critical components such as the reserve coolant tank are mounted on a single raft-type support structure. This structure is supported by the pressure hull from the sides and is isolated by a suitably designed mounting system.

The more conventional components can be shock qualified using standard design, fabrication and installation techniques. In the application of the AMPS 1000 plant to a typical 2000-tonne class diesel-electric submarine, the entire AMPS reactor and secondary plant can be accommodated in a pressure hull extension of less than 10 m. This represents a less
than 15% increase in overall boat length and results in only a marginal reduction in maximum speed of the submarine. Careful attention is paid to the overall weight of the hull extension (maintaining nearly neutral buoyancy), and to the distribution of weight in the submarine after integration of the AMPS. This ensures that submarine stability, manoeuvrability and reserve buoyancy margin are not compromised.

Operational Considerations

Load Following The large amount of energy stored in the batteries of a diesel-electric submarine effectively decouples the AMPS from the need to precisely track rapid and large changes in propulsion power demand. Power level changes can therefore be effected at a rate which ensures stable and orderly changes in process conditions over the entire load range. The large prompt negative temperature coefficient, a characteristic of the AMPS core, ensures that, over a very wide range of power level, power will vary linearly with reactivity change. Operational reactivity control is therefore very simple and easily automated; the reactor responds readily, and in a stable and controlled manner, to control rod movement.

Due to the very low core power density in the AMPS-1000, xenon transients following major power reductions do not pose an operational limitation. The maximum increase in xenon load following a reactor shutdown from full power is less than 1 mk. This represents a negligible increase in the core excess reactivity requirement and can be readily accommodated by regulating rod movement alone.

Warmup The AMPS 1000 is designed to permit a rapid warmup to full power hot from zero power cold operating conditions. Assuming a desired PHT system warmup rate of approximately 2°C/minute, warmup can be undertaken with the reactor at a power level of approximately 20% of full power. The plant can attain full power hot operating conditions within 90 minutes from cold shutdown. Less time is required if the system is not cold prior to beginning the automated startup sequence. The rapid warmup feature has advantages under...
circumstances where it is desirable to temporarily shutdown all plant systems in the submarine to minimize the likelihood of acoustic detection.

**Operator** The highly automated AMPS 1000 plant is controlled by a dedicated control and monitoring system, featuring on-line control, system monitoring, data logging and reporting functions. The control panels are designed so that only one operator is required to run the plant at any given time. All routine plant operations are fully automated. More importantly, once the reactor has been shut down no operator actions are required to ensure ultimate plant safety following an accident.

**Thermal Ballast of Reserve Coolant** The reserve coolant, which is normally maintained at a temperature of 60°C, represents a large thermal ballast which can store post-shutdown decay heat production in the event that reserve tank cooling is temporarily interrupted. Following shutdown, reserve coolant temperature can routinely be permitted to increase to almost 100°C. If RCT cooling is interrupted immediately after shutdown, more than 10 hours is available before the external heat sink has to be re-established. Under accident conditions, temperature can be permitted to rise to the coolant saturation temperature of 246°C. Under these conditions, at least 40 hours is available. This feature clearly provides significant safety benefits. It can also be applied in situations where the plant operator is seeking to temporarily reduce the thermal signature (and therefore the detectability) of the submarine, by temporarily eliminating heat rejection into the sea.

**SAFETY FEATURES**

One of the key elements of the safety design philosophy of the AMPS 1000 is the incorporation of principles of process-inherent safety. Safety features designed to minimize the likelihood of catastrophic release of radioactive materials to the environment include:

- A large prompt negative temperature coefficient which can compensate for accidental reactivity changes. This ensures that the reactor power will safely and automatically self-limit or be shutdown in response to reactivity insertions and cooling transients which would otherwise threaten fuel cladding integrity.

- Retention of fission products in the U-Zr-H fuel to a much greater degree than in reactors using alternative fuel types. Figure 6 presents experimental results which characterize this feature. Since the operating fuel temperatures are below 400°C, then considerably less than a 10⁻⁴ fraction of the fission product inventory is available for release following random or accident-induced failure of the cladding.

- A passive cooling system which ensures immediate and unconditional availability of a large mass of cool reserve water to the core under accident conditions and provides a vehicle for passively dissipating reactor heat to the ultimate heat sink.

A detailed qualitative review of the response of AMPS-type power plants to a range of accident scenarios is provided in Reference 1. More recently, a preliminary thermohydraulic analysis was performed which highlights the role and interaction of the unique neutronic and thermohydraulic features of the AMPS 1000 design in assuring safe plant response in the absence of intervention by engineered safety systems or the operator. Four specific cases are considered to represent the four generic accident categories to be considered. These are:

(i) Loss of Regulation (LOR). Ramp insertion of positive reactivity at a rate of 9 mk/s starting at a steady 100 % full power.

(ii) Loss of Coolant Flow (LOCF). Trip of both primary heat transport pumps starting at a steady 100 % full power.

(iii) Loss of Heat Sink (LOHS). Complete loss of secondary side heat removal starting at a steady 100 % full power.

(iv) Loss of Coolant Accident (LOCA). Guillotine break in a 3.8 cm diameter reactor piping outlet line starting at a steady 100 % full power.

These simulations were performed with the RELAP-5 (mod 1) systems simulation code [4]. Transient fuel cooling studies were performed using the COBRA-IV system simulation code.
Loss of Coolant Flow (LOCF). As indicated in Figure 8, following failure of the primary pumps at full power, exchange (ingress) flow driven by natural circulation is established immediately between the core and the reserve coolant tank via the hydrodynamic ports. Although core flow decreases to about 10% of its initial value, the effective core inlet temperature pulse which is terminated through a rapid increase in fuel temperature as shown in the results given in Figure 7. The limited installed excess reactivity of the AMPS 1000 core limits the quasi-steady state power level following the initial pulse to less than 200% of full power. Due to the low core power density and large thermal-hydraulic design margins at full power, this elevated power level is one at which the fuel is seen to remain adequately cooled (Minimum DNBR > 1) for some time. Hot element peak fuel temperature in this scenario remains well below the derived 950°C safety limit at which cladding failure could potentially occur.

Loss of Coolant Flow (LOCF) As indicated in Figure 8, following failure of the primary pumps at full power, exchange (ingress) flow driven by natural circulation is established immediately between the core and the reserve coolant tank via the hydrodynamic ports. Although core flow decreases to about 10% of its initial value, the effective core inlet temperature...
is now determined by the reserve coolant at 60°C. As a result, an adequate DNBR can be maintained and peak fuel temperature remains well below the safety limit. In the long term, the reserve coolant temperature will rise and reactor power will continually decrease until the heat generated in the reactor core matches that removed from the reserve coolant tank and ultimately deposited in the sea.

Loss of Heat Sink (LOHS) Following interruption of heat removal from the primary heat transport system with the reactor at full power, the primary coolant temperature will increase due to the mismatch between core heat production and heat removal from the PHT system. As indicated in Figure 9, this leads to a reduction in reactor power and the initiation of exchange flow with the reserve coolant tank which enhances through-core flows already being supported by the PHT pumps. Within a few minutes, the reactor is nearly at decay power levels, commensurate with the level of heat rejection to the reserve coolant tank via the hydrodynamic ports. The fuel remains well-cooled throughout the transient.

Loss of Coolant Accident (LOCA) Following a guillotine break in a 3.8 cm outlet line in the PHT system, system depressurization occurs and voiding occurs in the core due to flashing of the hot primary coolant. As indicated in Figure 10, this leads to an initial drop in reactor power. In this simulation, a reactor trip is assumed to occur within 2 seconds of the break initiation. The core is reflooded by cool reserve water entering the core region via the hydrodynamic ports. Fuel temperatures remain below the safety limit for the entire transient. The total mass of PHT inventory lost through the break during the blowdown transient is small relative to the mass of reserve coolant. Sufficient water remains within the reserve coolant tank to keep the upper hydrodynamic ports covered. This permits core decay heat to be removed via the normal shutdown heat removal method.

CONCLUDING REMARKS

The AMPS development program has been underway since 1985 and designs for a range of submarine applications are well advanced. Full scale testing of the key components of the AMPS technology at the 100 kWe level [6, 7] is providing a strong engineering and technology base to support the feasibility of the AMPS plants in the 700 - 1700 kWe range. Ongoing efforts to refine and optimize the AMPS 1000 design will contribute to the demonstration of safe, relatively simple and affordable nuclear technology for marine propulsion.

ACKNOWLEDGEMENT

The author would like to thank Messrs. A. Tahir and V. Tavassoli for their significant contribution to the performance and interpretation of the thermalhydraulics analysis reported herein, and to Mr. R. Gosling for valuable discussions on submarine integration issues.
FIGURE 10: TRANSIENT RESPONSE TO BREAK IN 3.8 CM LINE IN REACTOR OUTLET PIPING
(REACTOR TRIP AFTER 2 SECONDS)

REFERENCES


THE NUCLEAR BATTERY PROGRAM: PROGRESS AND FUTURE POSSIBILITIES

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Whiteshell Nuclear Research Establishment
Pinawa, Manitoba ROE 1LO

ABSTRACT

The Nuclear Battery is an advanced small reactor being developed by Atomic Energy of Canada Limited (AECL) to produce electricity, and/or high-temperature steam heat, in locations remote from utility grids or natural gas pipelines. It features a novel passive primary heat transport system based on liquid-metal heat pipes, extraordinary passive safety based on the use of coated-particle fuel, and burnable neutron poisons in a solid graphite moderator. The reference design is capable of producing about 600 kW of electricity or about 2400 kW of steam heat in a base-load mode for 15 full-power years without refuelling. This paper reviews the technical progress, present activities and future goals of the Nuclear Battery R&D support program. Also, some of the alternate design approaches to increase the thermal power output from the Nuclear Battery and, hence, to further reduce its unit energy cost, are briefly outlined and compared.

1. BACKGROUND

The Nuclear Battery is a small nuclear power supply designed to generate electricity and/or high-grade steam heat in remote locations. It is being pursued by AECL as a complementary, follow-up product to the Slowpoke Energy System district heat source [1], but is at a much earlier stage of development.

AECL's Nuclear Battery program originated in 1984 as a joint project with the Los Alamos National Laboratory (LANL) to develop a 20-kW(e) nuclear power supply, known as the Compact Nuclear Power Source (CNPS) [2], for unattended short-range radar stations in the new North Warning System (NWS). Subsequently, a scaled-up, 500-kW(e) version of the concept was considered as an air-independent auxiliary power source for hybrid diesel submarines, as part of the Canadian Submarine Acquisition Project (CASAP) [3]. These special-user projects involved detailed and very demanding requirements that focussed and stimulated the design and assessment activities, and sustained the development of the underlying technologies for several years.

Now, however, the Nuclear Battery program has been redirected toward broad-based commercial applications that are in the long-term national interest, specifically the displacement of fossil fuels used to generate electricity or high-temperature steam heat in remote locations. The key ingredients to success in this endeavor and to securing commitment for the timely construction of a demonstration or prototype Nuclear Battery unit are: superior passive safety features, supported by analytical assessments; reliable technology, based on component performance data; and convincing economic comparisons, founded on reasonable assumptions. This paper reviews the status of the R&D support program activities and its present plans.

One of the perceived limitations to further evolution of the Nuclear Battery concept is the maximum heat removal capacity of its liquid-metal heat pipes. Consequently, alternate heat transport options, based on pumped reactor coolants that may permit higher power densities and reduced unit energy costs, are compared.

2. REACTOR CORE DESCRIPTION

The basic features of the Nuclear Battery reactor core are shown schematically in Figure 1. Only a brief overview is given here, since the design and safety features have been described elsewhere [4] and have not changed appreciably in the past year.

Fuel for the Nuclear Battery is based on 19.75-wt% 235U-enriched TRISO (triply isotropic) coated particles, consisting of tiny spherical kernels of UO2 protected and contained by applied layers of pyrolytic carbon and SiC ceramic that have been amply demonstrated in high-temperature reactors. The TRISO coated fuel particles are mixed with a graphite matrix binder and formed into cylindrical compacts. Fuel rods consist of vertical stacks of the compacts and are embedded in solid blocks of graphite moderator. Overall, the cylindrical reactor core is approximately 2.5 m in diameter and about 2 m high, including the unfuelled reflector regions. The fissile inventory required to sustain operation at 2400 kW(t), at a nominal core graphite temperature of about 575°C for 15 full-power years, is about 35 kg of 235U.
Heat passes by conduction from the fuel rods through the graphite moderator to the 159 heat pipes regularly dispersed throughout the fuel lattice. The heat pipes are sealed metal tubes, about 5 cm in diameter and 3 m long, with a porous wire structure regularly dispersed throughout the fuel lattice. The hundred grams of potassium working fluid. Heat transport occurs in a passive manner by evaporation of potassium in the in-core, or evaporator, section of the pipe and condensation at the top, or condenser, section that protrudes above the reactor core.

Active reactivity control is provided by four axial control/shutdown rods. The reactivity control requirements of the rods are about 43 mk from cold to hot, 4.4 mk for equilibrium xenon and a maximum of about 7 mk (near mid-life) for long-term fuel depletion, for a total of about 54 mk. Passive long-term reactivity control and flattening of the radial power profile are provided by burnable neutron poisons inserted in holes in the graphite moderator between the fuel rods in the inner core region.

The graphite core is surrounded by a few centimetres of thermal insulation and rests on supports within a steel containment vessel. A helium cover gas system provides an inert core environment, promotes heat transfer across clearance gaps, and maintains a modest absolute pressure of about 110 kPa (about 1.1 atmospheres) inside the vessel to minimize the potential for air ingress.

Useful energy is extracted from the Nuclear Battery by the circulation of an organic coolant, such as toluene, in coiled-tube vaporizers surrounding the condenser portion of the heat pipes. The vaporizer housing forms an extension to the core containment vessel and is bolted to it at a flanged joint, which has a metal ring seal. A narrow, helium-filled gap between the heat pipe wall and the vaporizer housing serves to reduce the peak outlet temperature experienced by the toluene to about 400°C.

To produce electricity, supercritical toluene vapour is collected from the parallel vaporizer streams and fed to the turbine inlet of an organic Rankine cycle engine. Expansion of the vapour through the single-stage high-speed turbine converts a portion of the thermal energy into rotary mechanical motion to turn an AC generator and a Pitot feed pump. Exhaust vapour from the turbine is passed through a waste heat regenerator and then liquefied in a water-cooled condenser. The Pitot feed pump draws liquid toluene from the condenser and forces it through the regenerator, where it is preheated to about 225°C. The output from the regenerator is then distributed to the individual vaporizers to complete the cycle. The net conversion efficiency is estimated to be about 25% for a 2400-kW(t) Nuclear Battery.

Because the Nuclear Battery is a high-temperature heat source, the same reactor core module can be coupled through a pumped organic-fluid secondary heat transport system to a steam generator. In this case, the choice of heat transport fluid is not constrained by the thermodynamic requirements of a Rankine cycle engine, and other working fluids might be considered in addition to toluene. Also, a steam generator would be less expensive than an organic Rankine cycle engine conversion system.

3. R&D PROGRESS

At the Whiteshell Nuclear Research Establishment (WNRE), development of the Nuclear Battery concept has concentrated on three main areas: core neutronics design and analysis, heat pipe development and organic fluid stability. Recent progress in these activities is described below.

3.1 Core Neutronics Design and Analysis

A first iteration of the core neutronics design has been completed [5,6] in sufficient depth to permit preliminary cost estimates and safety assessments. The majority of this effort was directed toward minimizing the required fuel inventory, optimizing the use of burnable poisons and investigating the safety features of the design.

One area of particular importance to the licensing of the Nuclear Battery for unattended operation for prolonged periods is its response to postulated, large, positive reactivity insertions, such as might be induced by the rapid withdrawal of a control rod. This accident scenario is only possible for the Nuclear Battery during cold reactor startup, because the burnable neutron poisons limit the maximum available excess reactivity to about 7 mk, once the reactor has reached its hot, full-power, equilibrium-xenon operating state.

Recent work [7] has shown that the Nuclear Battery is remarkably resilient to large reactivity excursions. Indeed, it was determined that no damage to the TRISO fuel particles would occurs for insertions of about 22 mk, whereas the maximum that is available from the withdrawal of a single control rod is about 19 mk.

The calculated temperature responses of an average TRISO fuel particle and the moderator graphite for the Nuclear Battery to an instantaneous reactivity insertion of 20 mk from a cold, clean, critical initial core state are shown in Figure 2. For this super-prompt-critical transient, reactor power increases very rapidly, reaching a maximum of about 4500 MW(t) in less than 0.3 s.

![FIGURE 2: NUCLEAR BATTERY TEMPERATURE RESPONSE TO INSTANTANEOUS 20 mk ADDITION](image-url)
there is little time to transfer heat by conduction through the particle coating layers and matrix binder to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating there is little time to transfer heat by conduction to the graphite moderator. Thus, it is only the negative reactivity feedback imparted by the UO₂ fuel Mo release of fission products would occur because of about 800°C and the temperature at its SiC coating.

Notably, Figure 2 shows that the temperature at the centre of a typical fuel kernel reaches a maximum of about 800°C and the temperature at its SiC coating layer peaks at about 225°C. As a result of the power form factor for the reactor core, the maximum values for a single fuel particle are highest: about 1600°C for the fuel kernel and 425°C for the SiC coating. No release of fission products would occur because the TRISO particle is designed to withstand a temperature of 1600°C for more than 100 h without failure of its coating layers.

After the rise in fuel temperature has terminated the power transient, heat transfer to the moderator causes the fuel and graphite temperatures to equilibrate after about 1.5 s. Negative reactivity feedback from the rising moderator temperature compensates for the declining fuel temperature; thus, the reactor power will eventually stabilize at a value that balances the available means of heat removal.

3.2 Heat Pipe Development

The potassium heat pipes are the most novel technical feature of the Nuclear Battery, and they represent the critical technology requiring the most development. The heat pipe development program at WNRE has made steady progress, mainly using subscale, stainless-steel/potassium heat pipes for reasons of cost and expediency. A total of twelve potassium heat pipes have been built and tested at WNRE (seven within the last year) since the first was tested in the spring of 1986.

The first priority of the Nuclear Battery heat pipe program is to maximize the axial heat transport capability through the design of improved wick structures. The main consequence of a major improvement of the axial heat flux would be a reduction in the diameter or number of heat pipes, with a significant saving in the core fissile inventory, or, alternatively, an increase in total power output accompanied by reduced unit energy costs.

Heat pipe wick designs have evolved and improved substantially during the course of the Nuclear Battery program. The wick design for the CNPS heat pipes was a knurled interior surface and it eventually achieved a maximum axial heat flux of about 0.8 kW/cm² at the reference condenser entrance temperature of 500°C. Early work at WNRE with screen-mesh wick structures supported on coarse wire grids improved the maximum performance to about 1.0 kW/cm². Recent work using a corrugated screen mesh with three distinct layers has demonstrated a maximum axial heat flux of 1.1 kW/cm² at 500°C, well above the design requirement of about 0.9 kW/cm².

The temperature dependence of the axial heat flux data for this heat pipe is shown in Figure 3.

Figure 3 shows that heat pipe performance is presently constrained by the fundamental sonic limit, where the vapour velocity reaches the speed of sound and chokes at the entrance to the condenser, up to about 435°C. Above this temperature, performance is limited by liquid entrainment, where the upward vapour flow strips returning liquid from the wall of the evaporator.

Further performance improvement is possible through improved wick design. For example, early work at LANL by Kemme [8], using a sintered screen-mesh wick, achieved operation at the sonic limit up to about 485°C at the entrance to the condenser, corresponding to an axial heat flux of about 2.5 kW/cm².

3.3 Organic Fluid Stability

The use of organic compounds as reactor heat transport fluids is an area of unique technical expertise within AECL, founded primarily on the successful operation of the WR-1 organic-cooled research reactor at WNRE for 19.5 years. This expertise has been adapted to the study of Nuclear Battery organic Rankine cycle engine working fluids, as the working fluid will experience high temperatures and high-radiation fields at the vaporizers above the reactor core. The importance of this work is that the thermodynamic cycle conversion efficiency is dependent on the peak cycle temperature, and the rate of fluid degradation will determine fluid inventory lifetime, hence, the required maintenance interval.

Pyrolytic degradation tests have been performed for toluene and other candidate Rankine cycle working fluids in a comprehensive series of static-capsule and flowing-loop tests, over a range of temperatures up to a maximum of 470°C, using different vessel materials and surface area/volume ratios. The data indicate that operation at maximum toluene temperatures of up to 400°C at the vaporizers is feasible with fluid inventory replacement approximately every five years.

A similar series of gamma radiolysis experiments has been performed using static capsules, and flowing-loop tests at supercritical fluid conditions are in progress. Thus far, the data indicate that gamma radiolysis should not be a major problem for the dose rates estimated to be present at the Nuclear Battery vaporizers. Plans are being made to complete the database with neutron radiolysis tests, and an experimental irradiation rig has been designed for insertion into a Slowpoke-2 research reactor.
A small, 1-kW(e), demonstration toluene organic Rankine cycle engine has been operated for a total of about 2400 h, including 650 h in the longest, single continuous run. This engine has been converted in the past year from its original propane-fired configuration to an electrically heated mode that should provide a more delivery heat source and permit continuous measurement of the net conversion efficiency.

4. PRESENT PLANS

4.1 Integrated Heat Pipe Test Facility

The next logical step of the R&D support program is to demonstrate the overall technical feasibility of the Nuclear Battery concept in a non-nuclear, but otherwise comprehensive, test. Thus, plans are underway to build an Integrated Heat Pipe Test Facility that will use a full-scale Nuclear Battery heat pipe in graphite with electric resistance heaters to simulate the nuclear fuel rods. Also, the apparatus will be enclosed in an insulated, steel containment vessel with a helium cover gas system.

It is anticipated that the experiment will proceed in stages, initially with a Sytltherm secondary heat transport loop acting as a calorimeter, and subsequently with a toluene vaporizer loop. Eventually, it will be connected to the 1-kW(e) demonstration toluene Rankine cycle engine and possibly to a small steam generator.

The graphite blocks for this experiment have been ordered and siting preparations have begun. We are hoping to acquire a large vacuum glove box in the near future to facilitate the filling and sealing of the full-scale heat pipes in a controlled, contamination-free environment.

4.2 Materials Compatibility Tests

One of the most difficult issues to address in our R&D program is the question of reliable heat pipe operation for 15 full-power years without performance degradation. Therefore, we have initiated a series of long-term materials compatibility tests consisting of small heat-pipe capsules with refluxing potassium and various coupon samples to simulate wick structures.

An important issue to be resolved in the materials compatibility tests is the choice of heat-pipe wall material. At present, the niobium alloy, Nb-1 at% Zr, has been selected as the reference, because it is a reasonable compromise in physical and neutronic properties between stainless steel and zirconium alloys. The zirconium alloy, Zircaloy-4, is retained as the backup and would be much preferred on the basis of neutronic considerations.

4.3 Competitive Position Review

We are also preparing the documentation and assembling cost estimate data for a major review of the Nuclear Battery program that is expected to take place within the next year. This review will encompass technical, cost estimate and market considerations. After proceeding through normal internal review procedures, external participation from utilities and universities will be sought for a more formal review process. The program review is seen as a necessary step toward increasing the level of commitment to the program and establishing a reasonable timetable for a full-scale nuclear demonstration unit.

One of the major outcomes of the review is expected to be a clearer definition of the economic arguments for deployment of a Nuclear Battery. Presently, more detailed cost estimates support a unit energy cost of about 20 cents/kWh for electricity from a 600-kW(e) unit operating at 955 capacity factor for 15 full-power years, which basically confirms previous preliminary estimates [4]. However, the range between best- and worst-case estimates is broad, from 13 to 30 cents/kWh; thus, more work is required to reduce uncertainties.

5. ADVANCED REACTOR CONCEPTS

Traditional economies of scale suggest that the unit energy cost for the Nuclear Battery could be reduced substantially if the power output could be increased, ideally without significantly degrading its safety and reliability features. However, even if heat pipe performance was improved to approach the sonic limit at 500°C, a maximum power of only 6 to 7 MW(t) would result. Further increases in power output would be obtainable only by increasing the core diameter and adding more heat pipes or, perhaps by increasing the core average temperature. Alternative technical approaches to heat transport, involving pumped reactor coolants, begin to be attractive at higher power densities.

Our core neutronics work suggests that reactor operation for about 75 MW, or roughly double the energy output of the reference Nuclear Battery, may be attainable without refuelling, but with an increased fissile loading. Thus, a 7.5 MW(t) reactor core would last for about 10 full-power years before replacement. The main technical issue to be addressed would be reoptimization of the burnable poison loading scheme. Also, the equilibrium-xenon poison reactivity load would increase in approximate proportion to the power density.

5.1 Helium-Cooled Reactor

One straightforward approach to increased power density is to eliminate the heat pipes entirely, pressurize the helium cover gas, and circulate it through core coolant channels to an external helium/toluene heat exchanger. Examples of small helium-cooled reactors built and operated in other countries include: Dragon in the United Kingdom (20 MW(t)), AVR (Arbeitsgemeinschaft Versuchskernreaktor) in the Federal Republic of Germany (46 MW(t)), and Peach Bottom I in the U.S. (115 MW(t)).

Compared with the Nuclear Battery, one of the main technical advantages of a helium-cooled system is the elimination of neutron absorption losses to the heat pipe and potassium working fluid materials. Helium has a very low thermal neutron absorption cross section of only 7 mb, so that reactivity effects associated with changes in density and temperature would be very small. Also, replacement of the numerous heat pipe vaporizers with a single tube-in-shell-type helium heat exchanger/toluene vaporizer would simplify plumbing arrangements above the core.
The main disadvantage with helium cooling is that the core would have to be pressurized to at least 1.0 MPa (about 10 atmospheres) to achieve reasonable rates of heat transport. Also, the replacement of passive primary heat transport with active coolant circulation by helium blowers, which may be subject to common-mode failure, is an important qualitative distinction. A practical consequence of active pumping is that the operation of helium circulators requires input power, representing perhaps a few percent of the electrical output.

A rough figure of merit for comparing the efficacy of reactor heat transport technologies, in the present context, is the ratio of the total core power to the cross-sectional area occupied by primary system tubing or coolant holes (i.e., the average axial heat flux). For the heat pipes in the reference Nuclear Battery, this ratio is about 0.7 kW/cm² (including tube and wall thicknesses). In preliminary calculations for a helium-cooled, 4.25-HW(t) core of similar dimensions to the Nuclear Battery operating at 1.0 MPa and a mass flow rate of 4.7 kg/s, it was found that this figure of merit doubled to about 1.5 kW/cm². However, the average graphite temperature at the surface of the coolant holes would need to be increased to about 620°C to achieve an outlet gas temperature of 500°C with an inlet gas temperature of 381°C.

5.2 $^7$Li-Cooled Reactor

Pumped liquid-metal coolant systems are generally used wherever high power densities are required at high temperatures. Key technical advantages of liquid-metal systems include: low operating pressure, high thermal conductivity, good heat capacity and low viscosity. Also, because liquid metals conduct electricity, they can be circulated using silent, non-intrusive, electromagnetic (EM) pumps having no moving parts.

The main disadvantages of liquid metals include their potential for corrosive attack on materials and their chemical reactivity in air or water. Also, the startup of liquid-metal heat transport systems from the frozen state needs to be addressed, for example, through the use of trace heating systems outside the reactor core and a volume expansion tank.

For thermal reactor systems, the best liquid metal from a neutronic viewpoint is the isotope of lithium, $^7$Li. This has a thermal neutron absorption cross section of only 37 mb. However, $^7$Li, which has an atomic abundance of 92.5% in natural lithium, must be isotopically separated from $^6$Li (thermal neutron absorption cross section of 940 b) to a high degree of purity.

Notably, the coolant temperature and void reactivity effects of $^7$Li in this graphite-moderated thermal reactor are small and negative, because the presence of $^7$Li in the core is slightly beneficial to the core neutronics ($^7$Li provides a modest amount of neutron moderation). Also, $^7$Li does not become actinide like sodium and potassium with the delayed production of high-energy gamma rays. Therefore, radiation-shielding requirements for the heat transport piping outside the reactor should be minor.

We have performed preliminary heat transport calculations for an up-rated, 15-MW(t) Nuclear Battery based on the forced circulation of $^7$Li liquid-metal coolant through small-diameter, metallic, in-core coolant tubes. A schematic arrangement of the primary coolant loop is shown in Figure 4. An important neutronic advantage relative to heat-pipe- or helium-cooled systems is that the liquid-metal coolant tubes are small in diameter and filled with a neutron-scattering medium, so that axial neutron leakage losses are reduced.

The total $^7$Li inventory required for this concept is estimated to be about 234 kg, of which only 38 kg is in the reactor core. Liquid $^7$Li enters the core at about 325°C and at a nominal pressure of about 110 kPa and leaves at 500°C. The theoretical pumping power is estimated to be about 560 W(e), although EM pumps generally have low efficiencies of perhaps 20%. Notably, roughly 3 HW(t) could be transported by natural circulation if the height, H, in Figure 4 is 6 m.

For a 15-MW(t) reactor, the average fuel-rod centreline temperature would be about 815°C, and presents no problems. Also, the average core thermal power density is about 3.5 MW/m², which is still well below the value of 6.3 MW/m² demonstrated in the Ft. St. Vrain reactor or the value of 6.0 MW/m² demonstrated in the THTR (Thorium Hochtemperatur Rektor). The maximum thermal stress in the graphite, however, would be about 8.4 MPa and roughly matches the across-the-grain cracking strength. Because the coolant tubes provide a separation boundary, radial cracks in the graphite near the coolant tubes might be of no consequence.

The average axial heat flux of the heat transport system for this concept is about 29 kW/cm² at the 15-MW(t) power level.

6. CONCLUSION

The Nuclear Battery concept has evolved toward a flexible unit directed at broad-based commercial applications for base-load power in remote regions. Steady technical progress continues to be made in the development of the hardware components necessary to demonstrate the technical feasibility of the concept, such as the heat pipes and vaporizers. Further development and assessment is planned before a commitment will be sought for the construction of a full-scale nuclear demonstration unit. The Nuclear Battery program has also stimulated the consideration of novel design variants, based on pumped reactor...
coolants, to address operation at higher power levels with reduced unit energy costs.

REFERENCES


Recent interest focussed on small reactors has indicated that safety criteria specific to these reactors are lacking. This paper summarizes recent work undertaken by the International Atomic Energy Agency (IAEA) and by an inter-organizational Small Reactor Criteria working group in Canada to develop comprehensive safety criteria needed for small reactors. The work is not yet complete, but is presented at this time to provide a progress report and to stimulate discussion.

2. IAEA RESEARCH REACTOR PUBLICATIONS

The International Atomic Energy Agency (IAEA) has developed an extensive series of safety publications in support of the nuclear power reactor programs referred to as the NUREG (Nuclear Regulatory Agency Guidelines for Nuclear Reactors) series. Until recently the primary IAEA publication on the safety of research reactors was Safety Series No.35, "Safe Operation of Research Reactors and Critical Assemblies - Code of Practice". This document was originally published by the IAEA in 1968 and has subsequently been revised in 1971 and most recently in 1984.
Safety Series No.35 is used extensively worldwide in the research reactor community in both the developed and developing countries. It primarily treats safe operation as opposed to other life cycle activities and as such has limitations in considering siting and design aspects of new research reactor facilities. Beginning in the mid 1980's, the Agency recognized that more detailed guidance was required on all aspects of research reactor safety and instituted a program to develop a wide spectrum of small reactor safety publications.

The principal reasons for the IAEA attention are as follows:

(1) The international move to low enriched uranium fuel, involving considerable international research and revisions of safety assessments in member states.

(2) Re-licensing activities in member states, involving updating of safety information.

(3) Upgrades to existing reactors, necessitating new safety submissions for licensing.

(4) Numerous proposals for installation of new research reactors in developing countries.

(5) The recognition of a lack of expertise in developing countries and the need for guidance in many areas.

A number of informal TECDOCS on small reactor safety topics have been published over the past several years, covering such topics as siting, probabilistic safety assessments, and reduced enrichment. Work was also initiated on several new guides dealing with research reactor safety analysis, reactor upgrades, and design. One safety guide on decommissioning has been completed. (1) In 1988, a Research Reactor Publications Advisory Group was formed to oversee the IAEA efforts in the development of these research reactor safety publications. The tasks of the group are to prepare a comprehensive listing of proposed documents, to outline the content and to oversee preparation of the documents for consistency.

The structure of the documentation, as recommended by the Advisory Group, follows the new IAEA practice consisting of four levels of primary, "Safety Series" documentation: Safety Fundamentals, Safety Standards, Safety Guides, and Safety Practices shown in Figure 1 and described below.

(1) Safety Fundamentals form the top level and present safety concepts, safety objectives, and fundamental principles. These documents are expected to cover a number of application areas are thus considered as umbrella documents.

(2) Safety Standards are expected to be the top safety documents specific to an individual application area such as research reactors. These documents concentrate on basic requirements and present mandatory requirements as part of the Agency's assistance activities.

(3) Safety Guides present recommendations to fulfill requirements or principles of higher level documents. They provide more specific guidance than the higher level documents.

(4) Safety Practices give practical examples and detailed methods on how to implement certain specific requirements.

Two other types of safety related publication are employed which are not part of the Safety Series. These cover other topics in addition to safety, but can have a close relation to the safety documents or have sections dealing with safety.

(1) Technical Reports contain technical information, often covering a wide technical area. They pursue a variety of purposes such as giving an overview of a technical process or presenting the state of the art in a particular field.

(2) TECDOCS cover technical aspects similar to Technical Reports. However publications in this series present tentative, preliminary or trial material.

FIGURE 1 ORGANIZATION OF IAEA SAFETY PUBLICATIONS

CNS 10th ANNUAL CONFERENCE, 1989, 2-25
The Advisory Group recommended the preparation of a single Safety Standard, nine Safety Guides, and a minimum of nine Safety Practices. The Safety Standard would be developed at the end of the process, replacing the existing Safety Series No.35 and covering all life cycle activities. The Safety Guides would include a broad range of topics, such as regulatory aspects, the safety analysis report, emergency planning, and all life cycle activities including siting, design, commissioning, operation, utilization and decommissioning. The initial Safety Practices would cover specific aspects of the Guides on safety analysis, design and operations. This is an ambitious program and is currently under review. At present, the Agency is committed to producing another six or seven standards and guides, covering design, several aspects of operations, radiation protection, emergency preparedness, and safety analysis reporting.

Two of the authors of this paper have been involved with the development of the existing Safety Standard, of several Safety Guides and with the Publications Advisory Group. The work of the IAEA in this area has been most useful. The Agency forum provides for international cooperation and is resulting in exchange of safety criteria, analysis methodology, and design and operational experience. This cross fertilization of information and experience is benefiting both developed and developing countries and has done much to enhance a small reactor safety culture.

3. CANADIAN SMALL REACTOR PROPOSALS

There are a number of small reactors already operating in Canada, others are in advanced stages of design or being built, and yet others are at the conceptual design stage. Small reactors which have been in operation for many years include the research reactors ZED-2, PTR, NRX and NRU at the Chalk River Nuclear Laboratories site of Atomic Energy of Canada Limited (AECL) and the 5 MW research reactor at McMaster University in Hamilton. Over the last 10 or 15 years, seven SLOWPOKE-2, 20 kW reactors have been built and operated in Canada.

Several new small reactor designs have been proposed in Canada over the last several years and some have reached a development stage where licensing discussions have been initiated with the Atomic Energy Control Board (AECB). These are summarized in Table 1 and described below.

3.1 SLOWPOKE Demonstration Reactor

The SLOWPOKE Demonstration Reactor (SDR) is a 2 MW demonstration and test reactor designed and built by AECL and operated since 1986 at its Whiteshell Nuclear Research Establishment (WNRE) laboratories in Manitoba. (2) The reactor is being used to provide design and operational information in support of the commercial heating reactor being developed by AECL; it is also intended to be used as a demonstration of building heating at WNRE. Information is being obtained in the areas of physics and thermohydraulics to validate computer codes and develop design features.

SDR is a pool type reactor, H2O cooled and moderated, utilizing low enrichment uranium dioxide fuel. The reactor is cooled by natural circulation with coolant temperatures below the boiling point, at atmospheric pressure. Two in-pool heat exchangers are used to extract heat to a secondary coolant system, which in turn can provide usable heat to a tertiary system. Reactivity control is provided by absorber plates. Safety features include negative reactivity coefficients of temperature and coolant void, natural reheat injection, a large pool (passive heat sink), a slow acting reactivity control system, and two shutdown systems (absorbers and poison injection).

The reactor has been licensed by the AECB and is currently undergoing power ascension tests.

3.2 SLOWPOKE Energy Systems Heating Reactor

The commercial SLOWPOKE Energy Systems heating reactor (SES-10) is a 10 MWt heating reactor under development by AECL. (3) Since it is not a high-power reactor, it is being designed to be located near its load and would be sited in urban areas. The reactor is also being designed so that skilled operator action is not needed for at least 24 hours after an abnormal event.

Like the SLOWPOKE Demonstration Reactor, SES-10 is a pool-type reactor design, cooled and moderated with H2O at atmospheric pressure, and utilizing low enrichment uranium dioxide fuel. The reactor consists of a vertical core near the bottom of a large pool, cooled by natural circulation. The hot water leaving the reactor is directed up a riser duct and into two parallel in-pool heat exchangers, which remove the heat to a closed-loop, forced-flow, low-pressure secondary cooling system. A third heat exchanger transfers heat from the secondary cooling system to a heat distribution system. The safety features of this reactor are similar to those discussed above for the SDR reactor. The pool is double-walled and provides a large heat sink; decay heat can be absorbed passively by the pool for extended periods of time. SES-10 is being designed with low fuel ratings and hence low operating fuel temperatures. Two diverse shutdown systems are provided.

The reactor is being marketed and is now the subject of a feasibility study and environmental assessment for a proposed site at the University of Saskatchewan in Saskatoon.

3.3 MAPLE-X10 Isotope Production Reactor

The MAPLE-X10 reactor is a 10 MW reactor being developed by AECL for siting at its Chalk River Nuclear Laboratories site (CRNL). The reactor is a variant of a high-flux family of multipurpose research reactors being marketed by AECL (4); the acronym MAPLE stands for Multipurpose Applied Physics Experimental. The reactor at CRNL is designed to be a dedicated radioisotope production facility.
TABLE 1 RECENT CANADIAN SMALL REACTOR INITIATIVES

<table>
<thead>
<tr>
<th>REACTOR</th>
<th>POWER</th>
<th>REACTOR TYPE</th>
<th>APPLICATION</th>
<th>STATUS</th>
</tr>
</thead>
<tbody>
<tr>
<td>SDR</td>
<td>2 MW</td>
<td>POOL-TYPE</td>
<td>TEST AND DEMONSTRATION</td>
<td>OPERATING: POWER ASCENSION TESTS</td>
</tr>
<tr>
<td>SES-10</td>
<td>10 MW</td>
<td>POOL-TYPE</td>
<td>LOCAL HEATING</td>
<td>MARKETING AND LICENSING DISCUSSIONS</td>
</tr>
<tr>
<td>MAPLE-X10</td>
<td>10 MW</td>
<td>OPEN TANK IN POOL</td>
<td>ISOPORE PRODUCTION</td>
<td>UNDERGOING LICENSING</td>
</tr>
<tr>
<td>AMPS</td>
<td>1.5 MW</td>
<td>PRESSURIZED VESSEL IN TANK</td>
<td>MARINE PROPULSION</td>
<td>DEVELOPMENT AND MARKETING ACTIVITIES</td>
</tr>
<tr>
<td>NUCLEAR BATTERY</td>
<td>2.4 MW</td>
<td>GRAPHITE MODERATED</td>
<td>ELECTRICITY AND HEAT</td>
<td>UNDER DEVELOPMENT</td>
</tr>
</tbody>
</table>

the isotopes being produced primarily for medical applications. As well, the reactor will serve as a demonstration of the generic MAPLE concept.

The MAPLE-X10 concept consists of an open tank-type reactor assembly within a light-water pool. The reactor is cooled by pumped H₂O, at essentially atmospheric pressure and at temperatures below the boiling point. It combines the high fast and thermal neutron fluxes of the H₂O cooled and moderated MTR-type reactor but also has a D₂O reflector, which provides both a reasonable flux and additional space for use in isotope production. The fuel is low enrichment uranium silicide in an aluminum matrix and reactivity control is provided by annular absorber rods. Safety features include negative reactivity coefficients of temperature and coolant void, decay heat removal by natural circulation and a large pool acting as a passive heat sink. Engineered safety features are provided by a rod-type absorber shutdown system and an emergency, filtered ventilation system routed to a high stack.

Reactor development and design work is well advanced. MAPLE-X10 is currently undergoing licensing by the AECB at the Construction Approval stage.

3.4 The AMPS Marine Reactor

The Autonomous Marine Power Source (AMPS) is a small-scale, nuclear electric power plant design initially conceived for submarine applications requiring a long-endurance air-independent supply of electric power. The reactor concept is being developed by a private firm, Energy Conversion Systems Power Systems Inc. (ECS) of Ottawa.

The AMPS design is a light-water cooled and moderated reactor heat source coupled to a low-temperature Rankine cycle engine. The design prototype has a thermal power of 1.5 MWt and generates 100 kW of electric power. The core consists of uranium-zirconium-hydride TRIGA fuel elements, with a beryllium reflector to flatten the flux and erbium burnable poison for long-term reactivity balance. The core is supported within a reactor vessel which in turn is mounted inside a reserve coolant tank containing a large mass of water. The coolant system is pressurized at about 3.7 MPa and operates at about 195°C average. Reactivity control is by means of neutron-absorbing rods. Safety features include the strong negative temperature feedback effect of the TRIGA-type fuel and passive shutdown cooling. If the primary pumps should fail, or the normal process heat sink is lost, decay heat can be removed passively to a reserve coolant tank and hence to the ultimate heat sink. There is a safety shutdown system consisting of spring-loaded absorbing rods.

Reactor design and development has been underway for about three years and potential markets are being pursued by ECS.

3.5 The Nuclear Battery

The Nuclear Battery is a small nuclear power supply design which can generate electricity and/or high-grade steam heat. The name derives from its solid-state, passive nature which distinguishes it from more conventional water cooled and moderated reactors. The concept is being developed by AECL for potential applications in the area of small-scale, baseload electricity generation in remote communities, and as a high-pressure steam heat source for industrial applications. The commercial scale unit could deliver either 600 kW of electricity or 2400 kW of steam heat for 15 full-power years.

The Nuclear Battery is graphite moderated, and cooled by heat-pipes using potassium as a working fluid. Its graphite core block acts as a heat storage well, maintained at about 550°C, from which energy is extracted passively by the heat pipes. The fuel is similar to that used for high-temperature gas-cooled reactors, consisting of TRISO (triply isotopic) coated low enrichment
uranium dioxide fuel particles formed into rods. These rods are inserted into holes in the graphite, and left there for the life of the system. Active reactivity control is provided by axial control/shutdown rods inserted from above the core; passive long-term reactivity control is provided by burnable neutron poisons inserted in holes in the graphite. Safety features include a negative temperature coefficient of reactivity and the ability of the TRISO fuel to withstand high temperatures (1600°C for up to 100 hours) without failure.

The basic physics characteristics of the reactor have been confirmed in full-scale critical experiments carried out by the Los Alamos Scientific Laboratory for a similar concept. Development work in Canada has focussed on the heat pipes and on a Rankine cycle energy conversion unit.

4. SMALL REACTOR CRITERIA DEVELOPMENT IN CANADA

Recent licensing discussions between the AECB and proponents of these various small reactor programs have indicated that safety criteria are needed. Apart from the 5 Megawatt McMaster reactor licensed in 1959, there has been little development in Canada until recently of small reactors with power levels in the range of several megawatts. Consequently, regulatory criteria appropriate to these small reactors have not yet been formulated. Criteria are gradually evolving by precedent for the specific reactors undergoing licensing, but there is a need for a unified approach to guide designers, operators, and regulators.

4.1 Background

In response to this need, an inter-organizational Small Reactor Criteria working group was formed in 1988 to develop criteria for small reactors. The group comprises the four authors of this paper, from the principal organizations in Canada responsible for reactor licensing or involved in development or operation of small reactors. The objectives of the group are outlined below:

(1) To produce a top-tier document with a coherent and logical structure to serve as a framework for discussion and for subsequent development of more detailed guides.

(2) To produce general, comprehensive criteria that are applicable to a broad spectrum of small reactors, are compatible with the current work by the IAEA, and are consistent with Canadian reactor safety philosophy.

(3) To stimulate subsequent guide development by designers, operators, and regulators.

There is a considerable volume of reactor safety information in the literature; however, most of it pertains to power reactors. Various countries with nuclear power programs have formulated safety criteria in regulations, regulatory guides and in various national standards that are reactor specific and tailored to the national programs. The IAEA has also published a useful set of power reactor guides in the form of Codes of Practice and Safety Guides (Safety Series No.50) developed by international consensus through its Nuclear Safety Standards (NUSS) programme.

Criteria for small reactors are not so well developed. Some countries have produced a limited number of guides and standards; in particular the American Nuclear Society has published about a dozen standards for research reactors. Some other countries are in the process of formulating guides for research reactors, but little has been published to date. As noted above, the IAEA has recently been active in developing research reactor guides.

The published reactor safety information varies in degree and in detail, ranging from broad and general regulatory safety objectives that are descriptive in nature, to detailed and extensive requirements which can be quite prescriptive, such as the regulatory guides produced in the United States. Some of the general information is not complete and misses essential concepts; nor is it detailed enough to provide useful guidance. Often criteria are presented that are ambiguous or subject to interpretation and thus are not entirely appropriate for evaluating compliance with safety concepts. On the other hand, the prescriptive requirements provide great detail, usually specific to particular reactor technologies, but are difficult to navigate and the underlying safety concepts are not readily apparent. A balance is lacking.

Moreover, the guides and standards produced for particular applications often lack a logical, coherent structure and overall rationale. Recently, however, the IAEA-sponsored International Nuclear Safety Advisory Group (INSAG) has produced a compilation of criteria for nuclear power plants in the form of basic safety principles. This report provides a logical framework for understanding the underlying objectives and principles of nuclear safety and has been well received by the reactor community.

4.2 Criteria Development

It was concluded that criteria for small reactors should include the following elements in a "one-piece" document:

- high level statements of safety goals or objectives generally applicable to any reactor,
- fundamental safety concepts, such as defence in depth or quality assurance, that are either explicit or implicit in general treatments of reactor safety, and
specific criteria for various applications, in design or operation for example.

The first two types of criteria are essential to an understanding of the underlying concepts of reactor safety and provide a basis for development of more specific criteria. The third type of criteria provide the detailed guidance. It is important that criteria be presented in a clear, understandable and unambiguous structure that proceeds from the general to the specific, and should include all concepts important to safety.

At this point, the overall scope and structure for the criteria have been defined, as shown in Figure 2. Criteria are being developed in a hierarchical or tiered structure, with more detail at each subsequent level. The first tier forms a safety philosophy consisting of top-level objectives and supporting safety principles. The second tier presents criteria for specific reactor applications, such as regulation and design.

The treatment is comprehensive, covering all factors of major importance to safety, for all phases of a reactor life-cycle. However, since a top-tier document is being prepared, the criteria will be more descriptive than prescriptive. Nevertheless, an effort has been made to write down evaluation criteria that are testable and can be used to assess whether important safety aspects are in place. Figure 3 shows the relationship of criteria to each safety concept being formulated in the hierarchy, with increasing detail at each lower tier. The criteria are interrelated and must be taken as a whole. They are not meant to constitute a menu for selection.

Although it is believed that the criteria will be of use for any small reactor activity, the scope of the criteria and degree to which they are applicable may in some cases be reactor specific. Where this is the case, a more common reactor type has been chosen as the basis, namely a medium power pool-type reactor operating at relatively low coolant temperature and pressure, similar to the 5 MW McMaster and 10 MW MAPLE research reactors, or the 10 MW SES heating reactor. The criteria can be extended, with caution, to reactors of a different type or power level. However, reactors of quite different purpose, marine propulsion for example, would require additional criteria specific to this purpose. Other reactor types may be able to achieve the intent of the safety criteria by special inherent features, making some of the criteria not fully applicable; an example would include some low power reactors. Above the high power end of the spectrum under consideration, power reactor criteria would be more appropriate; reactors operating at power levels above several tens of megawatts, with higher temperature, pressurized coolants, would fall into this category.

4.3 Safety Philosophy

Basic concepts have been gathered together at the top level as a safety philosophy, comprising statements of objectives together with a number of fundamental principles of reactor safety. The objectives state what is to be achieved and the principles provide the basic safety concepts necessary to achieve these objectives. Evaluation criteria are being developed which are testable and can be used to assess whether elements of the philosophy are in place; all elements of the philosophy must be present to
achieve safety in the long term. These basic concepts can be thought of as similar to the IAEA top-tier Safety Fundamentals.

The safety philosophy is itself structured in levels as shown in Figure 4. The first level comprises a basic safety objective and three supporting objectives which are stated below. Evaluation criteria will be provided for each of these objectives.

(1) Basic Safety Objective:

The basic safety objective is to protect individuals, society and the environment by establishing and maintaining in small reactor facilities an effective defence against radiological hazard.

(2) Supporting Objectives.

(a) Radiological Protection Objective:
Radiation exposure within the facility and that due to any releases from the facility shall be kept as low as reasonably achievable and below prescribed limits in all operational states.

(b) Risk Limitation Objective:
Radiological consequences, if any, of accidents in small reactor facilities shall be maintained within acceptable bounds.

(c) Environmental Protection Objective:
There shall be no significant detrimental effects on the environment for all reactor operational states and those accidents taken into account in the reactor design; the impact resulting from accidents beyond the design basis shall be mitigated to the extent practicable.

At present the second level of the safety philosophy consists of fundamental principles together with their evaluation criteria. Two of the principles, "Defence in Depth" and "Safety Culture" are broadly based and encompass all the others; consequently, the principles have been formulated in two groups as shown in Figure 4.

---

**FIGURE 4  STRUCTURE OF SAFETY PHILOSOPHY**
Defence in depth can be described as multiple layers of overlapping provisions to prevent, correct or to compensate for deficiencies or failures. It should be in place for all safety activities, whether organizational, behavioral or equipment related. Defence in depth strategy comprises four key elements, namely: prevention, control, protection and mitigation. In turn, each of these elements is provided by two parallel and interlocking types of measures: measures related to equipment and measures related to human activities.

The individual principles identified in each group are indicated in Figure 4. Both the principles and corresponding evaluation criteria are still under development and the list will be augmented by the time of the final report. Nevertheless, it has become clear that the principles are strongly interlocking and that all must be in place to achieve a high level of safety. Application of defence in depth requires safety culture principles; the converse is also true.

Safety culture can be described as the pursuit of excellence in all matters pertaining to reactor safety. It involves a pervasive safety thinking, personal dedication and accountability, and should govern the actions and interactions of all individuals engaged in any activity which has a bearing on safety. Safety culture can be characterized by four elements: commitment, direction, competence, and assessment.

The second major level of the structure gathers criteria for applications in particular areas to provide some specific guidance. These are of two types as summarized in Figure 5: general applications such as quality assurance or safety assessment which are applicable over a wide spectrum of activities, and applications tailored to particular reactor life-cycle phases, such as siting or operations. An attempt has been made to provide comprehensive treatments in each area, and not excessively detailed and prescriptive requirements. Thus the emphasis has been to produce a checklist with evaluation criteria where practicable, and not a cookbook.

The criteria structure proposed here would be suitable for subsequent development of detailed guides. Such criteria would provide detailed guidance for designers, operators and regulators, and would be similar in scope to the IAEA reactor Safety Guides. These would be quite detailed and
reactor specific and are better produced by
groups of experts in each particular field; they
will not be attempted by our group. The success
of any subsequent programs would depend on
perceived need, interest, and available
resources.

The criteria being developed by the present
working group are meant primarily for
consideration by the sponsoring organizations.
However, it is intended to seek comments through
consultation and wide peer review. A preliminary
document will be completed later this year and
reviewed by each sponsoring organization.
Following this review process, an amended
document will be widely distributed for peer
review both in Canada and internationally. This
process of review and revision will improve the
document but is not meant, necessarily, to
produce a set of consensus criteria accepted by
any or all organizations. The decision whether
or not to adopt the revised document in whole or
in part would remain the prerogative of each
organization.

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SPORTS-M PREDICTION OF THE TRANSIENT BEHAVIOR
IN A MAPLE HEAT TRANSFER TEST FACILITY

by

P.J. HILLS, S.Y. SHIM, J.E. KOWALSKI AND K.O. SPITZ

ABSTRACT

SPORTS-M is a computer code developed to perform steady-state and transient thermalhydraulic analyses of two-phase flow in a piping network. To provide a database to develop heat transfer and void fraction correlations pertinent to the finned fuel in a MAPLE reactor, experiments have been conducted in a single-pin heat transfer test facility. A loss-of-secondary-side-cooling accident was simulated in the facility and SPORTS-M was used to predict the sheath temperatures during this transient. The results show the heat transfer correlations in SPORTS-M provide a conservative prediction of the wall temperature for the entire transient, when an equivalent sheath-mass pin radius is used in the fuel conduction model.

NOMENCLATURE

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
<th>Units</th>
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<tr>
<td>( \rho )</td>
<td>density</td>
<td>kg( \cdot )m(^{-3} )</td>
</tr>
<tr>
<td>( \tau )</td>
<td>shear stress per unit length</td>
<td>N( \cdot )m(^{-1} )</td>
</tr>
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<td>( \sigma_r )</td>
<td>surface tension</td>
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<td>( \xi )</td>
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<tr>
<td>( \alpha )</td>
<td>void fraction</td>
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<tr>
<td>( \phi^2 )</td>
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</tr>
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<tr>
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<tr>
<td>( \phi^2 )</td>
<td>two-phase multiplier</td>
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INTRODUCTION

MAPLE (Multipurpose applied Physics Lattice Experimental) (1) is a new research reactor developed by Atomic Energy of Canada Limited (AECL). It is a light-water-cooled pool-type reactor using fuel assemblies in flow tubes. The fuel assemblies contain U;Si-Al dispersion fuel with an aluminum cladding co-extruded with eight fins of rectangular cross section to improve the heat transfer characteristics. During normal operation of the reactor the coolant is forced upwards through the core, at a high velocity and high subcooling, causing heat transfer in the forced convection regime. However, subcooled boiling is predicted to be the dominant cooling mode during some postulated upset conditions. This regime is important thermally because very effective heat transfer from the heated wall is achieved without a great increase in the wall temperature. The void generated in this heat transfer regime also affects the reactivity coefficient and acts to limit the total power generated.

The objective of the thermalhydraulic studies being carried out by AECL is to develop computer codes for the MAPLE that are fully validated by separate and integral effects experiments. The validated codes will be used to study the normal and upset performance of the reactor in support of design and licensing.
The SPORTS-M computer code (2,3) has been developed and extensively used for the design and safety analyses of the MAPLE. The code performs thermalhydraulic simulations of the behavior of the reactor during various upset conditions. The fully implicit numerical scheme employed for all hydraulic and heat transfer variables makes the code particularly suitable for the analysis of slow transients and thermosiphoning flows in the presence of large volumes of fluid. SPORTS-M solves the conservation equations of mass, momentum and energy together with the equation of state for a one-dimensional (1-D) transient flow of a two-phase mixture.

A heat transfer package and a radial heat conduction module are coupled to the hydraulic modules. The package contains heat transfer correlations selected from the literature to be applicable for the subcooled boiling conditions of low pressure and high subcooling. The heat transfer correlations developed for the finned geometry (4,5) cannot be applied directly with the 1-D conduction model because they require temperature distribution around the periphery of a finned pin.

The purpose of this study was to benchmark the heat transfer correlations defined in SPORTS-M against experimental results from the MAPLE test facility. The heat transfer package used in SPORTS-M predicts subcooled boiling heat transfer using the Thom et al. (6) or Chen (7) correlations. These correlations may not be applicable for the MAPLE flow conditions or for heat transfer from finned fuel.

This paper presents numerical and experimental studies simulating the behavior of the MAPLE driver fuel element simulator (FES) during a loss of coolant to the secondary side of the heat exchanger. The experiments and their results are described. Measurements of the FES surface and fluid bulk temperatures and pressures along the FES are compared with numerical predictions. The validity of the heat transfer correlations are shown for two modeling assumptions of the 2-D FES with a 1-D fuel model. The SPORTS-M convergence scheme and the use of the homogeneous two-phase friction multiplier are demonstrated with the existence of large voids.

MATHEMATICAL MODELLING

The mathematical model of SPORTS-M consists of the hydrodynamic equations, constitutive relations, and the equation of state for the fluid plus boundary conditions for a homogeneous two-phase flow in a piping network.

Hydrodynamic Equations

SPORTS-M uses a one-dimensional two-phase flow model. The flow in a pipe is represented by variables averaged over the cross section. The governing equations for a one-dimensional two-phase mixture are as follows:

\[ \frac{\partial \rho}{\partial t} + \frac{\partial \rho u}{\partial x} = 0 \tag{1} \]

\[ \frac{\partial \rho u}{\partial t} + \frac{\partial \rho u^2}{\partial x} - \frac{\partial \rho}{\partial x} - \rho g \frac{\partial x}{\partial x} + \tau_v + \rho u \tag{2} \]

\[ \frac{\partial (\rho H - p)}{\partial x} = q'''' + \rho u \tag{3} \]

\[ \rho = \rho (p, h_L, \alpha) \tag{4} \]

where

\[ H = h - \frac{u^2}{2} - gE \tag{5} \]

\[ \nu = \frac{h_L}{\Delta x} \tag{6} \]

\[ \tau_v = \rho \beta \left( \frac{V}{\Delta x} - \frac{f}{D} \right) - \frac{u U}{2} \tag{7} \]

\[ q'''' = \frac{h_{ht} A_{ht} (T_v - T_s)}{A_f \Delta x} \tag{8} \]

Constitutive Equations

Wall Momentum Transfer. The following relations are used in the SPORTS-M code to evaluate the wall shear force in Equation (7):

\[ f = \begin{cases} f_1, & \text{for } \text{Re} \geq 2400 \\ f_2, & \text{for } \text{Re} < 2400 \end{cases} \]

Friction factor \( f \), a fit to the Moody curves using a 60-\( \mu \)m roughness coefficient, is given by (8):

\[ f_1 = 0.0055 \left[ 1 + \left( \frac{20000 \sqrt{\text{Re}}}{D} \right)^{1/4} \right] \]

\[ f_2 = \max \left( \frac{64}{\text{Re}}, f \text{ at } 2400 \right) \]

The two-phase friction multiplier is defined as unity when void fraction is zero. When void is generated the homogeneous two-phase multiplier (9) is given by:

\[ \phi^2 = \left[ 1 - x_t \left( \frac{\rho_v - \rho_g}{\rho_g} \right) \right]^2 \left[ 1 - x_t \left( \frac{h_v - h_g}{h_g} \right) \right]^{0.25} \tag{9} \]

The true flow quality is determined by using the relationship suggested by Levy (9):

\[ x_t = x_{eq} - x_t \cdot e^x \tag{10} \]

where

\[ x_{eq} = \frac{h - h_r}{h_{eq}} \]

\[ x_t = \frac{-0.0022 \frac{a^2}{D} \frac{C_{fs}}{C_{fp}}}{h_{fs} k_n} \quad \text{for } Pe < 70000 \]

\[ x_t = \frac{-154.0 \frac{a^2}{D} \frac{C_{fs}}{C_{fp}}}{\rho_m u h_{fs}} \quad \text{for } Pe \geq 70000 \]

\[ J = \frac{x_{eq}}{x_t} - 1.0 \]
The local pressure drop due to an abrupt area change is taken into account using Equation [11].

\[ p_d = p_u - \Delta p_{acc} - \Delta p_{ent} \]  \[ 11 \]

where

\[ \Delta p_{acc} = 0.5 \rho (u_1^2 - u_2^2) \]

\[ \Delta p_{ent} = \text{entrance pressure loss due to an area change.} \]

For a sudden expansion:

\[ \Delta p_{ent} = 0.5 \rho u_2^2 \left[ 1 - \left( \frac{D_2}{D_1} \right)^2 \right] \]

For a sudden contraction:

\[ \Delta p_{ent} = 0.5 \rho u_1^2 \left[ \frac{1}{C} - 1 \right]^2 \]

The loss coefficient \( C \) is given by Streeter and Wylie (10).

Wall-Fluid Heat Transfer. The heat transferred from the heated wall to the fluid is obtained from Equation [8] with the appropriate heat transfer correlations for \( h_{RT} \). The heat transfer package in SP0RTS-M addresses all the heat transfer regimes of the boiling curve. The correlations and heat transfer criteria in the package are summarized in Table 1.

**TABLE 1**

<table>
<thead>
<tr>
<th>Heat Transfer Mode</th>
<th>Criteria</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Single Phase</td>
<td>Lamihar</td>
<td>Re&lt;2000</td>
</tr>
<tr>
<td>Transition</td>
<td>2000 &lt;Re&lt;2500</td>
<td>Interpolation</td>
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<td>Flow Turbulent</td>
<td>2500 &lt;Re</td>
<td>Davis-Anderson (13)</td>
</tr>
<tr>
<td>OSN Temperature</td>
<td>Saha-Zuber</td>
<td>Chen (5)</td>
</tr>
<tr>
<td>Partially Subcooled Boiling</td>
<td>T_{SBN}&lt;T_{UB}&lt;T_{CE}</td>
<td>Interpolation using Thom (4)</td>
</tr>
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<td>OSN Temperature</td>
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<td>Chen (5)</td>
</tr>
<tr>
<td>Fully-Developed Subcooled Boiling</td>
<td>T_{CE}&lt;T_{UB}&lt;T_{CHF}</td>
<td>Bjornard-Griffith (16)</td>
</tr>
<tr>
<td>Saturated Boiling</td>
<td>T_{UB}&lt;T_{CHF}</td>
<td>Chen (5)</td>
</tr>
<tr>
<td>CHF Temperature</td>
<td>Groeneveld-Rousseau (15)</td>
<td></td>
</tr>
<tr>
<td>Transition Boiling</td>
<td>T_{CHF}&lt;T_{UB}&lt;T_{TR}</td>
<td>Groeneveld (15)</td>
</tr>
<tr>
<td>Heat Transfer Film</td>
<td>T_{TR}&lt;T_{UB}</td>
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</table>

Void fraction and true quality are defined as zero at the ONB heat flux. Equation [12] is used to determine a local heat flux, \( q^{void} \), corresponding to void fraction \( a^v \). This local heat flux is nominally the same as the heat flux at the OSV criterion. A true quality, \( \chi^v \), is defined at the \( q^{void} \) heat flux using Equation [10]. For a local heat flux greater than \( q^{void} \) and less than \( q^{void} \), true quality and void are assumed to be:

\[ a = \frac{\chi^v}{\rho_{voil}} \]  \[ 12 \]

where \( \rho_{voil} \) is the distribution constant.

\[ \beta' = \frac{X_{v}}{X_{t} - (1-X_{t}) \left( \frac{\rho_{v}}{\rho_{t}} \right)^{C-1}} \]

and the mean relative velocity, \( U' \) is given by:

\[ U' = 2.9 \left( \frac{X_{voil}}{X_{t}} - \frac{X_{void}}{X_{t}} \right)^{0.75} \frac{\rho_{v}}{\rho_{t}} \frac{\rho_{v}}{\rho_{t}} U \]

The subcooled boiling heat transfer regime is divided into partial and fully developed subcooled boiling. The partial subcooled boiling regime, defined to be between OSV and CHF, is important thermally because very effective heat transfer from the heated wall is achieved without the presence of significant void in the system. A fully developed subcooled boiling regime, defined to be between OSV and CHF, is both hydrodynamically and thermally important, since the generated void influences pressure losses in the system and the resulting nucleate boiling is an effective heat transfer mode.

### Void Model for Subcooled Boiling

The Zuber and Findlay correlation (9) used to calculate subcooled void fraction is given by:

\[ a = \frac{\chi^v}{C_{c}} + \frac{\beta'}{U'} \]

where \( \rho_{voil} \) is the distribution constant.

\[ \beta' = \frac{X_{v}}{X_{t} - (1-X_{t}) \left( \frac{\rho_{v}}{\rho_{t}} \right)^{C-1}} \]

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Particular emphasis was given to the detailed modelling of subcooled boiling since the subcooled boiling process is the dominant heat transfer and hydrodynamic mechanism in the MAPLE reactor. The subcooled boiling heat transfer regime is divided into partial and fully developed subcooled boiling. The partial subcooled boiling regime, defined to be between OSV and CHF, is important thermally because very effective heat transfer from the heated wall is achieved without the presence of significant void in the system. A fully developed subcooled boiling regime, defined to be between OSV and CHF, is both hydrodynamically and thermally important, since the generated void influences pressure losses in the system and the resulting nucleate boiling is an effective heat transfer mode.

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The Zuber and Findlay correlation (9) used to calculate subcooled void fraction is given by:

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where \( \rho_{voil} \) is the distribution constant.

\[ \beta' = \frac{X_{v}}{X_{t} - (1-X_{t}) \left( \frac{\rho_{v}}{\rho_{t}} \right)^{C-1}} \]

and the mean relative velocity, \( U' \) is given by:

\[ U' = 2.9 \left( \frac{X_{voil}}{X_{t}} - \frac{X_{void}}{X_{t}} \right)^{0.75} \frac{\rho_{v}}{\rho_{t}} \frac{\rho_{v}}{\rho_{t}} U \]

The subcooled boiling heat transfer regime is divided into partial and fully developed subcooled boiling. The partial subcooled boiling regime, defined to be between OSV and CHF, is important thermally because very effective heat transfer from the heated wall is achieved without the presence of significant void in the system. A fully developed subcooled boiling regime, defined to be between OSV and CHF, is both hydrodynamically and thermally important, since the generated void influences pressure losses in the system and the resulting nucleate boiling is an effective heat transfer mode.
The boundary condition at the fluid-wall interface is given by

$$-k \frac{\partial T}{\partial r} = h_{nt}(T - T_w)$$

The heat conduction equation is solved by a fully implicit scheme for wall temperature and heat transfer coefficient. This scheme is consistent with fully implicit hydrodynamic variables such as density, velocity, pressure and enthalpy. Since boiling heat transfer is usually described as a function of a wall temperature, this implicit treatment is important when an abrupt boiling regime transition takes place.

To facilitate modelling of the fuel pin, the user may divide the fuel pin into five radial regions of different materials. Each region can be discretized into a user-specified number of segments. The material properties such as density, specific heat and thermal conductivity may be input as a function of nodal temperature. A heat generation rate for each region may also be input as a function of time. An axial heat generation distribution may be specified as a function of axial distance along the fuel channel.

**Initial and Boundary Conditions**

The mathematical formulation of Equations [1] to [3] is complete when boundary conditions are specified. For a steady-state solution, SPORTS-M allows three types of boundary conditions: (a) inlet enthalpy, inlet pressure and outlet pressure for an open system; (b) inlet mass flow rate, inlet enthalpy and outlet pressure for an open system; and (c) pressure and/or enthalpy anywhere in a closed system.

Because of the node-to-node forward-marching advantage, a set of initial conditions is not needed; instead, only the pressure and enthalpy boundary condition and a velocity estimate at the boundary node are required.

**NUMERICAL MODELLING**

**Finite-Difference Formulation**

The hydrodynamic Equations [1] to [3] are discretized by taking the forward difference in space. The variables in the transient terms of these equations are evaluated to be the average between the two adjacent spatial nodes. All the variables are evaluated at the current time step. A detailed derivation of the discretized equations is given in reference 17.

The heat conduction equation is discretized by taking forward difference in time and space, as in reference 18.

**Solution Methods**

**Overall Solution Method.** The discretized equations of the hydrodynamics and heat conduction are solved iteratively in their primitive form. No matrix solution algorithm is needed to solve the hydrodynamic equations; a simple matrix algorithm TDMA (Tri-Diagonal Matrix Algorithm) is used to solve the heat conduction equation.

The solution method of the hydraulic part of SPORTS-M resembles that described by Chattoorgoon and Thibeault (17). The scheme maintains a fully implicit, forward-marching, finite-difference method. Figure 1 illustrates the coupling of the hydraulic and heat transfer parts of SPORTS-M.

![FIGURE 1: Solution Scheme Used in SPORTS-M](image)

Let the solution procedure begin at the current time step, n-1, and the spatial step, i-1, following the boundary node, i. The conservation equations are solved sequentially for \( \rho, u, p, h, \) and \( T \), where necessary, for node \( i \) at time \( n-1 \). The mass equation is solved for \( u_{i-1} \) using an estimate of \( p_{i-1} \) and the user-supplied initial estimate of \( u_{i-1} \). The momentum and energy equations yield \( p_{i-1} \) and \( h_{i-1} \), respectively, and an improved \( p_{i-1} \) is obtained from the equation of state and the void fraction if void is present. The conservation equations are solved again in an iterative manner using the improved density each time until convergence is obtained for the density at that node. Following the density convergence at a node, the solution proceeds from node to node until it reaches the exit pressure boundary node. The estimate of velocity at the entrance pressure boundary node is corrected, based on the difference between the calculated and boundary condition pressures. Once the density and velocity converge to a unique solution, the iteration process proceeds to the next time step.

When a heated channel is encountered in a system, the fluid energy equation, wall-fluid heat transfer, and the heat conduction equation are solved iteratively in series until convergence is obtained. As illustrated in Figure 1, the hydrodynamic variables are passed to the heat transfer package where a heat transfer coefficient is determined. The heat conduction equation is then solved for the wall temperature using the heat transfer coefficient and fluid temperature. The resulting wall-fluid heat flux is returned to the energy equation. The energy equation is then solved and an updated fluid temperature becomes available. This process continues until the fuel-wall temperature used in the heat transfer correlation agrees with the fuel wall temperature obtained from the conduction equation, and the nodal fluid temperature used in the heat transfer package agrees with the temperature resulting from the energy equation.

The true quality and void correlations defined by Equations [10] and [12] respectively, are determined during the solution of the energy equation. The density and enthalpy variables in the conservation equations are defined as two-phase values. Determination of the improved density at each iteration as shown in Equation [4] is a function of the void that exists at the node. During subcooled boiling the nonequilibrium effects of void are accounted for through the following definition using true quality:
The liquid enthalpy is used in the equation and the density is determined by:

\[ \rho = (1-a)\rho_f - a\rho_v \]

Heat removal from a heat exchanger may be with a wall model. The wall model determines removal from a plate-type countercflow heat exchanger. When a wall model is encountered in the SPORTS-M code an energy equation for the primary fluid, the wall and the secondary side fluid, node along the heat exchanger, into a solution. The solution of the matrix yields the temperature profile across the heat exchanger. The heat each node is determined from the energy equation and is returned to the mainline routine using the heat fluxes calculated the wall model and determines the primary side temperature at the heat exchanger exit. The solution converges when the heat exchanger exit temperature converges.

**Convergence and Accuracy**

With appropriate boundary conditions in an initial steady-state condition is obtained via a transient analysis. The solution at each step is then obtained based on the previous step. For transient solutions, the initial velocity at the beginning node for the new time step is the previous time step velocity, and the calculations proceed through the remainder of the system. Calculated pressure at the boundary agrees with the specified convergence criterion with the pressure, the calculations proceed to the next step.

SPORTS-M uses four iterative steps: (a) the convergence at every node, (b) the convergence for the overall system, (c) the drop convergence in a parallel branching, and (d) the convergence in channel. The solution accuracy can be contrived user-specified tolerance values for the above items. For instance, the tolerance value for pressure convergence has to be smaller for a siphoning flow than that for a pumped flow. The solution method at each time step ensures convergence criteria specified by a user satisfied.

**MAPLE HEAT TRANSFER EXPERIMENTS**

**Test Facility**

Figure 2 shows a schematic flow diagram of the MAPLE single-pin heat-transfer test facility. The apparatus consists of a surge tank, heat exchange circuit, pump driven by a varial motor, the test section and interconnects. The test section is made from 16-mm I.D. glass to allow visual observations of the heat transfer. An indirectly heated, fuel simulator (FES) is located inside the glass tube. The solution method at each time step ensures convergence criteria specified by a user satisfied.

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couples. The surface thermocouples were attached on the sheath at the center of the section between two fins, and on the fin tip. The inlet and outlet temperatures were monitored by means of resistance temperature detectors. The inlet and outlet pressures of the test section were measured along with pressure drops across each third of the heated section. The power input was calculated as the product of the measured voltage and current at the heater. Coolant flow rates were measured by a turbine flow meter. The signals from the instruments were logged by a Macintosh computer approximately every 5 s in continuous scan mode.

Experimental Procedure

Three experiments were conducted at constant pressure, power and flow. In each experiment, data collection was started then the cooling water to the heat exchanger was ramped closed. The inlet temperature was allowed to increase until the FES power was manually tripped when void existed all along the FES and CHF was anticipated. Table 2 shows the inlet parameters of the three experiments that were simulated. The initial thermalhydraulic conditions in Experiment A are the same as the predicted normal operating conditions (NOC) of a MAPLE-X10 driver fuel site. Experiment B was conducted to observe the transient with the same subcooling but at a reduced flow and power. Experiment C was conducted at normal operating flow but at 1.7 times the normal fuel pin power.

<table>
<thead>
<tr>
<th>TABLE 2</th>
<th>EXPERIMENTAL CONDITIONS</th>
</tr>
</thead>
<tbody>
<tr>
<td>EXPERIMENT</td>
<td>PRESS kPa</td>
</tr>
<tr>
<td><strong>A</strong></td>
<td>180</td>
</tr>
<tr>
<td><strong>B</strong></td>
<td>135</td>
</tr>
<tr>
<td><strong>C</strong></td>
<td>200</td>
</tr>
</tbody>
</table>

The transient pressure and enthalpy boundary conditions, taken from the experimental results, were imposed at the first and last nodes of the model. The initial pressure profile along the test section was matched to the initial pressure profile of the experiment by adjusting the minor loss coefficients for each segment in the model.

The fuel conduction model in SPORTS-M is one-dimensional; therefore, "pseudo" uniform temperature distribution around the fuel pin periphery has been assumed. Heat transfer was calculated based on two equivalent radii; the first defined a sheath radius such that sheath mass was conserved \( r_s \), the second defined the sheath radius such that the total heat transfer surface area over a pin was maintained \( r_s \). In the results that follow, a wall temperature labeled with a superscript \( M \) represents the predicted wall temperature when the fuel pin was modelled using radius \( r_s \). A superscript \( A \) was used to represent the prediction using radius \( r_s \).

**Results**

In the following figures, the experimental results are traced with a solid line and simulation results are shown with a dashed line. Predicted heat transfer criteria, ONB and OSV, are drawn with a dotted line.

**Simulation at MAPLE-X10 normal operating condition.**

Figures 5 to 8 compare SPORTS-M predictions with the transient results of Experiment A. The initial conditions of this experiment were matched with the normal operating conditions of MAPLE-X10. Figure 5 shows the rate of temperature rise at the test section inflow that resulted from shutting off the cooling water from
the heat exchanger. This boundary condition is imposed in the SPORTS-M simulation. The calculated fluid temperature at the test section compares well with the measured values. However, flow rate is slightly higher than the measured values. This may be due to SPORTS-M underestimating flow rate when a large amount of void is generated.

ONB is predicted to occur at approximately 500 s but was observed during the experiment a little more than 70 s later. OSV, observed at 670 s, would only have been predicted at this location some time after 2000 s. It is interesting to note that if a fuel pin radius were chosen so the predicted sheath temperature exactly followed the measured sheath temperature, ONB would be predicted about 300 s after it was observed. This demonstrates that for the current correlations specified in the heat transfer package no single equivalent radius will correctly predict the surface temperature and the point of ONB.

Figure 7 shows the measured vessel temperature predictions for the lower elevation, much closer to the lower elevation than they did at the middle elevation. This is because at this elevation much less void is being generated.

Figure 8 compares the differential pressures measured and predicted across the test section. The differential pressure between the test section inlet and the lowest pressure tap in the test section, labeled DP-in, decreases slightly during the transient. The trace of the simulation results shows a rise towards the end of the transient. The differential pressure between the lowest and highest pressure taps on the test section, labeled DP-low, displays a smooth exponential rise starting at 1000 s. The simulation shows the correct trend but is much more abrupt, starting at approximately 700 s. OSV was predicted to occur at node 34 at this time and higher void generation caused the predicted pressure differential to rise. The predicted pressure differential in the top part of the test section also shows the correct trend but again is delayed due to the void generation after 1700 s.
Simulation at Low-Flow Conditions. Figures show the transient results of Experiment E experiment was conducted to simulate the heat mechanisms during low-flow conditions. The flow rate of 0.2 L/s represents approximately 25% of flow in a MAPLE-X10 driver site. The transient temperature imposed in the simulation is shown in Figure 9. The simulated flow rate and outlet temperature follow the measured values of these parameters throughout the transient.

The coolant temperature prediction was very close to measured values up until void occurred and the measured value increased where the predicted temperature did not. The close agreement of the outlet temperature prediction to measured values as shown in Figure 9 indicates that total heat removal rate is correctly accounted for.

Figure 11 shows that OSV is being predicted at the lowest measurement location at the end of the experiment. In fact this was generally true for all three experiments. OSV was just happening at the FES inlet when large voids existed at the FES outlet and the experiment was terminated.

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Figure 12 shows the differential pressure measured across the test section inlet decreases by 0.5 kPa, whereas the predicted pressure drop increases by 0.5 kPa. This is the same trend observed in Experiment A.

The measured pressure drop across the FES stayed relatively constant till 1000 s then increased sharply as the amount of void increased. The predicted pressure drop shows the identical abrupt change about 90 s earlier. The pressure drop across the top of the test section shows the correct trend.

Figure 13 shows the predicted fluid density and void at the last three nodes modelling the FES. Significant void at nodes 34, 33 and 32 is predicted to occur between 940 and 970 s. The void is predicted to increase gradually until the end of the simulation when 45% void is predicted to be present at node 34. The smooth decrease of the nodal density was expected by the subcooled boiling model of Equations [15] and [16].

The void fraction along the length of the FES for three times near the end of simulation is shown in Figure 14. The figure indicates the rate at which the void front is propagating down the channel.

Temperature measurements at the middle elevation for the high-power experiment are shown in Figure 16. The predicted wall temperature using the equivalent radius $r_e$ is 20°C higher than the measured sheath temperature. This results in ONB being predicted at the start of the transient. The predicted wall temperature using the equivalent radius $r_e$ initially
traces the fin-tip temperature but near the end of the transient exceeds the sheath temperature. At the end of the simulation OSV still is not close to being predicted using either of the equivalent radius assumptions.

A two-dimensional fuel conduction model is being developed and will be implemented into the code along with the heat transfer correlations developed for the MAPLE fuel.

REFERENCES


CONCLUSIONS

It is concluded from this study that:

- SPORTS-M simulated the fuel heat transfer behavior well during a loss of secondary side cooling.

- FES surface temperature was predicted to be greater than measured sheath temperature when a sheath-mass equivalent radius was assumed in the fuel conduction model. This means the equivalent sheath-mass assumption gave conservative predictions for the three conditions studied.

- Using a sheath-area equivalent radius in the fuel conduction model resulted in a lower predicted wall temperature, sometimes less than measured fin-tip temperature. This assumption results in nonconservative predictions.

- The ONB correlation of Davis and Anderson predicted the onset of nucleate boiling reasonably close for the normal operating conditions of MAPLE-X10, however for low-flow and high-power conditions ONB was predicted very early.

- The Saha and Zuber correlation predicted the OSV too late in all three transients to accurately predict the transient pressure distribution.

FUTURE WORK

Void fraction and two-phase pressure loss experiments are planned for this test facility in the near future. Further validation of the SPORTS-M void correlation and two-phase friction multiplier against experimental measurements is planned.


Session 3:

CANDU Performance & Improvements

Chairman:

D.W. Bredahl, AECL-CANDU Operations
OXYGEN ADDITION TO THE ANNULUS GAS SYSTEMS
OF PICKERING NGS A, UNITS 3 AND 4

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ABSTRACT

An Oxygen \((O_2)\) addition system has been designed and installed in conjunction with the Annulus Gas System (AGS) of Pickering NGS A, Unit 3. It has been operational since January 1989. One of the short term purposes of \(O_2\) addition is to remove Carbon-14 (C-14) deposits on pressure tube (PT) surfaces through oxidation to carbon (14) dioxide which is subsequently removed by absorption/chemical reaction in a calcium hydroxide bed. Removal of C-14 deposits is necessary to minimize the occupational dose to workers during retubing.

Experience to date has shown satisfactory operation of the \(O_2\) addition system for Unit 3 AGS, which has provided controlled \(O_2\) concentrations up to 10 percent (V/V). The \(O_2\) bottle supply system has involved significant handling requirements by the station. This will be minimized for Unit 4 where a liquid \(O_2\) tank has been selected for the supply. The Unit 4 \(O_2\) addition system is otherwise similar to that for Unit 3 except that lower \(O_2\) addition rates are accommodated. This reflects the fact that \(O_2\) addition for Unit 3 is designed for purge operation only whereas that for Unit 4 is designed for both purge and recirculation operation. The latter mode is required for maintaining an impervious oxide layer on the surface of the PTs when satisfactory C-14 decontamination has been achieved. It is considered that this minimizes \(D_2\) ingress and hence the potential for PT hydriding. Unit 4 is scheduled to shut down for retubing in May 1991.

Throughout the design it has been ensured that the leak detection capability of the AGS is not impaired by \(O_2\) addition.

1.0 INTRODUCTION

Extensive examination of pressure tubes from operating Ontario Hydro nuclear reactors has led to the conclusion that a major contributor to the high deuterium (\(D_2\)) levels measured is \(D_2\) pick-up from the annulus side of the tube. Figure 1 shows a typical end fitting section of a fuel channel indicating the route by which \(D_2\) from the Primary Heat Transport (PHT) system is considered to enter the AGS. Subsequent hydriding of the PT can contribute to failure of the PTs.

It is considered that \(D_2\) pick-up is due mainly to insufficient oxidant in the annulus gas to maintain a protective oxide layer on the outer PT surface. Consequently, Ontario Hydro with the concurrence of AECL have produced the following chemistry specification:

<table>
<thead>
<tr>
<th>Compound</th>
<th>Specification</th>
</tr>
</thead>
<tbody>
<tr>
<td>Generic (O_2) range</td>
<td>0.5 to 5.0% v/v</td>
</tr>
<tr>
<td>Desired (O_2) range</td>
<td>0.5 to 2.0% v/v</td>
</tr>
<tr>
<td>(D_2)</td>
<td>&lt;0.1% v/v</td>
</tr>
<tr>
<td>Dew Point ((D_2\O))</td>
<td>≤0°C</td>
</tr>
</tbody>
</table>

As the present standard specifications for \(N_2\) or \(CO_2\) gas supplies for the AGS are <0.1% v/v \(O_2\), an \(O_2\) addition system is necessary. Meeting the \(D_2\) specification is aided by the \(O_2\) addition system removing \(D_2\) according to the reaction:

\[
D_2 + 1/2 O_2 = D_2O
\]

The limitation on high dew point (≤0°C) is based on leak detection requirements for the AGS as sensitivity to detection/monitoring is adversely affected at high dew points.

With the decision to retube Units 3 and 4 of Pickering NGS A in mid 1989 and 1991 respectively, a further reason for \(O_2\) addition was identified. During the retubing of Unit 1 in 1986 and to a lesser extent the retubing of Unit 2 occupational dose problems were experienced from Carbon-14 dust on the PT surfaces. It has been decided to minimize C-14 contamination problems for Units 3 and 4 by employ-
ing a C-14 Removal System in conjunction with the operation of the AGS where C-14 is produced from nitrogen (N₂) activation by an (n,p) reaction. C-14 will be removed in the gaseous phase and in order to gasify as much C-14 as possible, an O₂ addition system is required to convert C-14 deposits to C⁺O₂ and to a lesser extent suboxides. Only Pickering NGS A, Units 3 and 4 employ N₂ for the AGS. All other Ontario Hydro nuclear stations use CO₂ with which the production of C-14 is minimal.

The commencement of O₂ addition for Unit 3 AGS was January 1989 allowing 6 months before a June shutdown for retubing. It was decided that the entire 6 months should be devoted to C-14 decontamination.

The commencement of O₂ addition for Unit 4 AGS is scheduled for the end of May 1989. It is envisaged that it will operate with C-14 decontamination for the first 6 months and the last few months before shutdown for retubing in 1991. For the intervening 18 months it is intended to operate the AGS primarily for oxide renewal of the PT surfaces. This will minimize the risk of a PT failure due to hydriding and provide specimens of PT surface to examine the effectiveness of the oxide layer and establish subsequent operating parameters.

2.0 DESCRIPTION OF OXYGEN ADDITION SYSTEM

The O₂ addition systems for both Units 3 and 4 connect into the respective Annulus Gas Systems. The Pickering NGS A Annulus Gas System has been described in a previous CNS paper (Reference 1). Flowsheets of the Annulus Gas Systems for Units 3 and 4 with the O₂ addition systems incorporated are given in figures 2 and 3 respectively. There are a few differences between the O₂ addition arrangements for Units 3 and 4.

The former has a portable dolly with O₂ bottles arranged in two rows of four bottles each in an 8 pack arrangement. There is a common manifold, pressure regulating valve with integral relief valve and an isolating valve. It is located outside the reactor auxiliary bay in a weather proof housing. A second portable dolly is provided as back up to replace the first one when all the bottles are spent. Both dollies connect to a T piece via lengths of flexible hosing, which provide manoeuvrability for the 8 packs. 3/8 in. tubing connects the T piece to an instrument panel in the Reactor Auxiliary Bay.

The O₂ supply for Unit 4 is a liquid O₂ tank located on a concrete pad outside the Reactor Auxiliary Bay. There is a corresponding concrete pad on the adjacent roadway where a liquid O₂ tanker is situated during tank filling. A pipeline runs from the tank filling station into the top of the liquid O₂ tank. This tank has a vapourizer coil relying on ambient heat. A similar arrangement of pressure regulation, overpressure protection and isolation is provided as for the 8 pack bottle supply.

Significant differences between the two systems are the design pressures and storage capacities. The design pressure for the 8 pack bottles is 22.3 MPa (g) (3240 psig) whereas that for the O₂ tank is 1.7 MPa(g) (250 psig). The capacity of an 8 pack for purge operation at 5% v/v O₂ is 3 days whereas that for the tank is over 2 months. The 8 pack O₂ bottles for Unit 3 were obtained for a short time frame (6 months) and early delivery whereas the O₂ tank was obtained for extended operation (26 months). Also the handling demands on the station are greater for the 8 packs and would prove prohibitive for extended operation.

The control panels for both units include flow meters and a flow controller for accurate control and measurement of the O₂ addition rate. The total range of O₂ flow rates for purge operation is 0 to 50 L/
min (STP). The total range for recirculation operation is 0-4 L/min (STP). These figures are based on the chemistry specification for O₂ which has a maximum concentration of 5% v/v and the fact that the purge design flow rate is 340 L/min (STP). Allowance is also made for gas leaks from the AGS for Units 3 and 4 being 65 and 54 L/min (STP) respectively. Furthermore, there is provision for purge operation up to 10% v/v O₂ for C-14 decontamination, if required.

Unit 3 O₂ addition which refers to purge operation only has 2 flowmeters: 0 to 20 L/min (STP) and 0 to 50 L/min (STP). The former flowmeter covers the O₂ specification range for the AGS (0.5 to 5.0% v/v). The latter flowmeter allows O₂ concentrations up to 10% v/v to be monitored. Unit 4 O₂ addition which refers to both purge and recirculation operation has the above flowmeters for the former mode and a third flowmeter for the range 0-4 L/min (STP) to cover the latter mode of operation.

The flow controller ensures that the flow rate remains constant even though the system pressure where the O₂ enters the N₂ flow line may vary by 34.5 kPa (5 psi). As the pressure drop downstream of the O₂ delivery pressure regulating valve (PRV) is relatively small (except for high O₂ flow rates) the pressure variation could have a significant effect on the O₂ flow rate which may lead to the AGS O₂ concentration going out of specification.

The 3/8 in. tubing from the control panel to the point of intersection with the AGS is welded in sections to minimize O₂ leakage. An O₂ vent connection with isolating valve is provided at the intersection to allow venting of any air which may enter the O₂ addition system and hence the AGS. This is more likely with Unit 3 where a spent O₂ 8 pack dollies is replaced by a full one at intervals. Air contains Argon-40 which is converted to Argon-41 during passage through the reactor. The latter is a gamma emitter causing radiation fields on the AGS piping and increasing the occupational dose for operators.

Finally an O₂ monitor is provided for Unit 4 AGS to confirm that the O₂ concentration during recirculation is within the technical specification. The instrument is a Systech ZR 891 Oxymonitor employing a high temperature zirconia ceramic cell. It is located in a sample line downstream of the dew point monitors. The sample flow is less than 0.1% of the AGS design flow and is vented continuously to the Reactor Building Ventilation system. There is no O₂ monitor for Unit 3 where manual samples are taken regularly for analysis.

Safety Regulations for Oxygen Storage and Handling

It was necessary to site both the 8 pack dollies for Unit 3 and the liquid O₂ tank for Unit 4 outside the building. Otherwise a protective metal fireproof cabinet would be required to house the O₂ supply which is costly and hinders access in the Reactor Auxiliary Bay. It was also necessary to weld the tubing connections for the supply line to prevent O₂ leakage. Both the 8 pack dollies and the liquid O₂ tank have non-combustible canopies for weather protection. The liquid O₂ tank has a concrete pad for support and a separate one where the liquid O₂ tanker stands for refilling the tank. This minimizes the possibility of a fire if spillage occurs on an otherwise asphalt surface.

The appropriate regulations are contained in the Ontario Hydro Fire Loss Control Guide (Reference 2) and appropriate sections of the National Fire Protection Association Standards (References 3 and 4).

3.0 PRESSURE TUBE LEAK DETECTION CAPABILITY

A primary requirement of the design and operation of the O₂ addition system is that it will not compromise the D₂O leak detection capability of the AGS. It is also a requirement that the C-14 Removal Unit does not adversely affect the D₂O leak detection capability of the AGS. To meet these requirements it was necessary to place the C-14 Removal Unit downstream of the dew point monitors and operate the AGS in purge mode during C-14 decontamination.

This prevents moisture generated in the C-14 Removal System from recycling to the AGS and deadening any dew point response in the event of a PT leak. Consequently the C-14 Removal System must be isolated if the AGS is put into the recirculation mode. Also it must be confirmed that C-14 Removal is essentially complete so that C-14 emissions from the AGS via gas leaks do not significantly effect C-14 emissions. If the emissions are significant then O₂ addition must be stopped and isolated.

PHT leak detection requirements also preclude batch addition of O₂ to the recirculation mode operation. Owing to the level of gas leaks of the AGS (Units 3 and 4) addition of O₂ would require large and frequent batches. The variation in gas properties such as density, molecular weight etc. would be small because O₂ and N₂ are very similar. However, the manpower requirements for manual batch addition would be significant. Alternatively an automatic batch

![FIGURE 4](image-url)

Relation Between Steady State Purge Dew Point and D₂O Leak Rate Units 3 & 4
addition arrangement would involve greater complexity. Neither approach would be very satisfactory. However, the deciding factor would be the pressure pulse during batch O₂ addition which would significantly affect the dew point and distort the system dew point response in the event of a PT D₂O leak.

The leak detection criterion during C-14 decontamination, i.e., purge operation, is given in Figure 4. Depending on the magnitude of the D₂O leak there is a steady state dew point recorded by the hygrometers. Normally the AGS is purged if the dew point rises to -20°C. However, if it remains at or rises above this figure while on purge, this indicates a D₂O leak (1.1 g/h of D₂O) which is sufficiently significant to require the unit to shut down.

The leak detection criterion during recirculation operation is the average rate of rise of dew point over the range -35°C to -20°C. If this exceeds 7°C/h and is confirmed, the unit is required to be shut down. There is a minimum recirculation or purge flow rate of 283 L/min (STP) at which an alarm occurs. This ensures that the N₂ flow rate does not fall below the necessary flow to ensure the leak detection performance is maintained.

4.0 DISCUSSION OF O₂ ADDITION OPERATION (UNIT 3)

The O₂ addition system for Unit 3 AGS commenced operation at the end of January 1989. It operates in purge mode with the C-14 removal system exhausting to the Reactor Building Ventilation System. The initial O₂ concentration was 6% v/v. There was a corresponding burst of C⁰¹⁰₂ gasification for less than an hour during which the C-14 activity reached over 200 μCi/L. Subsequently the C-14 activity fell to a steady 2-3 μCi/L, corresponding to a removal rate of 0.8 Ci/day.

The O₂ concentration was reduced to 0.6% but it was evident that the C-14 release was not significantly affected by O₂ concentration. The O₂ concentration was then cycled to determine whether other chemical species were being produced preferentially and reducing the C-14 oxidation rate. Following this initial period, O₂ addition has been maintained between 0 and 6% with a steady removal rate for C-14 of 0.8 Ci/day. O₂ addition and decontamination will continue until unit shutdown in June 1989.

Gas samples were taken at regular intervals before and after the reactor section of the AGS as well as after the C-14 Removal Unit. The main chemical species which were analysed were O₂, CO₂ and CO (C-12 and C-14), NO, N₂O and D₂O. The D₂O concentration was about 60 ppm prior to O₂ addition. Following O₂ addition, D₂O rapidly disappeared from the AGS due to oxidation to D₂O. Similarly the dew point rose from -70°C to about -55°C. This shows no significant rise in humidity at such low dew points and is still below the trending range. Consequently, D₂O leak detection was not affected.

Another phenomenon which was observed was significant concentrations of N₂O and NO produced by reaction between O₂ and the carrier gas, N₂. The total concentration of N₂O and NO varied up to 1000 ppm at an O₂ concentration of 6%. This represents an O₂ consumption of 0.08% producing NO. This is not significant.

It was necessary to check the accuracy of the O₂ addition flowmeter and to obtain a calibration graph relating the steady state O₂ concentration in the AGS during purge to the O₂ addition rate. This enables the operator to set the desired O₂ concentration in the AGS from the O₂ addition rate. The above calibration was obtained from the correlation of measured O₂ concentrations from gas sample analyses and the corresponding figures from the flow meter measurements of O₂ addition rate. The total gas flow rate after the reactor section of the AGS and the measured gas leak rate from the AGS were also required. Care was taken to use O₂ concentration measurements when the AGS was operating with steady state gas concentrations throughout the system and constant O₂ addition for a period of time.

It was found that the O₂ addition rate was 2.75 times the actual reading on the O₂ flowmeter. First, a calibration graph was drawn up to correlate the O₂ addition rate calculated from the O₂ concentration measurements to the measured O₂ addition rate (see Figure 5). Then, the required graph (Figure 6) relating the AGS O₂ concentration to the indicated O₂ addition rate was produced to guide Operations in setting the specified O₂ concentration.

If the O₂ flowmeter reading is corrected for the actual system pressure 34.46 kPa(g) (50 psig) relative to its calibration pressure 55.13 kPa(g) (80 psig) using the theoretical relation:

\[
Q = Q(34.46 \text{ kPa(g)}) = Q(55.13 \text{ kPa(g)}) \times \sqrt{\frac{p(34.46 \text{ kPa(g)})}{p(55.13 \text{ kPa(g)})}}
\]

where \( Q(34.46 \text{ kPa(g)}) \) = O₂ flow rate (L/min (STP)) at 34.46 kPa(g) pressure
\( Q(55.13 \text{ kPa(g)}) \) = O₂ flow rate (L/min (STP)) at 55.13 kPa(g) pressure
\( p(34.46 \text{ kPa(g)}) \) = O₂ gas density (kg/L) at 34.46 kPa(g) pressure
\( p(55.13 \text{ kPa(g)}) \) = O₂ gas density (kg/L) at 55.13 kPa(g) pressure
The leak detection capability of the AGS is not compromised with O₂ addition. The variation in the properties of the gas mixture: density, molecular weight etc. are minimal for O₂ concentrations up to 10% v/v as the molecular weights of N₂ and O₂ are 28 and 32 respectively.

(b) The accuracy of sampling and analysis for O₂ concentration is not high enough to identify O₂ usage from gasifying carbonaceous deposits and/or oxide renewal of the pressure tubes.

(c) The O₂ flowmeters require calibration from O₂ concentration measurements to give accurate control of O₂ addition.

(d) A liquid O₂ tank will be more suitable than multi pack gas bottles for long term O₂ addition.

(e) The removal of C-14 to date (about 80 Ci) has been far less than anticipated but the C-14 Removal Unit has removed almost 100% of this radioclide.

(f) NO₃ concentrations are significant but are not considered serious for temporary periods of time. However, the conversion of the AGS for Units 3 and 4 from N₂ to CO₂ post retubing is important to avoid NO₃ generation during O₂ addition in the long term.

(g) The O₂ addition flowrate was found to be 2.75 times the reading on the O₂ flowmeter based on the O₂ concentration measurements. The theoretical correction factor for pressure did not resolve this discrepancy and so the calibration based on O₂ concentration measurements was used.

6.0 REFERENCES


ABSTRACT

Darlington is a four unit station, 50 miles east of Toronto. The units each have a net capacity of 881 megawatts. When originally started in 1977 they were to be in service by 1989. However, basic delays were introduced by the turn-down in demand growth and the first concrete placement was in 1981. When the project was delayed, the engineering was slowed down because of the other massive projects underway. As a result the engineering design has been done in parallel with the construction and commissioning. This has introduced further delay in the schedule.

Safety and licensing issue analysis, construction and commissioning experience have all pointed out problems which have been dealt with by engineering changes. These changes are still being fitted into most key systems. The overall result will be a very complex station. Staffing estimates for maturity have increased from 850 to 1150. This is driven by both actual field activities and massive record, procedure and followup requirements.

We have had a good relationship with the community and look forward to the same in the future. The present discussions on building permits will be resolved.

STATUS

Because of the way we construct four unit stations, Unit 2 and all the common systems are constructed and ready to go. There are many changes which are just being completed. Licensing submissions have yet to be completed and accepted by the AECB. We have reached the point where the AECB is ready to approve loading of fuel. They do not do this until they can see a reasonable sequence and time interval to criticality.

Unit 1 is 90 percent constructed and 30 percent commissioned. It is being slowed down by the Unit 2 work and the limited staff in the operations group. All reactor vessels are in place. Unit 4 turbine generator erection has started. Concrete is still being placed for the upper reactor structure to allow boiler installation later this year. The last major structure of the station, the east fueling facility, is well underway. The Tritium Removal Facility (TRF) has been operated for a short time, is well underway. The Tritium Removal Facility (TRF) has been operated for a short time. The Tritium Removal Facility (TRF) has been operated for a short time. The Tritium Removal Facility (TRF) has been operated for a short time.

Complexity

Darlington will be the most complex station in Ontario Hydro from many points of view. Some features will mean more people to run and maintain it. Some will make it easier to manage. It will theoretically be even more safe than previous stations, but complexity results in more potential for failures so we may have our work cut out to achieve our usual high capacity factors. The complexity is illustrated by a few areas which I will outline:

1. Computers
2. Environmental qualification
3. Protection of equipment from steam
4. Supports of pipe and equipment
5. Shutdown system software.

1. Computers

Computers permeate Darlington in every form from mainframes to programmable controllers in the thousands. Most relay logic has been converted to software. The TRF is controlled from terminals and has no conventional switch panel. We did not, however, go to a station databus system so we still have massive cabling carrying signals to the control centre. Manpower dedicated to computers will be 20 engineers and 15 technicians.

2. Environmental Qualification

A fundamental requirement for safety and safety-related systems is that they will operate under the accident environment and continue to do so for an extended period of time in some cases. This requirement was not fully addressed in the original design. Therefore, over the last few years and still continuing, a very large effort has gone into inspecting systems and equipment for the ability to meet this need. About 30,000 devices are being scrutinized (50 percent to date) and 10 percent have been replaced or modified in some way. The target has been to make the device suitable for the life of the station. Failing that, a maintenance routine is specified to keep up the qualification. A rough estimate of the life cycle costs indicated 60 million dollars and 15 technicians.

3. Steam Protection

Steam protection has made a major difference in the construction of Darlington. All sensitive/safety/electrical equipment is placed in rooms with sealed doors and separate ventilation or air conditioning. It is more difficult to move around the reactor area and may pose problems accessing devices when localized process leaks occur. It has made construction more complicated and contributed to the delays.

4. Supports

Waterhammer analysis and commissioning tests have resulted in extensive changes in the supports of emergency coolant injection piping. Other systems have been affected to a lesser extent. This is a construction and commissioning problem but does not add much to the long term work unless inappropriate restraint causes pipe failures.
5. Shutdown Systems Software

An original agreement was reached with the AECB that software would be used for the SDS logic. At a late stage in the development, serious concerns started to surface in the software world, about the reliability of software and documentation by the AECB. Concluded that there were problems. To make a long story short, for two years up to 70 people in Ontario Hydro and AECL, AECB and various consultants have been working on and discussing software.

What has resulted is much more detailed documentation, clearer coding, an ability to change code at a document level and know that basic reliability has not been affected. On the operations side, we developed a fully automated system for doing statistical numbers of random input test cases which allows us to prove a software set in the field environment to a reliability level of $10^{-6}$ in three days.

The design team, in company with the AECB staff and consultants, have devised a very complex procedure to inspect the coding to a rigorous level. They have proved it on the existing SDS-1 coding and will have to carry it out on the final issue of coding before we on the operations side will get clearance to start up the Unit 2 reactor.

The inspection on the final software revision will take place in August in parallel with our random testing and commissioning of the complete shutdown system field installation.

CHANGES AND AUDITS

Because of the project delays, and design and construction proceeding in parallel, changes at fairly fundamental levels have been inevitable. Commissioning follows closely on the heels of construction and finds things that do not or will not work. The result has been long lists of engineering changes, some of which have to be done before operation but many will be done during early shutdowns.

Throughout the project, audits have been a way of life. Quality assurance has been applied at a very detailed level. Audits have been carried out by the AECB, by our Central Technical and Quality Engineering Departments, by the station QA section, and by the Ontario Hydro Operational Audit Department. They have caught deviations that could have resulted in waste of resources.

STAFF

Safety of our people is top priority and our accident record is good despite the heavy work load. Total staff on the site at the present time is 7,000 including 1,000 operations people. We are short of operators and control technicians which is impacting on the schedule of the later units. The engineering staff is heavily loaded.

The authorized unit operators and shift supervisors, who have to be approved by the AECB, are in place for the first two units and the training program is continuing at high priority. The program to train them has been underway for three years.

The Darlington full scope simulator is in operation at the Eastern Nuclear Training Centre near Pickering. Much work has still to be done on it to bring it up to station level, but it can show many of the events already. The first group of operators and supervisors have exercised the abnormal event procedures this spring. It looks exactly like the normal control room. One excellent feature is the instructors control unit, which is a touch-sensitive graphics CRT. This unit probably portends the actual nuclear generating station control panels of the future.

PRESENT SCHEDULE

Preparations are underway to start loading fuel by hand in Unit 2 on June 13, 1989 and complete it by the end of the month. The fuel handling system is not used to load the fuel but will be ready before startup to handle any failed fuel. Extensive changes to the SDS software will be finished in late August. The first approach to criticality will be in September. Power generation will be achieved in October. We do not anticipate problems with the integrated control programs in the unit computers, nor with the turbine generator, but they have not been exercised to any extent yet in the field.

The other units will follow at about one year intervals. Major activities on Unit 1 in July will be loading of heavy water into the moderator and commencement of the cold flush of the boiler feedwater and condensate systems. An interunit feedwater tie is being installed and we must have the ability to feed from Unit 1 auxiliary boiler feedpump to Unit 1 by the end of 1989. This is to cover loss of normal feed to the boilers and is a licensing commitment.

Unit 3 turnovers will start in June when the unit computers are given to operations by construction. At that point we will be commissioning three units at one time.

When Darlington is fully in service, nuclear units will supply 65 percent of system energy (they now supply 50%).

THE FUTURE

At the station, we are contributing to the plans for future stations and obviously hope that they will be nuclear. Anything else does not meet the diverse demands. A small group headed by one of our managers is looking at our experience in Darlington and other stations, with respect to commissioning and operation in the future. It reports to an Inter-Divisional Task Force in Ontario Hydro. Our technical manager worked with the cost study group which was part of the government review.

Darlington’s experience will show that despite the added complexity and tough sledding we are going through the nuclear option is very much alive but in the next decade we also must keep our fossil and hydraulic resources in top shape.
ABSTRACT

Emerging computer-based information technologies can dramatically increase team productivity on engineering design projects. This paper outlines the development of the CANDID - Integrated Design System currently being used on the CANDU 3 nuclear power station design. The paper describes the current successes and areas under development. The early development of the three dimensional project model and data base in CADDS presented the opportunity to go electronic for all engineering design activity. A distributed general purpose computer ring now in use at AECL supports all aspects of the CANDU 3 project engineering design work beyond the CADDS-3D project model. Design and licensing analysis traditionally done on mainframe computers is now done on the network using workstations and a computer server. In addition to this, all project documentation is produced and managed on the computer ring under an integrated system called the Information Management System (IMS).

INTRODUCTION

AECL has been applying computer-based productivity tools for many years. These have largely been in specific labour-intensive or calculation-intensive areas where single point solutions have large and early payback.

Although several productivity areas still need to be implemented, the largest additional benefits will now come through integration of the various “islands of automation”. The shift in emphasis being on productivity improvement for an entire design team.

For the CANDU 3 reactor design program, CANDID (CANDU Integrated Design) is the term used that describes the way we work in an integrated manner using computer-based tools.

CANDID uses emerging information technologies to make major improvements in personal productivity by:
- shared use of a common body of technical and reference information,
- use of multiple-application workstations,
- use of CADDS 3-D graphics for design tasks,
- provision of ample, low, fixed cost computing resources, and
- use of computers to support work control processes, and
- use of computers for security, storage and communication of technical information.

Implementing the CANDID approach for the Engineering Design Phase of a project requires that many classes of users have access to the CANDID environment to complete their tasks. Typically, these uses can be grouped as follows:

<table>
<thead>
<tr>
<th>USER</th>
<th>TYPICAL TASKS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Designer/drafter</td>
<td>design, drafting, 3-D model, commodity data base</td>
</tr>
<tr>
<td>Specialist</td>
<td>preparation of computer based models, design and analysis, processing and interpretation of results, documentation production and review, communications</td>
</tr>
<tr>
<td>Engineer</td>
<td>data access, routine technical functions, data distribution, document production and procurement</td>
</tr>
<tr>
<td>Administrator</td>
<td>document, data base maintenance, report generation</td>
</tr>
<tr>
<td>Secretary</td>
<td>analysis, communication, correspondence, files</td>
</tr>
<tr>
<td>Project Engineers/</td>
<td>total data access and communication, material expediting</td>
</tr>
<tr>
<td>Manager/Client</td>
<td></td>
</tr>
</tbody>
</table>

In order to service the wide range of tasks associated with the above groups, the CANDID-Integrated Design Program has stressed the development of two integrated, loosely coupled computing environments of:

a. Computer-Aided Design and Drafting (CADD), and
b. General Purpose Computer Ring.

The benefits, status and future developments for these applications are discussed in the following sections.

CANDID-CADD APPLICATIONS

General

At CANDU Operations, CADD is applied to design activities. A 3-D electronic modelling approach, accompanied by
The 3-D modelling software packages for plant equipment, piping layout, structural steel and concrete are now operational. These packages continue to evolve with improved productivity features added regularly by the software vendor. The remaining software packages shown in Figure 2 (shaded by discipline) are being introduced progressively and timed to match the needs of the CANDU 3 design team. Where the software functionality is not commercially available in high impact areas, internal development programs are initiated with the assistance of AECL's Chalk River Nuclear Laboratories (CRNL).

Schematics, has been adopted as the central source of information from which all design drawings and information is extracted. A modelling concept in which all disciplines work together results in the most up-to-date design information available to all designers through a reference file approach. An overview of the CADD 3-D approach is shown in Figure 1.

To make sophisticated design tools available to the designers in all disciplines involved with the CANDU design, integration of between 35 and 50 software packages is being implemented. A schematic depicting the principle software application packages and their relationship to each other is shown in Figure 2.
Benefits

The benefits of CADDS can be quantified on the basis that this technology is used as a labour substituting or labour avoidance tool. The estimated cost of CANDU 3 has been reduced by 7% based on aggressive implementation of CADDS and a minimum of 70% of the drawings prepared using CADD techniques. The specific quantifiable CADDS benefits are summarized as follows:

TABLE 1: SUMMARY OF BENEFITS OF CANDU 3 CADD SYSTEM

<table>
<thead>
<tr>
<th>Gross Benefits</th>
<th>% Plant Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reduced cost of producing drawings</td>
<td>5.2</td>
</tr>
<tr>
<td>Reduced site interferences</td>
<td>3.0</td>
</tr>
<tr>
<td>Material savings</td>
<td>1.1</td>
</tr>
<tr>
<td>Reduced engineering rework</td>
<td>1.0</td>
</tr>
<tr>
<td>Reduced stress analysis costs</td>
<td>0.7</td>
</tr>
<tr>
<td>Reduced as-built program</td>
<td>0.3</td>
</tr>
<tr>
<td>Total Benefits</td>
<td>11.3</td>
</tr>
</tbody>
</table>

CADD computer system operating and capital costs (2.0)
Operator training (1.0)
Software Introduction (1.3)
COSTS
Total Costs (4.3)

NET BENEFIT
Total Net Benefit 7.0

A number of benefits of CADD are difficult to evaluate economically and are commonly presented without cost savings. Some of these benefits are:

- more accurate bids to clients and from sub-contractors.
- more complete and unambiguous design resulting in fewer contractual disputes.
- reduced lead time for tenders. Rapid changes in tender drawings can be made to accommodate clients’ questions.
- three-dimensional electronic models aid the marketing process.
- improved tools to operate a plant (e.g. electronically produced flowsheets can be altered to show the equipment status during maintenance outages).
- rapid transmission of design information to client and site offices.
- easier optimization of economic design of systems and structures.

Development Status

The CANDU 3 plant is being designed through the generation of an electronic 3-D model of the station. To accomplish this, AECL has adopted computer programs and hardware developed by “Intergraph Corporation, Huntsville, Alabama”. A substantial in-house effort is maintained to modify the various Intergraph computer programs to meet AECL requirements, and to make the many Intergraph, AECL and other computer programs work effectively together. Even software that is completely bug-free, effectively designed and suitable for AECL purposes must still have the following work done:

- load data for nuclear materials,
- load data about AECL design guides,
- prepare menus of operator choices,
- program links to AECL analysis programs, and
- design and record AECL work methods.

In some cases, the programs are not available but under development and the supplier can be guided to develop them in ways that are consistent with our needs. In other cases AECL (and more generally the nuclear industry) has software requirements that are not of current interest to the suppliers. As a result, we will have to develop some of the software ourselves. Our plan is to buy what we can, modify what we must and develop software as a last resort.

Current programs underway include the following tasks:

- developing software to route cables according to design rules,
- developing software to perform plant end-to-end wiring that will automatically extract information from computer-based control schematics,
- automating the production of embedded parts (EP) drawings, and linking EPs to the 3-D model, and
- developing programs to significantly decrease the cost of the pipe hanger design task and the concrete detailing tasks.

Completion of these tasks is timed to match the production needs of the CANDU 3 design team.

Operational Status

AECL has achieved outstanding success in piping and structures design. Electronic models are very fast to build. Drawings are extracted from the model in a straightforward manner. However, the effort required for detail drawing annotation has been greater than expected and additional effort is underway to reduce the labour effort during the detailing task.

Integration of tasks allows bills of material, isometrics, stress analysis geometry for piping and structures and geometry for thermohydraulic analysis to be prepared in a fraction of the time compared to conventional manual techniques.

Consistency across the documentation is assured since all information is derived from a common data base. Equipment and structural layout interferences are checked throughout the design process. These interferences are checked by module across all disciplines to ensure the modularization concept can be fully and effectively achieved.

For long lead items such as the Fuelling Machine, where there has been a lack of effective 3-D MCAE (mechanical computer aided engineering) software, a substantial number of machine design drawings are being produced using 2-D drafting techniques. Drawing office staff find that even a 2-D drafting approach is effective for this type of work.
Future Tasks

A principal advantage of the 3-D approach is that the construction documentation is based on a verified design. System layouts can be thoroughly evaluated, options considered and changes made to ensure a functional, error-free design is provided to construction forces. A significant portion of CADDs future development will stress the preparation of unambiguous design information that is organized and formatted in ways that help construction staff.

Effective use of CADD data bases requires that information is entered into the computer by the responsible person and early in the design cycle so that subsequent tasks using the specific design data can have the most benefit. The early involvement of experienced construction staff is, therefore, important to establish proper data structures that allow constructors to take advantage of the CADD information prepared during the design phase. This timely involvement will avoid costly changes to the design data bases after substantial design effort has been completed.

It is expected that CADD will contribute to controlling or reducing costs during the construction process in the following tasks:

- preparation of advanced piping isometric drawings. In addition to dimensional information normally associated with piping isometrics, these drawings can also contain requirements associated with the fabrication process such as sandblasting, welding and inspection. Piping isometrics will also contain fabrication record information.

- assisting turnover of systems from construction to commissioning. A nuclear reactor is designed by system, constructed by area (volume) and commissioned by system. Turnover of systems from construction for commissioning has required much effort to confirm that systems are actually complete. The difficulty is because the boundaries of systems and construction volumes do not coincide, many system and area drawings need to be reviewed to confirm that a system is complete. CADD can help the turnover process by preparing volume boundary envelopes for each construction package as overlays of system models prepared during the design phase. Construction envelopes would be coded progressively as incomplete, complete with exceptions or complete. The CADD computer, using a process like interference checking, would determine if a system passes through any construction volumes that are not fully completed.

- the CADD 3-D model can be used to plan the sequence and size of concrete pours.

- the CADD model can complement or even replace the plastic model. A CADD model can be viewed from all angles and orientations with parts added or removed to suit the particular situation. In fact, many CADD Systems can add motion to 3-D models to achieve animated “walk throughs” of the plant.

- the impact of late design changes on the construction process can be studied.

CANDID—GENERAL PURPOSE COMPUTER RING

General

The General Purpose (GP) computer Ring was developed in support of all non-CADD engineering design work. It was realized that only an integrated approach with team productivity enhancement tools would give the large benefits required to support ambitious CANDU 3 Project cost and schedule reduction goals.

With the increasing performance and decreasing costs of computers, there has been a spontaneous shift toward decentralization driven by the engineer’s need to meet demands for tighter schedules and lower costs. Hardware and software products that are now emerging offer the advantages of distributed computer resources – a high degree of availability (low cost analysis, dedicated CPU’s and large number of seats) while also matching the mainframes in performance and resource management for traditional engineering design applications. The opportunity to work with a single file image for all applications outside the CADD world on a distributed system meant that a whole class of information problems related to distribution, maintenance, and management of copies of documentation and data could be eliminated as staff would always find and work with the reference files.

A fundamental requirement of the GP Ring was that it had to support the many diverse applications previously discussed in the introduction, in a manner that enforces the work flow processes developed for the CANDU 3 program while automating the maintainence of the audit trail. Figure 3 shows the typical distribution of engineering effort that is deployed in a large scale project. The planning area shown refers to project administration and planning. The area shown as other on Figure 3 is largely communications related activities including preparation of approximately 400,000 memos and letters over the project life, meetings consultations and just plain looking for, and retrieving information needed to do a task. The design effort shown in Figure 3 relates to detail design now being done on CADD. As discussed in the previous section, CADD also supports analysis, and documentation efforts. The documentation and analysis estimates are based on typical distribution of effort, while each engineering area has different emphasis. For example in safety, the level of effort on analysis is significantly higher, while in process engineering documentation plays a larger role.

![Figure 3 Typical Distribution of Engineering Effort](image-url)
The CANDID GP Ring targets for work productivity improvement are a 10% reduction in each of the following areas:

a. engineering design and safety analysis,
b. engineering documentation, and
c. communications.

In addition to the above, budgeted costs for mainframe computer analysis have already been reduced by approximately $4 million.

Quantities and Type of Information on the Network. The quantities and types of information for engineering design are grouped in terms of formal documents, correspondence, computer generated lists, computer analysis output and drawings. Table 2 gives a summary of the scale of quantity of information that must be managed and archived and this is followed by a brief explanation of the basis for estimating those quantities.

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>STORAGE CAPACITY BY YEAR INTO PROJECT (GB)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
</tr>
<tr>
<td>Documents</td>
<td>.5</td>
</tr>
<tr>
<td>Computer Lists</td>
<td>.3</td>
</tr>
<tr>
<td>Correspondence</td>
<td>.2</td>
</tr>
<tr>
<td>Computer Output</td>
<td>.8</td>
</tr>
<tr>
<td>Compressed Drawings</td>
<td>.5</td>
</tr>
<tr>
<td>Total Storage Required</td>
<td>2.5</td>
</tr>
</tbody>
</table>

TABLE 2: SUMMARY OF REQUIRED INFORMATION STORAGE CAPACITIES REQUIRED BY PROJECT YEAR

The numbers shown are based on pro-rated estimates of quantities of documentation produced on earlier projects. There are approximately 2150 formal engineering design project documents averaging 50 pages of which it is estimated that approximately 20% of a typical document is made up of figures and tables of average density and 5% are assumed to be complex figures or raster images for estimating system storage requirements.

There are approximately 50,000 pieces of correspondence generated per year on a full project and it is assumed that 20% of these memos and letters letters will contain simple graphics and tables resulting in a need for a total capacity of 2 gigabytes for correspondence by the fourth year of the project.

Computer lists shown in Table 2 refer to computer generated bills of materials and equipment lists.

For computer output shown in Table 2, it is estimated that there are approximately 970,000 pages of computer printout requiring approximately 5 gigabytes of system storage space. Model input files will increase the required storage volume by an additional 20%. Data files saved for plots are assumed to add an additional 30% and copies of actual plots from the data will add an additional 30% to the storage volume requirements.

The total number of engineering design drawings for the project is estimated at 27000, with 20% estimated to be E-sized drawings, 55% D-sized and 25% assumed to B-sized drawings. The storage requirements for a typical E-sized drawing is 8-10 MB, unless a compressed representation is used. Typical compression ratios we have seen are in the range 15 - 25 to one. A compression ratio of 20 to one has been assumed in the table.
environments is also shown, including the Intergraph CADD system, the Business Vax 8530 computer, engineering documentation and illustration services. A CDC 830 computer manages communications to other sites while access to Bell’s DATAPAC by project staff is through the SL1 network. The TCP/IP Ethernet network is the main method of communicating amongst computer systems at the Sheridan Park site.

The CANDU 3 computer ring currently consists of:
- 52 general purpose Apollo workstations with Interleaf documentation software,
- 12 gigabytes of hard disk storage capacity,
- 3 Apollo graphics workstations used for ANSYS and PATRAN finite element model development and interpretation of results,
- 44 PC’s in the process of being connected into the Apollo Domain network with Interleaf software giving PC’s access to the document management system,
- 9 medium duty laser printers as network print servers,
- 1 Exabyte backup system that uses 2.3 gigabyte 8mm cassettes, and
- 1 FPS M64/35 minisupercomputer as a network compute server

This brings the total number of seats to approximately 50% of total project staff. With the Apollo network, each staff member accessing the network gets access to his own files and environment regardless of where that login access is made from. The FPS minisupercomputer that acts as a network batch mode compute server is a Sun workstation as its host computer through which it is connected by an Ethernet TCP/IP link into the network. The minisupercomputer is used mainly for running thermohydraulic safety analysis codes such as CATHENA and FIREBIRD, and commercial finite element codes such as ANSYS.

Analysis Support Status

Office Environment Computing - The project officially moved off the mainframe for safety and finite element analysis when the FPS M64/35 minisupercomputer was declared operational as an office environment network compute server in the fall of 1988. During the first year of evaluations, over $2 million worth of safety analysis was done on the compute server based on commercial mainframe rates. Currently, engineers are not charged for use of the computer, and we are finding that the computer is affecting the way analysis is being done. The high availability helps with schedule deadlines and analysts do not get into computer budget delays when things do not go exactly as expected.

Elimination of Line Printers – Another change in the way compute intensive analysis is being affected by the GP Ring is in the reduction of volume of computer printouts. There is no longer a need for line printers as analysis are expected to save all output electronically. They can print all or any portion of an output using one of the nine laser printers on the network. The laser printers are also used to print graphs of computer results and can also be used to print correspondence and documentation. Saving computer input, printouts and plot files electronically results in a significant reduction in printer noise and eliminates a computer printout, storage and handling problem. The use of a general purpose graphics software package allows the convenient generation of graphical displays of results. This has helped in the migration away from the need to see all computer printouts.

Direct data extractions from the CADD model – This is now being done for setting up geometric models in codes for design and safety thermohydraulic analysis. A program called CEDRA was developed at CANDU Operations for this purpose. CRNL and Intergraph have worked closely to extend this capability into extraction of piping data for input into pipe stress analysis.

Documentation Support Status. Interleaf documentation software has been installed on the workstations and all CANDU 3 Project documentation is now produced on Interleaf systems. Basic training of project staff and document services staff is almost complete. CANDU 3 documentation and information flow models are being developed and built into an Oracle database application to control the flow of Interleaf documentation during the preparation and approval phases in cooperation with Interleaf.

Documentation is about to take on a new role in the design process with the document central to the Information Management System (IMS), the name we have given to the integrated information environment being implemented on the GP Ring. With the integrated documentation environment we are implementing team productivity enhancement capabilities such as:

- distribution by notice of access to the reference document,
- automated distribution and recording of comments, and decisions during the document production cycle,
- logical change tools to assist with control and organization of multi-disciplinary documentation efforts,
- the ability to use input geometry and analysis results directly in documents is useful, and
- shared content documents linked back to their source.

With the new approach, all documents will have a consistent format enforced by the documentation software and consistent use of shared data. By making documentation a part of the design process, there will be no significant bottleneck in communicating the correct information to those who need it when it is needed.

Communications. The project staff now have access to files and documents that are the only copy. Distribution is being implemented where staff are notified of the availability of the reference copy of documents for review by staff. We are looking into mail systems that will reach AECL staff on other computer environments and sites for enhancement of communications.

Information archival, storage and retrieval systems have been under investigation for the past year and it is expected that in the next two months that a system can be selected for trials and implementation. Figure 4 shows a sketch of a mass storage and retrieval module which will have optical archival for all issued documents, drawings, correspondence and computer generated output. The storage module server will be used to control the import of data coming from a scanner.

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and support spontaneous searches of all documentation for specific occurrences sought.

In the documentation environment work will continue to progress on the building of shared data, shared document components, and the development of electronic distribution links to client, regulatory and consultants. The role of the GP Ring in materials management and procurement is under review. Once a significant document and information base has been established, the spontaneous access to that information will become important for marketing efforts including bid preparation. In a multi-project environment, the GP Ring will play a significant role controlling design change efforts, including the preparation and modification of the design documentation to suit site specific requirements.

**SUMMARY**

There are important lessons to be learned in developing and implementing a large integrated computer based system. The first priority is to obtain a corporate commitment to specific goals and objectives as there must be unanimous support on the need for change. Implementing new work methods and work processes requires significant training and personal adjustment which are only readily accepted by staff when they are part of the solution. The need to develop requirements specifications are not only essential to avoid wasted funds and time, but are an effective means of getting those most affected to become willing participants. It is important to develop those requirements based on corporate needs without specific hardware and software products in mind. Challenge the industry with clear requirements and follow up by insisting on in-house evaluations in the intended production environment. Pilots are an necessary part of the process where requirements can not be fully determined in advance or where products are in the process of being released.

Our experience has been that having challenging goals and following a systematic approach has paid off. We feel that our goal of saving 10% of the plant cost on the first project will be fully realized. A lot has been achieved but there is still a long way to go.

The improved quality of our product and increased design effectiveness of engineers with the right tools and training have potentially larger long term benefits in a multi-project environment where limited staff and resources will have to respond to increasing and changing demands. As the available computer–based support technology evolves, there will be additional changes in the way work is done. Having started down the road of change, we will have the staff and experience to capitalize on new opportunities as useful applications emerge.
NEW APPROACHES TO ALARM ANNUNCIATION FOR CANDU POWER PLANTS

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ABSTRACT

CANDU nuclear power plants have a number of relatively advanced alarm annunciation and display features compared to other plants of the 1970's and 1980's. However, alarm overload remains as a significant burden on the operator under some circumstances. Recent dialogue between AECL and CANDU utilities has produced new insights into the nature of the problem and into methods for resolving it. The solutions are multi-faceted, and involve a shift in approach in a number of areas including: staffing of the man/machine interface design teams, structuring of the design process, choice of presentation frameworks and programming languages, alarm categorization strategies, and display ergonomics.

THE CHALLENGE OF ALARM OVERLOAD

As a consequence of the relatively large number of systems, processes, and components which comprise a nuclear power plant, and in accordance with a philosophy of rapid, comprehensive fault detection and correction, nuclear power plants typically employ a large number of alarm variables. Following general process industry practice, annunciation was traditionally based on simple alarm threshold transgression. However, because of the wide range of process conditions encountered in a Nuclear Power Plant operation, circumstances can arise, typically after a safety-related automatic power reduction, where many alarm threshold transgressions may occur but are, for the most part, redundant or without significance. This can give rise to "Alarm Flooding" or "Alarm Overload" situations, which complicate the operator's task of discerning genuine fault indications from spurious ones.

In the nuclear power plant designs of the 1970's and early 1980's, Alarm Overload is a fact of life during plant upsets. This syndrome manifests itself, in conventionally instrumented plants, as an alarm window and status light "Christmas tree". With the introduction of computerized alarm message displays, a variety of computerized conditioning and suppression features have helped. However, the number of alarm points has also been increasing. Hence, under major plant upsets, Alarm Overload can still occur, in the form of flooding of alarm display screens and printouts.

For both conventionally instrumented and computerized plants modus operandi (i.e. operating procedures and training) have been developed to cope with situation. Basically this involves disciplined scrutiny, by the plant operators, of key measurements, indicators, and alarm summaries, after the occurrence of the upset, in order to chart an appropriate course of action. This has made the situation acceptable from a safety standpoint. However, it would be much better to eliminate the alarm overload situation entirely, or reduce it to insignificance. To achieve this, without loss of necessary information to the operators, substantial changes are needed in the alarm presentation schemes. The payoff for such changes includes:

- improved equipment protection,
- improved plant capacity factors, through faster trip recovery,
- reduced operator stress,
- simplified operating procedures and operator training.

The Case of CANDU

Compared with most other plants of the 1970's and 1980's, CANDU plants are highly computerized, with relatively well organized and flexible man-machine interfaces and display systems. However, this has not provided immunity to Alarm Overload.

In the CANDU designs, as exemplified by the CANDU-6 plants commissioned in the early 1980's, alarm annunciation is based on a combination of window alarms, and computer-driven alarm annunciation messages (1). The latter are supplemented by computer-printed alarm message logs and alarm status summaries. Compared to conventionally instrumented LWR's of that vintage this arrangement facilitated a relatively small number of alarm windows, (approximately 240 in total), distributed over the different system panels. Also, it allowed alarm conditioning and suppression features to be built into the computerized portion of the system. The number of alarm points accommodated by the computerized annunciation system was rather high and serviced (approximately):

- 1500 analog input variables,
- 2500 digital (i.e. binary) input variables,
- 800 control-program generated alarms.

This arrangement worked out quite well under "normal" operating conditions, including minor upsets and equipment failures. However, in major plant transients, (e.g. reactor trips, power stepbacks, loss of Class IV power), it yielded flooding of the annunciation CRT's. Although alarm suppression based on alarm priority helped, the CRT
message flow, at its peak, was still too rapid and too voluminous to be followed by the operator.

This meant that the operators had, in these circumstances, to rely on the other plant status indicators, or to consult the paper printouts of the alarm sequence, in order to carry out their monitoring and diagnostic duties. The paper printouts could be very voluminous (e.g. an inch-thick stack of paper, with 5,212 messages, for a three hour period following a loss of Class IV test at Pt. Lepreau GS in 1982). Hence, the diagnostic and monitoring task can be an onerous one.

The problem has been recognized for some time (2), and work at the operating stations, largely within the original system framework, has, over the last few years, improved the situation to some extent. However, a significant degree of alarm overload remains present in major plant transients, and more fundamental changes are necessary in order to resolve the problem.

In short, with computerized alarm handling systems, it is easy to present a very large volume of information which spans many different fault conditions having different degrees of importance and credibility. The challenge is to display only those alarms that are relevant to a particular situation, and to articulate the alarm information that is relevant to the remedial task at hand.

FRESH PERSPECTIVES ON EXISTING CANDU ANNUNCIATION SYSTEMS

Advances in computer system hardware and software technology, and the CANDU-3 design initiative at AECL, have spurred a new dialogue between AECL and the CANDU utilities, in the area of man-machine interface design. This recently initiated dialogue has produced a number of new insights into the nature of the problems with the present CANDU alarm annunciation systems. These insights are described below. Associated solutions are described in the next section.

Design Staffing and Design Process.

Responsibility for the specification of alarm variables and annunciation conditions has traditionally been assigned to the system designers of the various CANDU process control systems (e.g. Reactor Regulating System, Heat transport System, Emergency Core Cooling System, Electrical Power System, etc.). Implementation responsibility, on the other hand, was centralized in a control centre design group, which includes instrumentation and control engineers, panel designers, and computer system designers and programmers.

Following pre-established procedures and design guides, the design of these plants began with system requirements definition, and proceeded to design analysis and the production of system drawings, design manuals, and computer program rules by the system designers. From there it proceeded to control room panel design, software design and programming in the control and instrumentation group. Alarm summary sheets (e.g. high level in the HT condenser) the operator can correctly deduce the source of the heavy water. However, it would be much better to automatically distinguish the category of leak for the operator, and to alarm accordingly. This would substantially reduce the risk of an unnecessary production loss or costly ECC injection due to human error.

Examples such as these, indicate that the operating procedure perspective, plant overview knowledge, and experience of seasoned first operators or shift supervisors are necessary complements to the detailed knowledge of the system designers, if significant improvements in the alarm annunciation systems are to be achieved.

Combinatorial Logic and Conditioning.

Alarm conditioning, i.e. selective suppression of individual alarms based on the status of other plant variables, is used in the presently operating stations for some window annunciators and for some CRT alarm messages. In the case of the window annunciators, this conditioning is accomplished by designing and wiring up relay logic to establish the desired (conditioned) contact input to drive the panels and software. Formal design reviews along the way, with a multidisciplinary cross-section of reviewers, helped to ensure the integrity of this design process, and its product. For safety-related systems, probabilistic safety analysis at the end of the process provided additional assurance that all the alarms necessary for the protection of the public were, in fact, provided. Detailed plant operating procedures were produced by the utilities, as part of the plant commissioning process, after the systems and the design documentation had been delivered to the site.

Under these arrangements, the design process involved a concentration of expertise in the process, electrical, mechanical, control system, and computer design disciplines. This was beneficial from the viewpoint of producing annunciation systems which were generally strong in their fault coverage, and in their detailed technical correctness. Also, the centralization of the panel design and computer design, produced a beneficial uniformity in the form of information presentation to the operator, transcending the individual systems boundaries.

The design process was, however, less strong in integrating the design with plant operating procedures, and operational priorities. For example, a study of emergency operating procedures (2) revealed that information from several different panels and systems had to be accessed by the operator, in the execution of individual procedures, and that regrouping of some of the display information, along procedural lines would be advantageous.

Another example, drawn from Gentilly-2 operator feedback, concerns the absence of direct diagnostic indication for overpressure relief, or valve leaks, from the heat transport condenser into the associated sump tank. Such events have occurred in two instances in Hydro Quebec experience. Alarms on high sump tank level and on "beetle" indication of moisture in the collection system respond to this situation. However, the same alarms may also be triggered by leaks from the fuelling machines. In conjunction with other plant information, (e.g. high temperature and level in the HT condenser) the operator can correctly deduce the source of the heavy water. However, it would be much better to automatically distinguish the category of leak for the operator, and to alarm accordingly. This would substantially reduce the risk of an unnecessary production loss or costly ECC injection due to human error.

Examples such as these, indicate that the operating procedure perspective, plant overview knowledge, and experience of seasoned first operators or shift supervisors are necessary complements to the detailed knowledge of the system designers, if significant improvements in the alarm annunciation systems are to be achieved.
windows. In the case of the CRT alarms, the conditioning can be done in software by setting a conditioning bit in a status word associated with the alarm, and by providing up to three address pointers per alarm which point to the desired conditioning routines. The conditioning routines are programmed to implement the desired conditioning functions, based on the other variables residing in the plant database.

Both of the above settings are highly flexible in respect to the conditioning variables and conditioning logic which can be used. However, both are very cumbersome from a design modification viewpoint, which involves either detailed level hardware logic design or macro/assembler level programming.

A quick examination of the window alarm arrangement and the CRT alarm message lists reveals that a substantial proportion of the window alarms actually do employ conditioning (most often on low reactor power). However, in the computer-based alarm messaging system, conditioning is actually employed in only a very small fraction of the several thousand alarms.

The formulation of individual message conditioning logic tends to be a very detailed level design activity. Hence, in a quick look at the overall alarm inventory, the quantitative extent of opportunities for Alarm Overload reduction by this avenue is not obvious. It does seem clear, however, that a substantial amount of effort would be required, both to identify appropriate conditioning expressions and to implement them.

Alarm annunciation has an intimate relationship with maintenance and operating procedures, which are developed and maintained at the stations, after the initial design of the plant has been completed. Hence much of the insight into the use of conditioning in particular operational settings will tend to emerge only after the initial design has been completed. Software tools which simplify the post-design addition and modification of conditioning are therefore essential features if major improvements are to be achieved in the use of conditioning.

Use of Alarm Categorization.

CRT alarms in the existing CANDU-6 plants are categorized for display purposes according to:

- system group (Safety Systems, Reactor & PHT, Boiler & T/G, Electrical System, Auxiliary), and
- priority ("major", "minor").

The system grouping is used to colour code the alarm text message on the annunciator CRT, and to group alarms for alarm summary printouts. These printouts are available on demand, and display all inputs within the desired group, or for all groups, which are in the alarm state. In addition, similar alarm summary printouts can be requested for the "major" alarms, and for control program generated alarms (by program).

Control Panels in the main control room are also arranged on a system basis (Fig.1), and include the pertinent alarm windows and other status indicators associated with the system. Hence operator attention is directed to "problem" systems in upset conditions.

The formulation of individual message conditioning logic tends to be a very detailed level design activity. Hence, in a quick look at the overall alarm inventory, the quantitative extent of opportunities for Alarm Overload reduction by this avenue is not obvious. It does seem clear, however, that a substantial amount of effort would be required, both to identify appropriate conditioning expressions and to implement them.

A perusal of printed alarm logs from CANDU-6 plant upsets, (reactor trips, stepbacks, loss of Class IV power), shows that approximately 70 to 80% of the messages are in the "minor" category. However, even with suppression ratios of this order, experience shows that the CRT alarm flow, at its height (near the beginning of the transient), is too rapid for the human operator to follow.

The comprehensive printed log, because of its volume, can also be difficult to digest quickly in major upset circumstances. Nevertheless, it has play a useful role in helping to support an on-the-spot
event assessment in the control room, and, often, a more detailed "post-mortem" assessment, later, by the technical unit. Feedback from Gentilly-2 Operators indicates that the comprehensive time-stamped sequential alarm log feature should be preserved, because of its usefulness in diagnosing events which transcend multiple system boundaries. However the feedback also indicates that additional presentation modes, including rapid CRT-based access to alarm history and status, and selective sequential alarm history presentations (i.e. with alarms filtered by system), are perceived as measures which could significantly ease the operator burden.

On balance, much room remains for more selective presentations of alarm status and history, to support the various aspects and time frames, and vantage points of plant operation.

Use of Shape and Pattern Recognition.

Shape and pattern recognition are areas where the human operator excels. This fact is exploited to a modest extent in the existing CANDU annunciation systems.

Window alarms are organized on a by-system basis, and are implemented in a consistent form, i.e. an array of windows at the top of each system panel. This is beneficial in drawing the operator's attention to the general areas of difficulty in plant upset, even if he happens to be too far away to read the message inscribed on particular windows. Also, with training and experience the operator learns to recognize the significance of particular window positions, and combinations of lit window positions.

In the plant systems which have the larger arrays of alarm windows, (notably the shutdown systems and the Emergency Core Cooling system) there is some further geometric order. For example, Shutdown System trip windows are arranged in rows, with the leftmost window in each row indicating a channel trip condition, and with the other windows in the row indicating specific parameter trip conditions contributing to the respective channel trip. Two of three channel trips produce a reactor trip. Hence, the pattern of lit windows, indicates the generic situation severity and the contributory trip parameters.

Panel indicators of system status, such as status lights and electromagnetic indicators, (EMIs), are other forms of status annunciation, which exploit shape and pattern recognition in a modest way. For example, "discrepancy lights", used for a variety of remote-manual motor-operated valves, indicate that a valve stem position differs from the handswitch-demanded position. The location of these lights in physical proximity to the associated handswitch, together with the CANDU "dark panel" design philosophy, makes their significance readily apparent to the operator. In a similar vein, for automatically controlled valves, associated EMI's on the control panels use bars aligned with, or crossing, a flow mimic line, to indicate whether the valve is open or closed.

On the other hand, there are also situations where indicators could exploit shapes and patterns, but do not. For example, a single light per panel is used in the CANDU 6 plants to indicate that one or more of the handswitches on that panel are in an off-normal state, (e.g. a backup pump selected as "off" instead of "standby"). To identify the handswitch in question, a close-up examination of switch positions is needed. It seems clear, that suitable CRT graphic displays or suitable geometric arrangements of panel displays could be used advantageously to more clearly highlight and identify off-normal conditions involving handswitches, or other plant equipment, particularly if the conditions have personnel or equipment safety ramifications.

Despite the relatively high level of uniformity in the form of annunciation, feedback from Gentilly-2 operations personnel has indicated that there is room for improvement in the uniformity of utilization style of annunciation, from one system to another. For example, window annunciation, with many windows and quite pointed information content, (e.g. "Channel J Boiler Level Low") is present in the Reactor Shutdown Systems. However, in the heat transport system, (which is largely controlled, monitored, and annunciated through the station control computers), there are very few annunciation windows, despite the complexity and importance of that system.

In the Electrical system, another system of great importance, there is a moderate window count, (approximately fifteen), with about half the windows being fairly specific (e.g. "class I and II channel A undervoltage"), and about half being "trouble" windows which point in a non-specific way to areas where there is a problem, (e.g. "Generator Control and Synch Circuits Trouble").

The necessary information is, in all three of the above cases, retrievable by the operator. However, a greater consistency, among the different systems, in the style of information presentation would clearly be beneficial.

Overall, more and better use of shapes, patterns, and display geometry in the alarm presentation would be advantageous, as would greater consistency in the style of usage of annunciation features from one system to another.

Handling of Nuisance Alarms.

One of the features of the CANDU CRT-based annunciation is the treatment of "return to normal" messages. These are handled on-screen by changing a one-letter alarm prefix from "A" (for "Annunciation") to "N" (for "Normal"), for the alarm in question. This is done in lieu of writing a new line of text, unless the original message is no longer on the screen, and is helpful in reducing the flow of alarm text on the screen.

The printer log is handled differently, with each change in state producing a new message, for a comprehensive time-stamped alarm history. Each of the state changes produces an audible alarm, to alert the operator. This audible alarm is silenced when the operator pushes an "acknowledge" button.

These alarm state-change handling features have some undesirable side effects in upset or off-normal conditions where one or more alarm variables may be hovering near the alarm threshold. In particular, on the printed log (but not the screen), this situation can produce a continuous flow of "A" and "N" category messages for the same alarm, thereby greatly increasing the alarm message volume, and contributing very little new information. Also, each "N/A"...
reversal activates the audible alarm, making alarm acknowledgement a nuisance. The scale of the problem can be significant: In a reactor trip at one of the CANDU-6’s, in 1983, an analysis of the printed alarm log revealed 1872 “nuisance” alarm messages, out of a total message count of 3191, over a period of 70 minutes after the trip. These nuisance alarms were all attributable to 15 alarm variables.

Gentilly-2 Operations staff report that their remedial actions for nuisance alarms include:

- software “jumpering” of the offending alarms, (i.e. disabling them, individually with a menu-assisted command feature), or
- temporarily patching modified alarm thresholds into the computer (if the offending alarm is of the analog type).

For program generated alarms, similar situations can arise. However, patching in these cases would involve alteration of the control program software, which can have serious adverse consequences if performed incorrectly; hence, for control program “nuisance alarms”, temporary corrective actions are not usually undertaken.

In summary, on-line manual software patching and software jumpering features which are provided in current CANDU’s, are useful as remedies for some plant upset nuisance alarms. However, they are not comprehensive and are far from ideal. Development of more satisfactory solutions to the nuisance alarm problem is desirable.

NEW APPROACHES TO ANNUNCIATION

The identification of problem areas, as described in the preceding section, has stimulated development of solutions, both in retrofit and new plant design contexts. These are described below.

AECL/Utility Co-operative Efforts

Recent co-operation on development of retrofit improvements and enhancements for existing stations includes:

- development of Emergency Operating Procedure (EOP) entry cue alarms, and EOP-related status overview displays (NBEPC and AECL CANDU Operations). The EOP-entry cue alarms use Boolean combinations of plant status information, to infer that execution of a particular EOP is called for. The status overview displays are special CRT displays which consolidate plant information necessary to support the execution of particular EOPs.

- development of a prototype expert system known as SADAU, for root-cause diagnosis of reactor trips (Hydro Quebec and AECL CANDU Operations). This system is based on examination of the potential precursor conditions for SDS1 and SDS2 trips, and is aimed at facilitating rapid recovery from trip, in spurious trip scenarios.

- development of Computerized Emergency Operating Procedure (EOP) access, (NBEPC and AECL Research Co.). Closely related to annunciation although not annunciation per se, this work is aimed at providing rapid computer-based access to EOP procedures. Off-line computerized procedure prototypes have already been produced. Future plans include integration with the on-line plant database, including associated alarms and entry conditions, as well as other plant information.

Co-operation on the design of the CANDU-3 Man-Machine Interface, recently ongoing or planned, includes:

- meetings with utilities in early stages of MMI design to help establish design requirements,
- meetings and survey contributions from Pt. Lepreau operations staff regarding “link analysis” of the control centre, (i.e. man/man and man/machine communication locations and frequencies), to support control centre layout design,
- inclusion of utility staff in various formal and informal design reviews,
- planned inclusion of one or more qualified CANDU operator in the CANDU-3 Control Centre design team.

Provisions for On-site Alterations to Annunciation Features

It is recognized that many annunciation design details (thresholds, alarmed variables, conditioning functions, status summary contents, etc.) must be finalized at the station, and refined and updated throughout station life. Accordingly the design effort will be directed to ensure that man-machine interface design features provide a high level of support for such activities.

In particular, high level, easy to use programming languages, and pre-programmed software features will be used. These will allow station personnel who are not programmers, to make the necessary modifications and additions easily, with a high degree of confidence in the results, and with appropriate control and feedback measures to ensure the continuing integrity of the product.

The current state of the art in computer hardware and software technology, and the precedents for using high level languages in other CANDU plant monitoring and safety system applications (3) readily support this direction for alarm annunciation in the next generation of CANDU systems.

Task-oriented Design Process

Advances in the field of Human Factors Engineering, and a growing general awareness of the need to design the man-machine interface to cater to the strengths and limitations of the human operator, encourage the systematic treatment of the human operator as another, albeit very special, system component. This approach demands the explicit definition the functions of the human element, in the course of the control system design process. In the Canadian regulatory arena, this approach has been endorsed by the AECB’s Advisory Committee on Nuclear
Safety in ACNS-9, a proposed policy statement on human factors requirements in the design and operation of Canadian nuclear facilities.

There are many facets to the modern Human Factors Engineering approach. However, one which stands out prominently in the literature, (IEC Control Room Standard (5), EPRI Guidelines (6)), is an emphasis on clear, systematic task definition, as a foundation for ensuring the compatibility of the control system design with the capabilities of the human operator.

In the CANDU-3, preliminary Emergency Operating Procedures, written as a prelude to man-machine interface and system design work, will provide an up-front operator task definition for safety-critical operator functions, and will serve as the foundation for the associated portions of the man-machine interface design, including annunciation (7).

For the other functions in the plant, a similar approach is being pursued, with system designers defining the respective system-oriented operator functions at an early stage of their design work, in accordance with Human factors design guides and checklists prepared by the control centre design group. Integration of the man-machine interface features from the different systems, along function-oriented lines, will also be done systematically, and is expected to draw heavily on inputs from personnel who are experienced in plant operations.

This design approach will help to ensure that all operator tasks have the support of appropriate plant information displays and controls, including appropriate alarm annunciation features. By being started in the early stages of the system design, this approach will facilitate early and meaningful review input from plant operations groups, and improved integration of alarm handling across system boundaries.

New Alarm Categorization Strategies

The CANDU 3 design approach stresses the explicit identification of the information user(s), the required operator action, and the required time-frame of response, as part of the specification of each alarm.

This will help to ensure the proper organization and presentation of the necessary information to the operator, in an appropriate form and time-frame.

The presentation of the alarm information will be based on this information, with categorization according to:

- system functional status, (i.e. reactor shut down or at power, heat transport system pressurized or cold, Class IV power available or not),

- user category, (i.e. first operator, assistant operators, shift supervisor, technical unit),

- action time, (i.e. action required in less than one hour, less than one shift, more than one shift),

It is expected that these categorizations will be instrumental in:

- allocation of alarm information to "windows", summaries, or other media of expression,

- the organization of alarm status information displays, and

- the specification of alarm suppression, and alarm "filtering" features, which provide the operator or other station personnel with selective views of the alarm status and history.

More Use of Shapes and Patterns

Two main areas of application are contemplated, for increased use of shapes and patterns to facilitate information assimilation by the operator:

- animated plant overview mural mimics (Fig.2), depicting the overall health and status of the critical plant systems, and including numeric values for a small number of basic plant parameters of general significance, (e.g. reactor thermal power expressed in MWth, the thermal cycle efficiency expressed in percent, and the electrical power output expressed in MWe. These mural mimics will help to provide an integrated overview of plant status to the operator and will help to direct his attention to problems that may be ongoing simultaneously in different parts of the plant,

- CRT based, animated, plant status displays, which include features such as:
  - hierarchically structured plant mimics which include prominent indication of alarm states, and provide easy access to related information,
  - methodically organized overview alarm displays (e.g. critical safety function displays), with easy access to more detailed levels of information,
o graphics, icons, and display attributes such as blinking and highlighting, consistently applied and providing suitable emphasis of key information.

These features exploit the strengths of the human operator in shape and pattern recognition, and are expected to be beneficial in ensuring more rapid and more reliable diagnosis and remedy of complex plant upsets.

Renewed Attack on Nuisance Alarms

To a significant extent some of the new status-oriented modes of alarm information presentation outlined above will tend to reduce reliance on CRT message displays and associated alarm history printouts, where nuisance alarms have been most bothersome. Nonetheless, time-stamped alarm histories, (with convenient CRT-based access), are expected to remain an important diagnostic tool. Moreover, careful attention is needed to the handling of nuisance alarms in the other modes of alarm information presentation, particularly in respect to associated audible or visual attention-getting devices.

Features that are planned for purposes of nuisance alarm reduction include more comprehensive and more easy-to-use:

- alarm thresholding functions that vary individual alarm threshold as a function of plant state or power level,
- hysteresis provisions that can readily be tailored to suit individual alarms,
- software patching and jumpering features, (with built-in safeguard measures to avert operator entry errors or oversights), for temporary removal or modification of selected alarms.

CONCLUSIONS

In conclusion, there are some progressive features in the man-machine interface systems of currently operating CANDU’s, but also much room for improvements. In particular, the alarm annunciation system and the information display system can now be significantly improved. In this area, recently started cooperative efforts between AECL and the CANDU utilities have identified some fundamental changes in approach. These promise to significantly lighten the burden of the operator in plant upset conditions.

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REFERENCES


ABSTRACT

The CANDU 3 fuel handling system is a version of the CANDU 6 and Pickering designs based on proven CANDU technology, updated with improvements resulting from on-going development and operating experience.

The CANDU 3 design is based on the following requirements:

- Single-ended fuelling, one fuelling machine
- The fuelling machine dedicated to one reactor unit
- Applied proven CANDU technology
- Competitive capital and operating cost and implementation schedule

One significant aspect of CANDU 3, to meet competitive cost targets is to adopt single-ended refuelling. Previous CANDU plants carry out refuelling by the coordinated operation of two fuelling machines, one accepting used fuel, the other inserting new fuel at opposite ends of a selected fuel channel. Single-ended fuelling permits refuelling with only one fuelling machine, at the downstream end of the fuel channel. This results directly in significant capital cost saving and indirectly to a smaller reactor building.

With the previous CANDU 6 and Pickering systems, the upstream fuelling machine inserts new fuel bundles into the channel. In the central core high-flow channels, hydraulic drag on the fuel bundles is sufficient to move the fuel string along the channel. In the outer, low-flow channels, a flow-assist ram extension (FARE) tool, which is basically a free piston, is inserted by the upstream fuelling machine to create the required impedance to move the fuel string downstream. The outlet end fitting is provided with a shield plug and a channel closure operable by the fuelling machine, as in the previous system, to achieve on-power refuelling.

With single-ended refuelling, each upstream fuel channel end fitting is provided with a resident fuel pusher, which is basically a modified version of the FARE tool, to create the impedance required to move the fuel string. The fuel channels have balanced flow adjusted for equal enthalpy and all channels have a fuel pusher to provide the required impedance to move the fuel string downstream. The outlet end fitting is provided with a shield plug and a channel closure operable by the fuelling machine, as in the previous system, to achieve on-power refuelling.

Magazine capacity is arranged to suit single-ended refuelling allowing the operation to be carried out during a single visit to a fuel channel. Adaptation of the existing Pickering fueling machine head has been accomplished by incorporating modifications restricted mainly to the magazine rotor.

This paper will describe single-ended refuelling and the equipment integral with the refuelling operation. Modifications from present CANDU reactors incorporated to expedite maintenance and operations associated with single-ended refuelling based on proven CANDU experience will also be discussed.

INTRODUCTION

The conventional CANDU refuelling concept involves the coordinated operation of two fuelling machines, connected to the ends of a selected fuel channel. For CANDU 3, the refuelling concept requires the use of only one fuelling machine, at the outlet end of the fuel channel, resulting directly in lower fuel handling capital equipment and operating costs. This also results in a smaller reactor building, resulting in additional cost reduction of the plant.

The CANDU 3 fuel handling concept is based on proven CANDU technology. This is accomplished as follows:

(a) Adaptation of the existing main features of CANDU 6 and Pickering "b" fuelling machine design with a modified magazine rotor, involving addition of two magazine rotor stations. The downstream fuel channel end-fitting is modified to omit the liner tube and to substitute a baffle sleeve at the outlet feeder which incorporates a floating flow-through shield plug, latched spacer plug and a channel closure. Much of the existing fuelling machine and associated hardware is used in its present form.

(b) The new inlet end-fitting requires a latched shield plug and a resident fuel pusher to push fuel towards the outlet end of the channel into the fuelling machine. New simplified coolant axial inlet connections located at the end, rather than radially as in the conventional design, are adopted since refuelling from the inlet end is no longer required.

(c) The fuel pusher is required to push the fuel string towards the outlet end in low flow channels and, in all fuel channels, to push the last two bundles into the fuelling machine magazine. A fuel pusher is resident in each fuel channel at the upstream end. The capability is provided to completely defuel the channel with single-ended refuelling. Sufficient magazine rotor capacity allows refuelling to be carried out in a single visit to the channel.

The major benefits in adopting the single-ended refuelling system are:

(a) A reduction in the reactor building size and the accompanying plant reduction.

(b) Greatly decreasing the amount of fuel handling equipment to fabricate, construct, test, commission, operate and maintain. This significantly reduces costs.

(c) A single access ended fuel channel for refuelling facilitates a factory built and inspected composite modular assembly which includes the pressure tube and calandria tube. This composite modular assembly reduces initial site installation time and cost, and contributes significantly to a shorter duration construction schedule for the plant also resulting in a major cost reduction.

* Author to whom enquires should be sent.
(d) The replacement of all the fuel channels in the reactor core within a 90 day outage becomes attainable by having a modular factory built and inspected fuel channel assembly. This is only achieved by adopting a single-ended refuelling system.

DEVELOPMENT HISTORY

An in depth study was carried out by AECL CANDU Operations in the early 1980's. This study considered the concepts for the design of a fuel pusher and the alternatives for the type of fuelling machine best adapted to the single-ended fuelling design. For the single-ended application of CANDU 3, adaptation of the Pickering/CANDU 6 fuelling machine was recommended as the preferred concept, since it best met the overall requirements as the preferred concept, since it best met the overall requirements of safe and reliable operation with a minimum development effort.

FUELLING SEQUENCE

Dual-Ended Fuel Changing

For a typical CANDU, on-power fuel changing equipment consists of two identical fuelling machines (Figure 1). Each fuelling machine is suspended in a carriage, from tracks on a bridge at each end of the reactor that extends the full width of the reactor face. The bridge traverses vertically and the carriage traverses horizontally to allow access to all the fuel channel end-fittings. Shielding doors separate the reactor vault from the maintenance lock and, when closed, allow access to the fuelling machine in the maintenance lock area, the fuelling machines also have access to the new fuel port to receive new fuel, to the service ports for calibration or maintenance and to the rehearsal facility for testing and check-out operations.

On dual-ended systems, following positioning at the fuel channel end-fittings both fuelling machines move forward to home and lock onto the fuel channel. Each fuelling machine, which is filled with heavy water, is then pressurized to match the heat transport system conditions. The fuelling machines remove the fuel channel closure plug, shield plugs, and store them in the fuelling machine magazines. New fuel is then inserted at the inlet end while used fuel is discharged from the outlet end of the fuel channel.

Two fuel bundles are inserted from each magazine rotor station containing new fuel in the fuelling machine head. Four to eight new fuel bundles are generally inserted on each visit, thus replacing four to eight of the twelve fuel bundles in the fuel channel. Either fuelling machine acts to load or accept fuel depending on the direction of flow in the particular fuel channel being refuelled since refuelling is carried out in the same direction as the coolant flow on CANDU 6 and Pickering reactors.

When the required number of fuel bundles have been inserted, the shield plugs and channel closures are replaced and the closures are leak tested by the fuelling machines. The two fuelling machines then traverse to the used fuel ports where the level of the heavy water in the CANDU 6 fuelling machines is lowered below the discharge snout, and the used fuel is discharged.

With four bundle shift refuelling, the fuelling machine can refuel one channel, and then refuel a second one, before returning to the used fuel ports. For a CANDU 6 reactor, refuelling is carried out on fourteen fuel channels per week, with eight fuel bundles being discharged at each visit to the reactor. The refuelling rate varies proportionately with the size of the reactor.
1 Fuelling machine clamps onto fuel channel

2 Fuelling machine ram removes channel closure

3 Fuelling machine ram removes latched spacer plug and stores it in magazine

4 Fuelling machine ram removes flow-thru shield plug and stores it in magazine

5 Fuelling machine ram discharges fuel bundle pairs and stores them in magazine

6 Fuelling machine ram is in fully retracted position and ready to charge new fuel

7 Fuelling machine ram charges new fuel and original fuel bundle pairs from magazine

8 Fuelling machine replaces shield plug and latched spacer plug

9 Fuelling machine ram removes guide sleeve and replaces channel closure

**FIGURE 2 FUEL MOVEMENT SEQUENCE FOR SINGLE ENDED REFUELLING**
CANDU 3 Single-Ended Fuel Changing

The CANDU 3 reactor may be refuelled on-power using a single fuelling machine located at the outlet end of the reactor fuel channels. The coolant flow through the reactor is in the same direction in all fuel channels. Fuel pushers, resident in the inlet end fitting of each fuel channel, assist in pushing the fuel from the channel by means of the coolant hydraulic flow force.

The CANDU 3 fuelling machine holds a minimum of eight pairs of fuel bundles. This capacity is suitable for a four-bundle shift operation with one visit to the fuel channel during refuelling.

The fuel movement sequence for the CANDU-3 single-ended refuelling is shown diagramatically in Figure 2.

On-power fuel changing is performed under fully automatic and remote control. The fuelling machine locks to the new fuel port and accepts new fuel. The shielding door is opened and the fuelling machine is transported by the carriage along the tracks to the face of the reactor. The fuelling machine is positioned so that it is located opposite the selected fuel channel.

The fuelling machine moves forward and locks onto the fuel channel end fitting to form a leak-tight joint with the channel. The plug from the snout of the machine is removed and the machine is pressurized to the same pressure that exists in the end fitting after a leak test operation.

The fuelling machine removes the channel closure plug and injection flow is established. A guide sleeve is installed and the latched spacer plug is then removed allowing the resident fuel pusher at the inlet end of the channel to move the fuel string and the outlet shield plug towards the fuelling machine. The latched spacer plug is separated from the outlet shield plug and stored in the magazine. A ram adaptor is added to the fuelling machine ram and the shield plug, and irradiated fuel bundles which are separated in pairs, are stored in the fuelling machine magazine rotor. The entire fuel string is removed from the fuel channel during refuelling to allow reinserting of the fuel in any order as required by the fuel management program.

The fuel management dictates the irradiated bundles being sent to the irradiated fuel storage bay, and the arrangement of the fuel string with the new and partially used fuel bundles to be returned to the channel. The fuel string together with outlet shield plug, the latched spacer plug and channel closure are returned to the channel. The fuelling machine installs the snout plug in the snout, checks the cavity for leaks to confirm the channel closure plug seal, and releases from the end fitting. The fuelling machine is then retracted from the end fitting and returned to the fuelling machine maintenance lock where it unloads the irradiated fuel to the fuel bay through the irradiated fuel port.

Figure 3 shows the fuel handling system design layout of the CANDU 3 fuelling machine vault, maintenance lock and fuel transfer system areas.
The fuel changing equipment located in the reactor building includes a partially shielded fuelling machine stored in the fuelling machine maintenance lock and suspended in a transport carriage. The carriage is on tracks connecting the maintenance lock with the reactor outlet vault. A powered shielding door separates the maintenance lock from the reactor vault and, when closed, provides a fully biologically shielded access area for maintenance of the fuelling machine while the reactor is operating at power.

While in the maintenance lock, the fuelling machine can lock to the new fuel port to accept new fuel, to the service port for maintenance or service, or to the irradiated fuel port for unloading irradiated fuel.

FUEL CHANNEL

Refuelling is carried out in the direction of channel coolant flow in CANDU 6 and Pickering reactors. In the central core high-flow channels, hydraulic drag is sufficient to move the fuel string towards the fuelling machine located at the outlet end of the channel. In the outer low-flow channels, a Flow Assisted Ram Extension (FARE) tool is used to create the necessary impendence to move the fuel string. The FARE tool is introduced into the channel during refuelling by the fuelling machine located at the inlet end of the channel.

The conventional fuel channel has a latched shield plug assembly in each end of the fuel channel, located within the vertical planes of the end shield.

In the CANDU 3 design, the outlet end of the channel contains a flow-through shield plug with no latch mechanism which is axially supported by the latched spacer plug at the outlet end of the channel as depicted in Figure 4. The conventional channel closure is retained while the end-fitting liner tube is replaced by a short baffle sleeve at the feeder end. A powered shielding door separates the maintenance lock from the pusher, effects a continuous thrust against the fuel.

At the inlet end of the channel the conventional shield plug is replaced by a combination of a special shield plug and a fuel pusher component. A channel closure is no longer required in the inlet end. The resident fuel pusher replaces the FARE tool which was previously inserted by another fuelling machine. The single-ended fuelling machine must have the capacity to carry the required new fuel for refuelling the fuel channel and also to completely defuel the channel.

The fuel pusher, part of which fits over the inlet shield plug has a cylindrical shape. The channel coolant flow, through an integral orifice in the pusher, effects a continuous thrust against the fuel.

During single-ended fuelling, the channel flow must be maintained at all times within the channel bore up to the feeder outlet tube, for adequate fuel cooling. The flow is also required for the fuel pusher, so that it can propel the entire fuel string down the channel until it rests on the fuel separators at the fuelling machine. That is, the fuel pusher length must subtend the distance between the feeder outlet baffle holes and the fuelling machine separator in order to push the last two fuel bundles into the fuelling machine magazine station. Also, the fuel pusher length is compatible with the two-bundle length of the magazine so that the fuelling machine will have the capability of replacing any fuel pusher. The fuel pusher has the added capability to push the flow-through outlet shield plug with the fuel string.

FUELLING MACHINE

The design of the CANDU 3 fuelling machine is shown in Figure 5 and is adapted from the Pickering 'B/CANDU 6 design which includes:

(a) A standardized manual drive system is incorporated into the 'B' ram, latch ram and the magazine rotor. An improved manual override is fitted to an upgraded separator safety latch. Also incorporated are standardized 'B' and latch ram ball screws with a modular gearbox for easy removal and maintenance of the ballscrew seal packages.

(b) Fuel sensor and separator assemblies to detect when irradiated fuel bundles are being separated from the fuel string are incorporated. The fuelling machine separator assemblies ensure fuel bundle detection is maintained throughout the fuel handling sequence and provides interlock permissives for reliable and safe handling of irradiated fuel, which prevents damage to fuel.

(c) The Pickering 'B/CANDU 6 type of fuel channel end-fitting is included for CANDU 3. The fuelling machine inserts a guidance sleeve to create a single fuel-size bore for handling the complete fuel string.

(d) The Pickering 'B/CANDU 6 type fuelling machine has two positioning rams. A mechanical 'B' ram for handling channel tools and components and a hydraulic 'C' ram, for pushing fuel. This is considered an advantage with the single-ended refuelling operation. The normal method of positioning the fuel string will be by the more accurately controlled mechanical 'B' ram using its precision ballscrew drive system. If the ballscrews malfunction, the hydraulic 'C' ram is available as a "back-up" ram and provides an alternate means of pushing the fuel string back into the channel away from the fuelling machine snout to channel end-fitting interface area.

The CANDU 3 single-ended fuelling machine is equipped with a valve to isolate the fuelling machine ram from the magazine housing assembly. This feature allows ram exchange to be carried out in minimum time and increases the modularization of the ram assembly.

(e) The Pickering 'B/CANDU 6 type fuelling machine uses fuel pushers (i.e., F.A.R.E. tools) in low flow channels. The single-end refuelling concept for CANDU 3 uses fuel pushers residing in each fuel channel inlet end fitting. The fuel pusher concept of operation is proven in the Pickering 'B/CANDU 6 fuelling handling systems. The single-ended fuel channel and hardware for CANDU 3 is designed with fuel pushers that push the full fuel string to the fuelling machine separators allowing complete defuelling of the fuel channel during reactor operation.

(f) Extra magazine stations made to fit within a slightly enlarged fuelling machine magazine housing and rotor allows single-ended four-bundle programs with one visit of the fuelling machine to the fuel channel. Equal spacing of the magazine rotor stations is maintained.

The CANDU 3 fuelling machine, is designed to operate with the following single-ended fuel channel hardware: fuel pusher, latch ram, flow-through outlet shield plug and channel closure.

The fuelling machine design is based on the CANDU 6/Pickering 'B' design modified to accommodate single-ended refuelling by increasing the magazine rotor storage capacity to handle 16 fuel bundles (eight pairs). As well, a valve is added between the rear end of the magazine assembly and the ram assembly at the connecting Grayloc assembly to allow ram replacement even if the channel closure cannot be re-installed.

MAINTENANCE AND SERVICING

All fuel handling equipment, including the control equipment, is accessible for maintenance with the reactor at full power. Maintenance of the fuelling machines and ports is performed in the maintenance lock in the reactor building, with the outlet vault shielding door closed. Servicing may include replacing modular subassemblies on the fuelling machine with spares. A crane in the maintenance lock facilitates servicing the heavy equipment.
A combined ancillary and rehearsal port penetrates the wall of the reactor building and permits replacement of channel closures, ram adapters and shield plugs in the fuelling machine. Another port provides facilities for calibration of ram force loads. Calibration of the ram forces is facilitated by the use of a load cell and an adapter. The adapter enables the ram to exert both advance and retract forces on the load cell, thus measuring the actual forces exerted by the ram assembly: that is, measurement of the pre-set value of each force less any retarding forces related to the ram action.

The ancillary port is also used as a rehearsal facility to check functions of the fuelling machine for correct operation in a low radiation environment.

EQUIPMENT TESTING

The fuelling machine head design is based on proven CANDU technology.

As in all previous CANDU stations, the fuelling machine is assembled and operated in a test facility under simulated reactor conditions with respect to temperatures, pressures and flows before being accepted and delivered to site. The test facility at AECL CANDU Operations' Engineering Laboratory includes hot water loops and a full size fuel channel assembly which is loaded with production fuel and all fuel channel internal hardware to represent the reactor conditions. The tests are conducted under fully automatic computer controlled fuelling operations using operational fuelling sequences.
ABSTRACT

A number of fuel bundles defected during 1984 in the Bruce-3 reactor due to circumferential cracking in the end cap weld zone of outer fuel elements. A COG-funded program has been started to study defect mechanisms and thresholds, and review fuel element specification. Methods of mechanical testing, metallography and stress analysis used in the program are described, and factors studied so far discussed. Mechanical tests include internal pressurization with gas, and loading with an expanding mandrel. Both methods produce the same type of cracking as observed in the in-reactor end cap weld defects. The program has identified the main parameters governing the cracking of the end cap welds. Combining the stress analysis of fuel power histories in-reactor, of axial and radial interferences between pellets and the cladding, and of the load systems used in mechanical tests, makes it possible to assess fuel operational limits (in-reactor) on the basis of out-reactor tests.

INTRODUCTION

In 1984 circumferential cracks developed in the end cap weld zone of outer fuel elements in a number of fuel bundles in Unit 3 of the Bruce Nuclear Generating Station of Ontario Hydro. The defect occurred after the bundle was shifted from a low to a high power position, and was directly related to the condition of the weld zone and to loads caused by pellet-clad interaction (1). Modifications were introduced that corrected the situation; nevertheless, isolated cases of the defect have been detected since.

Our objective is to find how the conditions of the tests compare with the conditions of the fuel in-reactor, i.e. how the results of out-reactor tests can be applied to fuel behaviour in-reactor.

MATERIALS AND METHODS

The Fuel Element

An outer element in the Bruce fuel bundle consists of six components characteristic of CANDU reactor fuel (2): a stack of uranium dioxide pellets is contained inside the Zircaloy-4 sheath provided with a thin graphite layer on its inside surface, resistance-welded end caps at both ends, and brazed spacers and bearing pads on the outside surface. The locations of the bearing pads are axially asymmetric. One of the end bearing pads ("out-board") is close to the end cap, so that the braze heat-affected zone (HAZ) reaches the end of the sheath and overlaps the end cap weld HAZ. The "in-board" bearing pad is further from sheath end, so that there is always as-received (AR) sheath microstructure between the braze and weld HAZ.

Figure 1 shows a schematic of the geometry near the end cap, and defines some of the terms used here. From intuition, we can expect three areas of stress concentrations: re-entrant corners R and R', and discontinuities DW in the weld.

In-reactor, the fuel element is exposed to several types of loads.

Figure 1 shows the loads acting on the sheath/weld/end cap region. Because of the heat generation, the pellets expand radially and axially. Also, they adopt an hourglass shape, i.e. the amount of radial expansion is larger at pellet end-planes, and smaller at its mid-plane. The expanding pellets push the sheath in radial direction, and stretch it in axial direction. Additional loads on the sheath, weld and end cap come from internal gas pressure, coolant pressure, hydraulic drag, and thermal stresses caused by non-uniform temperatures in the sheath and end cap.

The input for the stress analysis included the following parameters describing the features of the fuel element:

- Sheath, end cap, and end cap weld geometry (dimensions);
- Mechanical properties of the sheath AR and braze HAZ sections; of the end cap; and of the end cap weld HAZ;
- as-manufactured, and
- at the start of the power ramp;

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* COG - CANDU Owners' Group. CANDU, CANada-Deuterium - Uranium is AECL-owned protected trademark

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Stress Analysis of the Fuel Element

Approach. To evaluate stresses and strains in the fuel element, we first calculated the pellet displacements and the gas pressure by using the ELESTRES code (4). From this, we calculated the pellet/sheath and pellet/end cap interactions. The results were input to the FEAST code (5) along with the pressures and loads, to calculate the stresses and strains in the sheath/weld/end cap region of the cladding.

Description of Computer Codes Used. ELESTRES (4) models the behaviour of CANDU nuclear fuel elements under normal operating conditions. It models a single element by accounting for the radial and axial variations in stresses and displacements. The constituent models are physically based and include such phenomena as pellet-to-sheath heat transfer; temperature and porosity dependence of thermal conductivity; neutron flux depression; fission gas release; stress-, dose- and temperature-dependent constitutive equations for the sheath. ELESTRES also accounts for variations of pellet local density and for its effect on thermal conductivity and on temperature.

The finite element model for pellet deformation accounts for: thermal, elastic, plastic and creep strains; swelling and densification; pellet cracking; changes of yield strength with temperature. It uses the variable stiffness method for plasticity and creep calculations, and combines it with a modified Runga-Kutta integration scheme for rapid convergence and accuracy.

Code predictions compare well with experimental data for gas release and for sheath strains at pellet mid- and end-plane (4). The database covers linear powers up to 124 kW/m and burnups up to 300 MWh/kg U.

In calculating pellet displacements, ELESTRES accounts for mechanical interactions between the sheath and the pellet. Therefore, these effects need not be considered separately in the calculations for the sheath/weld/end cap part of the cladding.

FEAST (5) uses the finite element method to perform stress analysis on two-dimensional bodies, i.e. either plane-stress or axisymmetric calculations can be done. They can include elastic, plastic, creep and thermal strains. Triangular, constant strain finite elements are used. Plastic strains are calculated from the incremental theory and from the Hencky-von Mises yield function (6). The formulation for creep is similar to that for plastic strain.

For axisymmetric analysis, FEAST results include:
- Radial (r) and axial (z) displacements for each node;
- Radial, hoop, axial, shear (in the r-z plane) and effective components of the stresses and strains for each finite element;
- Principal stresses for each finite element.

Figure 3 shows a typical finite element mesh. The mesh is finer in the regions where we expect sharp concentrations of stresses. The analyzed region is represented by 320-350 triangular finite elements, made from 200-230 nodes. The triangular elements are generally arranged in a hexagonal pattern. A theoretical analysis concluded that for a given
number of finite elements, a hexagonal pattern gives the most accurate results (7). Our experience has confirmed this. In some cases, accuracy improved dramatically when we switched from rectangular to hexagonal arrangement of elements (4).

**FIGURE 3: TYPICAL MESH FOR FINITE ELEMENT ANALYSIS (WITH A FLAW MODELLED IN THE WELD)**

**Finite Element Analysis of Crack Growth**

In addition to the evaluation of stresses and strains, we used finite element analysis to determine the stress intensity factor at crack initiation in internally pressurized test specimens.

We assembled finite element models of typical cross-sections using eight-noded axisymmetric elements. We assigned to each element its material property, based on measurements of local microhardness and on the correlation between microhardness and the flow stress.

We then applied internal pressure to each model, and determined the location of the maximum equivalent stress. Assuming that a crack would initiate at the location of the maximum equivalent stress, we remodelled the meshes to simulate the presence of a flaw. We used degenerate, eight-node, "quarter-point" elements for that. Using the virtual crack extension technique, we determined the energy release rate, and then the stress intensity factor $K$.

**Mechanical Tests: Expanding Mandrel**

We devised a method simulating pellet-clad interaction in the sheath/weld/end cap part of the cladding. In this test, a mandrel imposes radial and/or axial loads on the end cap weld region of the cladding specimen. This is a modification of the expanding mandrel method described previously (8,9).

The mandrels are manufactured from a high thermal expansion material with the thermal expansion coefficient about five times higher than Zircaloy (10), and centreless ground to the required diameter.

Radial straining of the sheath is provided by the difference of thermal expansion of the sheath and the mandrel, minus the initial gap. Axial straining is induced by a bolt which is pre-tightened against the mandrel at room temperature, plus the differential thermal expansion during heating - see Figure 4. The loading bolt is restrained by an external clamp acting against a tight-fitting nut.

**FIGURE 4: SCHEMATIC OF THE EXPANDING MANDREL TEST**

The radial gap, torque and the rotating angle of the bolt after contact are measured at the beginning of the test. We calibrated the correlation between the torque and rotation angle for different diametral clearances.

A big advantage of the method is the possibility to apply different ratios of radial and axial loads. In addition, the test is easy to prepare and conduct. As a disadvantage, the loads can be determined only approximately.

**Test Assembly and Encapsulation.** In preparation for the test, each specimen, a 50 mm long piece of fuel element cladding, is loaded with a mandrel. For axial loading, the nut and bolt are inserted and the external clamp fastened. The axial load is set by the torque wrench.

The specimens are then encapsulated in Pyrex capsules. For the tests of unirradiated material, the capsules are evacuated and flame sealed. For the tests of irradiated material in the hot cells, we developed re-usable capsules employing O-ring joints.

To test in the presence of iodine, we use a small vial containing a measured quantity of iodine. The vial is placed in the capsule and is broken at the beginning of the test.

**Testing, Inspection and Examination.** The capsules are placed in a furnace at 300°C and inspected periodically - every hour during the first day, and daily until the end of the test. Failure of the specimen is established by the presence of a visual crack (Figure 5).

Following the test, specimens not displaying visual cracks are leak tested under low air pressure. Optical microscopy is then used to study the cross-sections for cracks, or scanning electron microscopy is applied to fracture surfaces.

* B. Leitch, AECL-RC/WNRE worked on this part of the program.
FIGURE 5: TYPICAL CRACK PRODUCED BY COMBINED AXIAL AND RADIAL LOADING IN IODINE AT 300°C

Mechanical Tests: Pressurized Cladding

Tests with internally pressurized specimens of fuel cladding offer several advantages:

- The loads are well defined, and remain relatively constant during the tests; and
- The stresses at the critical spot or crack tip can be calculated.

This is important if the stress state at the crack tip is needed to interpret the test results. As the crack grows, the stress intensity factor at the tip increases because the load remains constant. Thus the technique measures the threshold of crack initiation.

There are disadvantages, too, if compared with the expanding mandrel test. The test arrangement is more complicated; and it is more difficult to change the load system.

Test Assembly. The specimen is about 130 mm long. It has the standard end cap and end cap weld at one end, and a zirconium plug welded at the other. The plug has a small tube leading to the interior of the specimen - see Figure 6.

FIGURE 6: SCHEMATIC OF THE TEST SPECIMEN FOR INTERNAL PRESSURIZATION TEST

A calculated quantity of argon is introduced into the specimen through the small tube. The tube is then sealed by TIG welding. The quantity of argon is such as to produce the required pressure at the test temperature of 300°C.

To prevent the ballooning of the sheath, and thus uncontrolled decrease of internal pressure during the test, we put a tightly fitting steel retainer on the sheath. The presence of the retainer affects the loading system of the test. We considered this factor in the stress analysis.

Iodine is introduced inside the specimen by a method similar to the one described in the previous Section. We use 10 mg of iodine per specimen.

Testing, Inspection and Examination. Each specimen with its retainer is placed inside an argon-filled stainless steel container. The pressure in the container is monitored; an increase indicates a leaking specimen.

The inspection and examination after the test is as described for the specimens tested with the expanding mandrel.

Test Material and Its Treatment

So far, we have tested the effect of the following factors affecting the condition of the standard cladding material:

- As-manufactured microstructure of the out-board and in-board ends of fuel element cladding. Braze HAZ is located next to the weld HAZ in the first case, and AR microstructures in the second.
- Geometry of the re-entrant corner R (Figure 1). In most tests up to now, the radius of curvature was approximately 10 μm, and in a limited number of cases 250 to 300 μm.
- Burnup. We first used specimens of unirradiated cladding to develop the methods and get the basic information. Tests on irradiated cladding (burnup of 100 MWh/kg U and higher) have been started. This paper reports on the data obtained with unirradiated material only.
- Presence of hydrides or deuterides in the microstructure, and heat treatment of the specimens (see below).

Hydrides and Deuterides in the Microstructure. The content of deuterium and hydrogen in the cladding of discharged fuel elements is usually high enough to leave some deuterides undissolved at the testing temperature of 300°C. Such specimens can therefore be used to study the effect of deuterides/hydrides (such as DHC) without further treatment.

Hydrogen content in unirradiated cladding is so low that it all dissolves in the solid solution below 300°C. To study the role of hydrides in unirradiated material, we hydrided a number of specimens as follows. We introduced hydrogen into the surface of the specimens either electrolytically at 80°C, or by gaseous absorption at 400°C. Then, we homogenized the specimens in vacuum, in the first case for 168 hours at 300 or 350°C, in the second at 400°C for 100 hours. Thus, we obtained uniform hydrogen contents of approximately 50, 100 and 300 ppm.

For comparison, we heat treated some unirradiated specimens with low as-manufactured hydrogen the same way (400°C for 100 hours), but without the addition of hydrogen.

To study the effect of re-orientation of hydrides in high stress regions, we applied temperature cycling under stress to some of the unirradiated pre-hydrided specimens. We loaded them with the
expanding mandrel and temperature cycled between 300 and 100°C once a day at a cooling rate of 2°C/minute. This was considered optimum for hydride redistribution (11) and compares well with the cooling rates in power reactors.

DISCUSSION OF METHODS AND RESULTS

To discuss how well the tests simulate fuel conditions in reactor we will start by comparing the stress systems in out-reactor test specimens and in fuel cladding in-reactor. Then, we will consider the morphology and microstructure of fractures obtained in the test specimens and in fuel defects. Finally, we will discuss the defect thresholds obtained in the tests so far.

There is one important difference between fuel in-reactor and the out-reactor tests we are evaluating here: the effect of irradiation. Defect thresholds applicable to fuel in-reactor will be determined with the use of irradiated material. However, some effects such as stress configuration can be simulated by unirradiated material. Moreover, the behaviour of unirradiated material is representative of some fuel in-reactor, e.g. of fresh fuel shortly after loading in the reactor. Therefore, testing of unirradiated specimens is an important part of the program.

Stress Systems

We chose the out-reactor testing methods for this program on the basis of past experience including the analysis of Bruce-3 fuel defects. We compared the loading conditions of the test specimens and of fuel elements in-reactor with the use of stress analysis methods described in the preceding Section. The results are summarized in Figures 7 to 10 and in Table 1.

Figure 7 shows the loading conditions used in out-reactor tests, expressed as stresses in the sheath remote from the end cap weld. Typical cases we analyzed are listed in Table 1.

We used the "reference" power history in Figure 2 for the initial stress analysis of fuel cladding in-reactor. As the next step, we completed a "sensitivity study" exploring the effect on stresses and strains, of changes in the values of single parameters or of their combinations. Table 1 documents eight such cases.

Three parameters appear to be important for the state of stresses, both for the test specimens and for the cladding in-reactor:

- Pressure difference \( \Delta P = P_T - P_Q \), or fission gas pressure \( P_F \). Coolant pressure \( P_Q \) provides a compressive component to the stresses at the re-entrant corner. This factor is absent in the expanding mandrel tests, and of course in the internally pressurized tests. A highly tri-axial stress system can develop with fission gas overpressure.
- Distance \( L \) between the weld and the end of the mandrel, or the nearest ridge on the sheath. High tri-axial stresses can be expected in the weld if there is radial interference close to the re-entrant corner (if \( L \) is small).
- Radius of the re-entrant corner.

Based on the results of stress analysis, we can consider the stress systems of the out-reactor tests and of pellet-clad interaction sufficiently similar.

Figure 8 describes the conditions of the expanding mandrel test in more detail. For axial loads, Figure 8a illustrates how the onset of plastic
yielding relieves the loads at the inner surface of the sheath near the weld. Figure 8b shows the case of radial expansion, when the mandrel is axially too far away from the re-entrant corner. Under these conditions, high stresses and strains sufficient for crack growth may develop in the sheath earlier than at the weld.

Figure 8 shows the effect of outside overpressure. Figures 9 and 10 show the effect of the re-entrant corner radius, another of the important parameters.

It is worth noting that the internally pressurized test reproduces very well conditions of fuel in-reactor under fission gas overpressure. This may be important, e.g. for high burnup studies.

**Microstructures**

Location and shape of cracks is similar in in-reactor defects of the Bruce-3 excursion and in our out-reactor tests. The following photographs document this.
In the Bruce-3 excursion, very few defects were caused by cracking at the in-board end. Figures 12a and 12b show cracks obtained during the expanding mandrel tests at the in-board welds of the same design as in the Bruce-3 defects. The growing crack crosses through zones of hydride platelets oriented normal to the radial direction. Obviously, hydrides act here, either directly or by affecting the stress fields, as barrier for the growing crack. This agrees with earlier experience with hydrides in AR tubing (12).

For as-manufactured as well as pre-hydrided material tested in iodine vapour, we have repeatedly identified the cracking mechanism as SCC – see Figure 13a for an example of SCC fracture surface. However, there is also evidence of DHC in our tests, such as in the fracture of a pre-hydrided out-board weld in Figure 13b.

The orientation of hydrides after temperature cycling is shown in Figure 14. These tests are still in progress. This treatment has not been found, so far, to have a large effect.
Resistance to Crack Growth

Figure 15 summarizes the results of internally pressurized tests for one type of welds (R>10 μm). For this case, the defect threshold is at the pressure of 20 MPa. The difference between the out-board and in-board ends, especially in the presence of hydrides, is obvious, and so is the effect of iodine. Two conditions, labelled "(H)" and "(H+I)" agree with the behaviour of the fuel during the Bruce-3 excursion. One of them, the "(H+I)" option, develops cracks at lower loads, and therefore is a more likely candidate for the mechanism. The "(I)" case does not qualify because of the similar resistance to SCC, found for the in-board and out-board welds in the absence of hydrides. This would not match the Bruce-3 experience, where nearly all defects occurred at the out-board side of the fuel element.

As stated earlier, this may be modified if irradiated cladding is tested. Also, the difference between the deuteriding rates in the braze HAZ and AR microstructures of the sheath may introduce an additional factor.

Expanding mandrel tests of the in-board material give similar correlations. In addition to results as in Figure 15, we found an improvement of the resistance to SCC after the 400°C annealing.

CONCLUSION

1. The results of stress analysis demonstrate that the out-reactor tests adequately simulate the stress systems of the pellet/clad interaction. The internally pressurized test can also be used to study fuel behaviour under fission gas overpressure.

Pressure difference, distance between the weld re-entrant corner and the nearest pellet/sheath radial interference, and the radius of the re-entrant corner have been identified as significant parameters influencing the stresses in the critical part of the weld.

2. Metallography confirmed similar morphology and character of cracks in the out-reactor test specimens and in-reactor defects.

3. The defect threshold of unirradiated welds with >10 μm radius of the re-entrant corner is 20 MPa pressure (internally pressurized tests with restrainer), if tested in the presence of iodine, with or without hydrides in the microstructure.

Significant factors influencing the resistance to crack growth are: sheath texture combined with the presence of hydrides and presence of iodine.

4. The agreement between out-reactor tests and reactor fuel experience is encouraging. It gives us necessary confidence to work on the more difficult task of the program: to study the properties of the irradiated end cap welds.

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IRRADIATION-INDUCED DEFORMATION IN CANDU® FUEL CHANNELS

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ABSTRACT

Irradiation damage due to neutrons causes plastic deformation of CANDU® fuel channels via creep and growth processes. There is a need to predict the resulting strains over the reactor lifetime. This paper gives a simple description of the deformation mechanisms on which these predictions are based.

INTRODUCTION

In the CANDU® system, the zirconium alloy fuel channel, comprising the pressure tube and the calandria tube, is exposed to the full fast neutron flux during reactor operation. It is known that the primary effect of the energetic neutrons is to modify the microstructure of the fuel channel components, and hence their mechanical properties. As the overall result is a general deterioration of the properties, we refer to this phenomenon as "radiation damage".

The major engineering consequence of radiation damage is that the fuel channel components become distorted, in a time-dependent fashion. This is usually referred to as radiation-induced deformation. The purpose of this paper is to review our current understanding of radiation-induced deformation in a form which is accessible to the non-expert in this field. Further details will be found elsewhere [1].

RADIATION-INDUCED DEFORMATION DEFINED

When a small tensile stress is applied to a metal, a correspondingly small elongation strain results instantaneously. If the stress is removed, the strain is fully recovered; this is elastic deformation. If the stress is increased, the strain increases until the yield stress is reached. At this point, irreversible, rapid, plastic deformation occurs and the resulting shape change is essentially permanent. Many design engineering codes require that the stress experienced by the structure must be less than 2/3 of the yield strength. In practice, it is found that, for stress levels considerably below the yield stress, plastic strain will gradually accumulate with time. Such deformation is termed creep. The rate at which creep occurs often defines the operating lifetime of the structure, which is limited by a total allowable strain.

It has been found that the creep rate increases in the presence of radiation. In addition, irradiation introduces other deformation modes. Thus, under zero stress, zirconium will undergo a shape change, called "growth". This does not occur in the absence of irradiation. Deformation via creep and growth are volume-conserving. Irradiation can also produce a fine dispersion of internal voids. This gives rise to a bulk swelling under irradiation.

These deformation modes depend on the existence of defects within the metallic lattice. We will examine the nature of these defects and how they proliferate under the influence of irradiation.

THE FORMATION OF POINT DEFECTS UNDER IRRADIATION

The exposure of zirconium to energetic neutrons causes atoms to be displaced from their normal lattice positions. (In a single year of reactor operation, every zirconium atom has been displaced at least once, on average.) As a result, two kinds of point defects are created. The displaced atom is called an "interstitial" and the corresponding empty lattice site is called a "vacancy", as represented schematically in Figure 1.

![Figure 1: The Production of Vacancies (V) and Interstitials (I) by Irradiation](image)

The displacement process produces equal numbers of vacancies and interstitials, initially grouped into high-density cascades, containing about 1000 displaced atoms. Within the cascade, a large number of vacancies and interstitials recombine in very short
time periods. Following this period of mutual annihilation, the surviving fraction (~ 0.5) of the point defects are free to migrate (via diffusion) throughout the crystal until they are absorbed at other types of crystal defects (described below), generally referred to as "sinks". It is the absorption of point defects at these sinks that gives rise to the deformation strains.

THE EXISTENCE OF OTHER LATTICE DEFECTS

In addition to the point defects described above, there are other types of defects that exist in any metal. These are responsible for the mechanical behaviour of a metal, particularly its deformation properties. Within the context of this paper, two of these defects hold prominence:

1. The Grain Boundary. Metals are generally in the form of a polycrystalline assembly. Each grain, or crystal, abuts its neighbour along a planar interface called the grain boundary. The orientation of the grain (as measured by a specific direction related to the major axes of the crystal) changes across the grain boundary. The degree of randomness of grains in a polycrystal can be determined using X-ray diffraction techniques and is represented by a parameter called the crystallographic texture.

In zirconium alloys, the crystal structure is hexagonal-close-packed (see later). Such a crystal class is highly anisotropic in its deformation modes. Because the orientation of the grains in a polycrystalline zirconium component is typically non-random (i.e., highly textured), the macroscopic deformation properties are also highly anisotropic. This feature of zirconium alloys has important consequences for the radiation-induced deformation properties of the fuel channel. In addition, the grain boundaries can act as sinks for the point defects and therefore plays an important role in irradiation-induced deformation processes.

2. The Dislocation. In general, it is the presence of dislocations that dominates the deformation behaviour of a metal. The basic concept is that a dislocation represents a discontinuity in the regular lattice and allows deformation to occur.

Figure 2(a) shows the arrangement of atoms around the dislocation. As can be seen, there is an extra half plane of atoms associated with the dislocation, which disrupts the local order of the lattice. The dislocation per se is the line which marks the bottom edge of the extra half plane. Figure 2 also shows the sequential movement (induced by an applied shear stress) of the dislocation from left to right, to produce a unit step of strain or slip: the crystal in Figure 2(c) is now sheared by one unit of lattice spacing. This is a case of plastic deformation by slip, often referred to as dislocation glide.

As with the grain boundary, the dislocation line is also a potential sink for point defects and plays a primary role in irradiation-induced deformation.

ORIGINS OF RADIATION-INDUCED STRAIN

Having described the basic configuration of point defects, grain boundaries and dislocations, we can now examine the ways in which these defects can produce a deformation strain during irradiation:

FIGURE 2: THE MOVEMENT OF A DISLOCATION TO PRODUCE A UNIT SHEAR STRAIN

Void-Induced Swelling

Strain results when the vacancies agglomerate to form microscopic, spherical voids in the material. The net effect is that the density of the metal decreases. The resulting increase in volume causes macroscopic swelling to occur. Void-induced swelling is a common deformation mode in steels. Although electron microscopy studies of irradiated zirconium have revealed the presence of some voids, their number is insufficient to cause a significant volume increase.

Dislocation Climb

When point defects are absorbed at a dislocation, the dislocation will undergo a motion called climb. This is depicted in Figure 3, which shows the extra half plane of atoms in Figure 2. When interstitials migrate to the dislocation, the addition of atoms causes the half plane to be extended (the dislocation climbs downward), as shown in Figure 3(a). Conversely, Figure 3(b) shows the upward climb of the dislocation when atoms are removed by vacancy absorption. The climb of a dislocation causes a strain, i.e., deformation ensues. This is represented by the large arrows in Figure 3. As interstitials are absorbed to cause downward climb (Figure 3(a)), the crystal extends in the direction of the arrows. In contrast, as vacancies are absorbed, the crystal contracts, as indicated in Figure 3(b).

Dislocation Climb and Glide

In Figure 2, we saw how deformation will occur when a dislocation undergoes glide in its slip plane. The amount of glide that occurs can be limited by
obstacles encountered by the dislocation. For example, when a dislocation is held up at a shear-resistant precipitate; deformation ceases. However, the obstacle can be overcome by the dislocation absorbing point defects and climbing onto a new glide plane which circumvents the obstacle and allows deformation to continue. Note that the same result is obtained whether the dislocation climbs upwards (absorption of vacancies) or downwards (absorption of interstitials).

Thus, in the presence of irradiation-induced point defects, dislocations are given an extra degree of freedom through climb. This, itself, results in deformation and also assists dislocations to glide by overcoming obstacles. This is the basic reason why deformation is enhanced by irradiation.

Formation of Dislocation Loops

A striking feature of the microstructure of irradiated metals is the existence of circular dislocation loops. An electron micrograph of such loops is shown in Figure 4. They are a major source of radiation-induced strain in zirconium alloys. The origin of this strain is depicted in Figure 5. The dislocation loop in Figure 5(a) is shown in cross section in Figures 5(b) and 5(c), where the layers of atomic planes are indicated. In Figure 5(b), a large number of vacancies have collected into a two-dimensional "raft", thus eliminating a circular disc of atoms. The inward collapse of the adjacent atom layers causes a contraction strain, indicated by the arrows. That this defect is a dislocation becomes apparent when Figure 5(b) is compared with Figure 3 (there are now two extra half planes of atoms).

As more vacancies migrate to, and are absorbed at, the vacancy-type dislocation, the loop expands and the strain increases. This is exactly the same process as dislocation climb, discussed above. Similarly, as more interstitials are absorbed by the interstitial-type dislocation (Figure 5(c)), the extension strain increases.
PHENOMENOLOGICAL OBSERVATIONS

The manifestations of radiation deformation modes that hold engineering significance for the CANDU fuel channel are as follows:

1. The diameter of the pressure tube increases with time. This mainly comprises creep (under the action of the coolant pressure), growth contributing in a minor fashion.

2. The pressure tube elongates. This is largely due to radiation growth, although a creep contribution is also present. In contrast, the elongation of the calandria tube over the reactor lifetime is expected to be minor.

3. The diameter of the calandria tube decreases by creep under the action of the hydrostatic pressure of the surrounding moderator.

4. The pressure tube undergoes creep sag within the unsupported span between the garter springs (exacerbated by misplacement of the garter springs), eventually making contact with the calandria tube.

5. The total fuel channel assembly sags under its own weight (including the fuel bundles). This is controlled by the creep rate of the calandria tube, which has greater strength due to its low operating temperature.

All of the above dimensional changes are linked to the fact that the deformation of zirconium components is highly anisotropic. This arises from the intrinsic crystallographic anisotropy of the zirconium lattice, much of which is preserved at the macroscopic level by the intense texture developed during the manufacturing process.

The crystalline structure of zirconium is hexagonal-close-packed (hcp). "Close-packed" refers to the fact that the zirconium atoms are stacked in an arrangement which minimizes free space. "Hexagonal" refers to the overall geometry of the stacked atoms. Unlike many other metals (e.g., stainless steels), which have a cubic structure, the hcp crystal is anisotropic in its mechanical properties. The significance of the anisotropy becomes apparent when it is realized that stainless steel does not undergo radiation growth, and its creep properties are generally isotropic.

A single crystal unit, showing hexagonal symmetry, is shown in Figure 6. Figure 6(a) shows the major crystallographic directions, \( \langle c \rangle \) and \( \langle a \rangle \). Figures 6(b) and 6(c) show the planes referred to as basal and prismatic. It is well established that radiation growth occurs by elongation along the \( \langle a \rangle \)-direction and shrinkage along the \( \langle c \rangle \)-direction. This will occur if, for example, atom layers are added to prism planes (via the growth of interstitial dislocation loops lying in the prism planes) and/or removing atom layers from basal planes (via the growth of vacancy loops lying in the basal planes), i.e., a combination of Figures 5(b) and (c).

Although this is a simple concept, the detailed mechanisms by which this occurs are complex and are at the focus of understanding the process. Electron microscopy has confirmed that irradiated zirconium contains both interstitial and vacancy loops. This is very different from the cubic materials, where only interstitial loops are observed. The problem of explaining the co-existence of both types of loops has been a primary objective of our research program.

PREFERENTIAL POINT DEFECT ABSORPTION AT SINKS

A crucial concept is that the strain mechanisms involving point defect absorption at sinks require the diffusional migration of vacancies and interstitials to be imbalanced. Thus, because the strain associated with the absorption of each vacancy (a removal of an atom) is equal and opposite to the absorption of each interstitial (an addition of an atom), if vacancies and interstitials arrive at a sink in equal numbers, there will be no net strain (e.g., dislocations will not climb, voids and dislocation loops will not form). For this reason, a fundamental requirement for radiation-induced deformation is that the rate of arrival of vacancies at a sink must be different to that of interstitials. Seeking an explanation for this has been the nub of the development of theoretical understanding.

The diffusivity of interstitials is generally a lot higher than that of vacancies. Thus, in the earlier stages of irradiation, interstitials will arrive at all sinks, at a rate in excess of vacancies. However, as the vacancy concentration increases, a self-balancing steady state develops where the flux of point defects to the sinks is identical for both vacancies and interstitials (and equal to the production rate). This would result in mutual cancellation and the sink would produce no net strain. There must be some other factor that causes some differential, a partitioning, between the vacancy and interstitial flux.

The classical explanation for this lies in secondary "drift" terms which enter the flux equations. The origin for this is found in the
elastjc interaction of the point defects and the stress field associated with some types of sinks. This is particularly true of the dislocations. It so happens that the drift term is much greater for the interstitial than the vacancy. The net result is that the dislocation shows a preference for interstitials, i.e., the diffusive flux of interstitials will be much higher than that of the vacancies and the dislocation will climb by extending its extra half plane (addition of atoms). We say that the dislocation sink is "biased" towards interstitials.

In order that interstitial absorbing climb can continue indefinitely, the excess vacancies must be absorbed at some other sink. This can occur at a sink which is biased towards vacancies or, more typically, to a sink which has no bias at all; this is termed a neutral sink. A neutral sink is one that has no stress field associated with it. Examples would be a grain boundary, or a void. Thus, in the presence of an interstitial-biased sink (a dislocation), there is a partitioning of point defects, where interstitials are preferentially absorbed at the dislocation and, by default, the neutral sink (e.g., a grain boundary) absorbs the corresponding excess vacancies.

It is important to point out that within this theory of the partitioning of point defects to sinks, dislocations are always preferentially biased toward interstitials. Consequently, any interstitial-type loops will grow by interstitial absorption, while any vacancy-type loops present will shrink, and be annihilated, by interstitial absorption. This was the position of the so-called "conventional" theory up to about 1979. More recent developments in understanding the interstitial-type dislocations have introduced significant modifications to this overall picture, as discussed below.

**A SURVEY OF RADIATION DEFORMATION MECHANISMS PUBLISHED PRIOR TO 1979**

Radiation-induced deformation has been studied, from a mechanistic point of view, for several decades (starting in about 1955, when deformation was observed to accompany fission in uranium). Consequently, there is an abundance of theoretical models in the literature. A review of the understanding of these up to 1979 will be found in the proceedings of the conference on creep and growth, held at CRNL [2].

As growth is a major contributor to the radiation deformation of zirconium, much of the theoretical attention has been devoted to growth mechanisms. Early work attributed the observation of \(<a>\)-axis expansion and \(<c>\)-axis contraction to the formation of interstitial loops on prism planes and vacancy loops on basal planes. A consequence of this model is that crystallographically textured polycrystals would grow according to the law:

\[
\dot{\varepsilon} = 1 - 3F \tag{1}
\]

where \(\dot{\varepsilon}\) is the growth rate and \(F\) is the resolved fraction of basal-plane normals along the measurement direction.

It is important to note that for macroscopic growth to occur in polycrystalline material, there must be some crystallographic texture present. Thus, if the grains were randomly orientated, the average growth strain of the multi-grain assembly would equal zero. In such a case, \(F = 1/3\) and equation (1) would predict zero growth. For the pressure tube and calandria tube, growth is at a maximum along the tube axis, where \(F \approx 0.07\).

This explanation was disputed by subsequent electron microscopy that showed no evidence of \(<c>\)-component dislocation loops on the basal planes. To overcome this difficulty, it was suggested that growth could result from a net flow of interstitials to all \(<c>\>-type dislocations (via the interstitial bias effect), accompanied by a net flow of vacancies to the grain boundaries, which act as neutral sinks. For equiaxed grains, this would also result in the growth law given by Equation (1).

**RECENT (POST-1979) DEVELOPMENTS IN UNDERSTANDING**

Since the CRNL Conference in 1979, considerable progress has been made in the understanding of creep and growth phenomena in zirconium: a good summary of this will be found in the proceedings of the 1987 conference on creep and growth held at Hecla Island [3]. It is instructive to note that many of these developments have focussed on the intrinsic anisotropy of hexagonal-close-packed zirconium.

**Key Experimental Observations**

The growth behaviour of annealed material is shown, schematically, in Figure 7. Three stages can be distinguished. The first stage, an initial rapid growth, is followed by a steady-state period, which at over temperatures approaches a saturation condition, where the strain rate is close to zero. However, at higher temperatures, the growth rate accelerates into a third stage, where the strain rate increases with fluence and is referred to as "growth breakaway". From a practical engineering point of view, it is important to know whether growth breakaway will occur in the CANDU calandria tubes.

**FIGURE 7: RADIATION-GROWTH IN ANNEALED ZIRCALOY**

During the first two stages of growth, electron microscopy has revealed the development of \(<c>\>-type dislocation loops on prism planes. The acceleration of growth in the third stage is accompanied by the presence of \(<c>\>-component dislocations which have been formed by the growth of large basal plane vacancy loops. The evolution of dislocation loops during the three stages of growth is summarized in Figure 7. From this it can be concluded that growth breakaway, or rapid growth in general, requires the formation of \(<c>\>-component vacancy loops on basal planes. This important observation is consistent with the behaviour of heavily cold-worked Zircaloy or Zr-2.5\% Nb (e.g., pressure tube material, which is cold-worked by about 25%). Whereas annealed material contains no \(<c>\>-component dislocations, cold-worked material...**
contains copious quantities of \(<c>\)-component dislocations introduced by the cold-working step itself. The growth rate of cold-worked material is about equal to that, in the growth breakaway stage of annealed material. This leads to the general conclusion that whenever high growth rates are observed, \(<c>\)-component dislocations are present, whether introduced by irradiation, or cold work.

For materials that have had a very small amount of cold work (~1.5%), such as calandria tubes, an added complexity is observed. These show prolonged growth transients, where the strain rates violate the 1 - 3F law of equation (4). The growth strains can even become negative. A study of such results indicates that the transient behaviour is a consequence of internal stress introduced during fabrication.

Negative growth and violation of the 1 - 3F law has also been observed in a prototype pressure tube (called the Task Group 3: Route 1 material). This is a 2r-2.5% Nb alloy that has undergone a modified fabrication route to that used for standard CANDU pressure tubes. This growth behaviour is not a consequence of internal stresses, but rather is due to the different microstructure developed during manufacture and is discussed further below.

Key Theoretical Developments

Each individual grain in a polycrystalline assembly is a single crystal, whose deformation properties are anisotropic. When such a material is irradiated, each grain will attempt to deform (by irradiation growth) anisotropically. However, each embedded grain will be constrained to deform, self-consistently, with its neighbours. The resulting strain incompatibilities across the grain boundaries will produce local intergranular stresses. Such stresses (together with those introduced during tube fabrication) will give rise to creep (even in the absence of an external stress). This results in a coupling of creep and growth processes. These effects have to be taken into account in formulating the engineering design equation and in interpreting experimental data. Theoretical models have been developed to calculate the intergranular stresses and incorporate their overall effect on the macroscopic strain behaviour. These have been very successful in explaining strain transients in calandria tube material, departures from the 1 - 3F law and the general anisotropic behaviour of the pressure tube.

A breakthrough in the understanding of the evolution of the irradiation microstructure came through the realization that anisotropic behaviour in zirconium extended to its diffusion properties. The motivation in this research topic was to explain the striking anomaly of the coexistence of vacancy and interstitial loops. As explained above, the conventional theory of the dislocation sink strength predicted that only interstitial loops should be stable.

Early attempts to explain this discrepancy resulted largely in failure. However, a solution was found in the expectation that the diffusion of point defects (particularly interstitials) in zirconium was anisotropic, i.e., the rate of diffusion in one crystallographic direction is more rapid than in some other direction. The dependence of the bias on the orientation of the dislocations will now arise simply because of the fact that the point defects have different inherent mobilities in the \(<a>\)- and \(<c>\)-directions. In fact, the balance of evidence would suggest that interstitials diffuse faster within the basal planes (i.e., in the \(<a>\)-directions), whereas vacancies diffuse faster in the \(<c>\)-direction. Based on this behaviour, it is quite simple to envisage how the bias of sinks will be dependent on their orientation with respect to the crystallographic axes. Thus, an interstitial loop on a prism plane will receive more interstitials than vacancies and will grow. Conversely, a vacancy loop on the basal plane will receive an excess of vacancies, and will also grow, i.e., interstitial and vacancy loops can coexist, as the electron microscopy shows is the case.

This is a solution to the long-standing contradiction of the conventional theory and electron microscopic observations, described above. Furthermore, quantitative calculations show that the growth deformation rate is significantly increased by the presence of \(<c>\)-component dislocations. This offers an explanation for the higher growth rates in the cold-worked material (which contains \(<c>\)-component dislocations) compared to annealed material (no \(<c>\)-component dislocations) and accounts for the growth breakaway in the latter, which coincides with the spontaneous formation of \(<c>\)-component dislocations. In addition, the theory shows that a grain boundary can no longer be considered a neutral sink, but is biased according to its orientation with respect to the \(<a>\)- and \(<c>\)-directions.

The new bias calculations have been very successful in explaining many unusual features of the microstructure and deformation characteristics (both growth and creep). For example, the theory explains the negative growth observed in the Task Group 3, Route 1 material, mentioned above.

Engineering Design Equations

The engineering design equation (EDE) is a mathematical formula that allows the change in diameter or length of pressure tubes and calandria tubes to be calculated. Such information can be used by the reactor designer or operator to predict the dimensional evolution of the fuel channel. For this capability to extend over the lifetime of the components (30-40 years), the EDEs must have a predictive capability beyond the range of current empirical data. In addition, they facilitate calculations of fuel channel sag, the displacement of the reactor end shields, and hydride blister formation following pressure tube/calandria tube contact.

In the 1960s and early 70s, attention was largely focussed on the diametral (transverse) creep strain (\(\varepsilon_t\)) of the pressure tube. The EDE developed for this case was

\[
\varepsilon_t = K \sigma_t \phi \exp(-Q/RT)
\]

where \(\sigma_t\) represent the transverse (hoop) stress resulting from the coolant pressure; \(\phi\) the neutron flux; \(Q\) the activation energy; \(T\) the temperature; and the constant, \(K\), was estimated by fitting the equation to available experimental data for cold-worked Zircaloy-2, cold-worked and heat-treated 2r-2.5% Nb. The same equation (with an appropriate value of \(K\)) was assumed to apply to the calandria tube (annealed Zircalloy-2) to estimate creep sag. In this case, \(\sigma_t\) is replaced with \(\sigma_b\), the bending stress.
The above equation was derived in a purely empirical manner, the constants in the equations being determined from best fits to the limited experimental data. However, during the 1970s, a much fuller understanding of the underlying mechanisms was emerging and this was having increasing impact on the formulation of the EDEs. The outcome was a major restructuring of the equations. This accounted for the elongation of pressure tubes, as well as diametral strain, recognizing the explicit contribution of growth and the inherent anisotropy. The resulting EDEs for transverse, longitudinal and radial strain rates were as follows:

\[ \dot{\varepsilon}_t = (A \sigma_t + D \dot{\varepsilon}_t) \exp(-Q/RT) + G \dot{\varepsilon}_t \] (3a)

\[ \dot{\varepsilon}_l = (B \sigma_l + F \dot{\varepsilon}_l) \exp(-Q/RT) + G \dot{\varepsilon}_l \] (3b)

\[ \dot{\varepsilon}_r = (C \sigma_r + G \dot{\varepsilon}_r) \exp(-Q/RT) + G \dot{\varepsilon}_r \] (3c)

where \( A, B, C, D, E \) and \( F \) are factors related to creep anisotropy and to the hoop stress, \( \sigma_t \), and compressive stress, \( \sigma_c \), due to end loading. \( G \), \( G \) and \( G \) are directional growth factors. Numerical values for all parameters were derived from data fitting for cold-worked Zircaloy-2, cold-worked and heat-treated Zr-2.5% Nb.

In 1979, the EDE was again modified, based on the general formulation contained in Equation (3). However, important changes were incorporated. The creep term was split into three contributions, the first representing thermal creep (i.e., that which would occur even in the absence of irradiation), the other two consisting of radiation creep, one component being temperature-dependent and the other, temperature independent. In addition, a single equation was used to represent any stress state and any directional strain. This was achieved by a more sophisticated treatment of the anisotropy factors. This equation is

\[ \dot{\varepsilon}_d = K_0 \sigma_d^2 c_d \exp(-Q_0/RT) + (\text{thermal creep term}) \]

\[ K_0 \sigma_d^2 c_d \exp(-Q_0/RT) + K_1 \] (radiation creep term)

\[ K_0 \sigma_d \exp(-Q_0/RT)/(1 + K_5 \exp(-Q_0/RT)) \] (growth term)

where \( \dot{\varepsilon}_d \) is the steady-state rate in direction \( d \), \( \sigma_d \) and \( \sigma_t \) are effective stresses, which together with the creep anisotropy factors, \( c_d \) and \( c_t \), are calculated from a knowledge of the distribution of the transverse, longitudinal and radial stress components, together with anisotropy factors based on a plastic yield criterion. These factors are calculated from quantitative measurements of the crystallographic texture (based on X-ray data). \( G \) is the radiation growth anisotropy factor which is calculated from the crystallographic texture and grain shape, i.e., it is of the form of Equation (1).

The values of the parameters in the equation were based mainly on diameter gauging and length change measurements of power reactor pressure tubes, supplemented by tube gauging in NRU and WR-1, together with small specimen data.

A similar analysis was also used in 1979 to establish the EDEs for calandria tubes. In this case, separate equations were given for the axial elongation, diameter changes and sag. The parameters were estimated from small specimen creep and growth measurements conducted on calandria tube material irradiated in NRU.

This was the status of the EDEs up to 1979. The major advances in the formulation of Equation (4) were the recognition of the separate constitutive contributions to the overall strain and a methodology for handling their anisotropic characteristics. These improvements came about via the progress in understanding of the underlying deformation processes. However, most of the numerical coefficients, and other parameters, were evaluated from fitting the equations to available experimental data. Consequently, the validity of the EDEs was limited to a reactor operating time period of about 15 years. It was recognized that if the equations were to be predictive over total reactor lifetimes and applicable to future tubes with modified metallurgy, operating under different reactor conditions, they must be placed on a more fundamental foundation. This has been the main motivation for mechanistic studies during the 1980s, as described in this paper. These have resulted in recent updates of the EDEs for pressure tubes and calandria tubes. The general format of these is similar to that found in Equation (4), but with the following refinements:

1. The thermal creep component was considerably modified (stress and temperature dependence) as a result of extensive data from the WR-1 reactor.

2. The linear dependence of in-reactor deformation on stress and flux was substantiated by both experimental data and theoretical models of the mechanisms.

3. A large database for pressure tube elongation was generated for the first time (390 pressure tubes measured over 12 years). Similarly, extensive growth data for calandria tubes were available from small specimen testing.

4. Improved mechanistic models of growth, based on vacancy loop growth via anisotropic diffusion, resulted in a major revision of the growth anisotropy factors.

5. Intergranular interactions were taken into account for the first time. It was shown that although this causes a coupling between creep and growth mechanisms, the theory supports the separable and additive features of the strain rate terms in the EDEs.

**GENERAL DISCUSSION**

The significant engineering and economic consequences of fuel channel deformation provide ample incentive for accurate predictive capability and improved materials. A key tool towards this end is the improvement of the engineering design equation. As is evident from Equation (4), there is a large number of independent variables and constants in the present design equations that must be evaluated from the database. Fortunately, our phenomenological and mechanistic understanding has supported the choice of stress and flux dependencies (both linear) and the calculation of the crucial anisotropy factors. The greatest remaining uncertainty resides in the values of the activation energies and the strain rate coefficients (i.e., the \( Q_s \) and \( Q_g \) in Equation (4)).

Representing the temperature dependence of deformation through an \( \exp(-Q/RT) \) term is supported by the thermally activated mechanisms incorporated into the rate theory. The \( Q_s \) are then linked with the
diffusional mobilities of the vacancies and interstitials. However, there is a dearth of information on point defect properties in zirconium, making an unequivocal assignment of Q values difficult, even if the controlling mechanisms were well defined. This means that the values of Q must be largely determined empirically, a process that is not independent of the evaluation of the Ks, in a data fitting procedure. Further complication arises if the evolution of the microstructure causes a change in the mechanism (and therefore a transition to a different Q). It is this kind of uncertainty that underlines the requirement for mechanistic understanding for reliable extrapolation.

The numerical factors, Ks, are also obtained mainly by empirical methods, reflecting the lack of mechanistic understanding. This is where the rate theory approach can make the greatest contribution. Thus, in principle, it should be possible to make explicit calculations of the irradiation-induced production rate of point defects, the rate at which they are absorbed at dislocations and grain boundaries and therefore the final deformation rate. In addition, many of the microstructural factors are contained in the K constants. These would include grain size, and phase compositions, dislocation morphology, alloy effects and the modification of all of these entities by metallurgical fabrication processes. In general, these factors are unknown and therefore do not appear explicitly in the design equations. The fact that we have separate EDEs for the pressure tube and the calandria tube is a consequence of this deficiency.

In order to have a comprehensive design equation (more properly referred to as the fuel channel deformation model - FCDM - in what follows), the deformation models must be mechanistically based. Their development will be evolutionary, being updated as our knowledge base advances. As they will derive from an understanding of radiation damage in general, they will also find application in the fields of in-reactor fracture, corrosion and hydriding. With Figure 8 as a guide, we can summarize the present status of this goal and outline future requirements. This figure breaks down the FCDM into its constituent parts, with an indication of the dimensional scale of its architecture:

**Box A: Point Defect Properties**

Because of the intrinsic experimental difficulties in this area, it is unlikely that all point defect parameters will be measured in the near term. Best effort estimates will be required based on computer simulation models and the extant literature.

An important item is to determine the precise number of vacancies and interstitials produced by each neutron-induced collision cascade, to establish the flux effect.

**Box B: Creep and Growth Mechanisms**

The main body of this paper has described the impressive progress that has been made in this area. Before such information can be inserted into the FCDM, experimental effort must be directed toward the selection of those mechanisms which dominate in zirconium. Such experiments, on well-characterized material, will employ in-reactor and accelerator simulation facilities. Electron microscopy will also play a vital role.

**Box C: Single Crystal Model**

At this scale of dimension, we are now asking the question, "How do the proposed mechanisms for creep and growth express themselves in each individual grain?" We currently have a better answer for growth than for creep. From an experimental point of view, we might take the approach of making measurements on single crystals, however, these are not available for the commercial alloys. Another, more practical, approach is to extract the single crystal information from tests on polycrystalline specimens. An analytical method has been devised for this.

Electron microscopy is an important experimental tool at this level of investigation. This can be used to analyze the microstructural information; in particular, the types of strain producing dislocations on various crystallographic planes. The influence of microchemistry can also be examined.
**Box D: Polycrystalline Model**

This has been the area of most significant progress to date. The grain-grain interactions, the relaxation of intergranular stresses and its relationship to the crystallographic texture, has been a notable achievement. This has yielded improved anisotropy factors which have had a dramatic impact on the design equations.

**Box E: Component Model**

Developments in this area are also at an advanced state. We are now at the engineering level, where the global distortions of components and component assemblies can be modelled. Existing computer programs provide this capability. These link the polycrystalline models to the actual physical measurements taken at the reactor site. The only anticipated refinement in this area will be a more comprehensive elastic-plastic finite element analysis.

The overall goal represented by Figure 8 is realistically achievable. Its time scale will be a function of the available manpower and other resources. Assuming a status quo scenario, a reasonable estimate is 10 to 15 years. This corresponds to the time period where it is judged there will be a resurgence of power reactor construction in Canada and the rest of the world. At such a time, the deformation model will be required to serve tens of thousands of fuel channels.

**REFERENCES**


The issue of calandria tube integrity following a sudden rupture of the pressure tube in Pickering NGS A reactors is addressed. Based on operating experience, only fish-mouth ruptures of the pressure tube are considered to be credible. The calandria tube response to the pressure tube break is delineated into three distinct stages, i.e. the initial transient response during the annulus filling stage, transient overpressurization and the final steady-state loading after bellows failure. The annulus response in the second stage is dominated by a waterhammer type overpressure transient with attenuation of this transient due to plastic straining of the calandria tube. The annulus pressure transients for various breaks and the sensitivity of the results to various parameters are presented. The strength margins of the calandria tube are evaluated to be relatively large.

INTRODUCTION

The main function of the calandria tube (CT) is to minimize the heat loss from the primary coolant to the cool moderator by providing an insulating gas annulus. In the Pickering NGS this is achieved by circulating dry CO2 gas in the gas annulus system which also acts as a means of detecting any leaks in the pressure tubes. The calandria tubes are subjected to the internal pressure of the annulus gas and to the external pressure due to the moderator. However, during accident scenarios involving pressure tube (PT) rupture, the surrounding calandria tube is subjected to severe pressure and impact loading depending on the nature of PT failure. The issue of calandria tube integrity under a pressure tube failure scenario in Pickering NGS A reactors is addressed here with the objective of quantifying the CT strength margins for failure. The method of obtaining the pressure transient in the PT/CT annulus and the influence of various parameters pertinent to the primary heat transport (PHT) system on the pressure transients is studied. The influence of calandria tube plastic deformation on these transients is examined next. Based on these results the calandria tube strength margin to failure is evaluated both under transient and steady-state loading.

The pressure tubes which form the primary pressure boundary in the reactor core are designed as ASME Class I pressure components and are manufactured from Zr-2.5 percent Nb material which has a high fracture toughness. Hence the pressure tubes are expected to show leak-before-break and any failures are likely to be small leaks which propagate in the axial direction. A detailed review of the operating experience with the pressure tubes in Ontario Hydro's reactors given in Reference 1 has indicated that all pressure tube failures in Ontario Hydro's operating reactors can be attributed to a single cause, i.e., delayed hydride cracking (DHC). This review has also confirmed the design intent that the only likely failure of a pressure tube during normal operation is a fish mouth rupture which might propagate some length in the axial direction. Based on this experience, for present analysis purposes only sudden axial ruptures of the PT are considered to be credible.

PT/CT INTERACTION FOLLOWING PT FAILURE

The experimental observations (References 2 to 4) indicated that the interaction of pressure and calandria tubes following a pressure tube rupture can be divided into three distinct stages, i.e.:

(a) Initial transient stage which usually lasts up to about 5 ms during which the annulus starts to fill with water.

(b) Transient overpressurization stage following annulus filling which can last anywhere between 200 and 500 ms depending on the coolant condition and the size of the PT break.

(c) A quasi steady annulus pressurization stage before reactor shutdown.

In Stage (a), discharge of the coolant causes a large asymmetric pressure distribution on the calandria tube which results in an impulse loading in the transverse direction. The pressure tube is subjected to a reaction thrust force which results in the movement of pressure and calandria tubes in opposite directions culminating in bodily impact of the PT on the CT. Following the impact the CT is subjected to inextensional deformation which causes ovaling of the CT. During this stage of annulus fill up other minor impacts on the CT due to the transient loads is minimal. Similar conclusions have been reported by studying the response of guard pipes to a sudden failure of inner pipes (Reference 6).

In Stage (b), the CT is subjected to uniform hoop stress due to internal pressure. However, the magnitude of the internal pressure can exceed the mean header pressure due to sudden deceleration of the coolant at the ends of the CT (i.e., at the bearing sleeves in the end fittings which act as flow constrictions). If the stress level in the CT
CT. The response of the CT to the pressure straining of the CT. A detailed description of the code FSTI and the results of sensitivity studies of input parameters to the code are given in Reference 7.

In Stage (c), the CT will experience a steady-state pressure which will be slightly below the mean header pressure due to the steady discharge of the coolant through the gas annulus system bellows.

**THERMAL-HYDRAULIC RESPONSE**

In this section a detailed sensitivity analysis of the thermal hydraulic response of the fuel channel and the gas annulus in stage (b) is presented. The pressure transients are evaluated by using the MINI-SOPHT computer code which assumes that the entire pressure boundary between the headers is rigid. A brief description of the computer code is given in Reference 3. The pressure transients are evaluated by using the MINI-SOPHT computer code which assumes that the entire pressure boundary between the headers is rigid. A brief description of the computer code is given in Reference 3. The magnitude of the pressure rise in the annulus is dependent on the fluid discharge rate into the annulus which in turn is dictated by the hydraulic resistance characteristics of the heat transport system.

The main parameters considered in the sensitivity study are different break lengths, break locations (inlet end, outlet end, break over the entire length of the core), inlet feeder temperatures, inlet and outlet feeder resistances, break discharge area, bearing clearances, header pressure, channel power and channel flow. Results obtained using the MINI-SOPHT code for a number of channels in Pickering A are reported. These results and the pressure pulse data are later used as input to the FSTI code to assess calandria tube integrity.

The thermal hydraulic behaviour of the channel/feeder system was simulated for a number of typical channels covering a range of power and flows (i.e., channel G16, H19, A08 and L11) in the sensitivity study. Figures 1 to 3 show the schematic nodalization used to simulate inlet end break, outlet end break and full length break respectively. The simulation matrix for the sensitivity study for all channels and a summary of the predicted results are shown in Table 1. In all of the above simulations the break discharge area was assumed to be 140 cm². In order to study the sensitivity of the peak pressure to the discharge area, a 1.20 m break was used in all the remaining sensitivity analysis. The results in Table 2 indicate that doubling the break area increases the peak pressure by about 5 percent while increasing the break area by a factor of 3 only increases the peak pressure by 16 percent. By comparison, a factor of 3 increase in the break length results in a 50 percent increase in the peak pressure (Case 1 versus Case 6). As the peak pressure is not strongly dependent on break area, only a realistic estimate of the break area (i.e., 140 cm²) was used in all the remaining sensitivity analysis. This break area is close to the maximum break discharge area observed in the full-scale burst tests reported in References 2 to 4.

**Effect of Channel Power and Flow**

The effect of channel power has been assessed in channel G16 with a 1.98 m long outlet end break for two channel powers, i.e., 4.74 MW and 5.25 MW (Cases 1 and 3). The maximum predicted annulus pressure at a channel power of 5.25 MW is about 12.1 MPa compared with the peak of 15.2 MPa at 4.74 MW. For a given set of channel conditions these results indicated that the pressure peak increases as channel power decreases due to reduction in the outlet temperature which lowers the coolant subcooling.

To study the effect of nominal channel flow on the maximum pressure, a 1.20 m break at the outlet...
TABLE 1: SIMULATION MATRIX OF CASES CONSIDERED AND RESULTS OBTAINED

<table>
<thead>
<tr>
<th>Tent Case</th>
<th>Location</th>
<th>Size (m)</th>
<th>Inlet Temperature (°C)</th>
<th>Power (MW)</th>
<th>Flow (kg/s)</th>
<th>Predicted Peak Pressure (MPa)</th>
<th>Pressure Pulse Width (ms)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>outlet</td>
<td>1.98</td>
<td>249</td>
<td>4.74</td>
<td>21.7</td>
<td>15.2</td>
<td>30.5</td>
</tr>
<tr>
<td>7</td>
<td>outlet</td>
<td>1.20</td>
<td>249</td>
<td>4.74</td>
<td>21.7</td>
<td>14.3</td>
<td>31.37</td>
</tr>
<tr>
<td>3</td>
<td>outlet</td>
<td>1.98</td>
<td>249</td>
<td>5.25</td>
<td>21.7</td>
<td>12.1</td>
<td>32.5</td>
</tr>
<tr>
<td>4</td>
<td>outlet</td>
<td>1.20</td>
<td>249</td>
<td>4.74</td>
<td>23.8</td>
<td>14.5</td>
<td>32.7</td>
</tr>
<tr>
<td>5</td>
<td>inlet</td>
<td>1.20</td>
<td>249</td>
<td>4.74</td>
<td>23.8</td>
<td>15.2</td>
<td>30.25</td>
</tr>
<tr>
<td>6</td>
<td>full length</td>
<td>5.94</td>
<td>249</td>
<td>4.74</td>
<td>23.8</td>
<td>23.5</td>
<td>35.3</td>
</tr>
</tbody>
</table>

Channel H19

| 7         | outlet   | 1.2    | 249                    | 4.379      | 21.84      | 14.5                        | 31.08                    |
| 8         | outlet   | 1.2    | 200                    | 4.379      | 21.84      | 17.2                        | 32.0                     |
| 9         | outlet   | 1.2    | 150                    | 4.379      | 21.84      | 18.75                       | 42.93                    |
| 10        | outlet   | 1.2    | 100                    | 4.379      | 21.84      | 20.25                       | 80.30                    |
| 11        | inlet    | 1.2    | 249                    | 4.379      | 21.84      | 16.0                        | 31.01                    |
| 12        | inlet    | 1.2    | 200                    | 4.379      | 21.84      | 19.1                        | 30.07                    |
| 13        | inlet    | 1.2    | 150                    | 4.379      | 21.84      | 20.5                        | 44.0                     |
| 14        | inlet    | 1.2    | 100                    | 4.379      | 21.84      | 22.2                        | 61.6                     |
| 15        | outlet   | 1.2    | 249                    | 4.379      | 19.21      | 13.5                        | 29.75                    |
| 16        | outlet   | 1.2    | 249                    | 4.379      | 22.69      | 15.0                        | 30.2                     |
| 17        | outlet   | 1.2    | 249                    | 4.379      | 22.84      | 15.15                       | 30.3                     |
| 18        | full length | 5.94 | 249                    | 4.379      | 21.84      | 24.0                        | 35.5                     |

Channel A08

| 19        | outlet   | 1.98   | 249                    | 2.85       | 16.16      | 13.56                       | 32.5                     |
| 20        | inlet    | 1.98   | 249                    | 2.85       | 16.16      | 15.90                       | 31.0                     |
| 21        | full length | 5.94 | 249                    | 2.85       | 16.16      | 23.45                       | 32.05                    |

Channel L11

| 22        | outlet   | 1.98   | 249                    | 4.8        | 23.37      | 13.53                       | 32.09                    |
| 23        | inlet    | 1.98   | 249                    | 4.8        | 23.37      | 15.65                       | 29.03                    |
| 24        | full length | 5.94 | 249                    | 4.8        | 23.37      | 23.14                       | 32.15                    |

TABLE 2: EFFECT OF DISCHARGE BREAK AREA ON PEAK PRESSURE IN CHANNEL H19

<table>
<thead>
<tr>
<th>Break Area (cm²)</th>
<th>Peak Pressure (MPa)</th>
<th>Pulse Width (ms)</th>
</tr>
</thead>
<tbody>
<tr>
<td>140</td>
<td>15.5</td>
<td>30.0</td>
</tr>
<tr>
<td>280</td>
<td>16.3</td>
<td>31.0</td>
</tr>
<tr>
<td>420</td>
<td>18.0</td>
<td>30.3</td>
</tr>
</tbody>
</table>

end in channel G16 (Cases 2 and 4) and channel H19 are considered (Cases 7, 15, 16 and 17). The channel flow was varied by changing the frictional flow losses in both feeders. The results obtained show that the maximum pressure increases as channel flow increases. This can be explained by considering the two effects that accompany an increase in channel flow, i.e., an increase in channel flow causes not only an increase in the flow into the annulus (as in case of increased break size) but it also reduces the outlet temperature for a given channel power. This combined effect of increased flow and reduced subcooling results in a corresponding increase in peak pressure.

Effect of Temperature

The effect of the coolant inlet temperature on the maximum pressure has been studied for channel H19 with a 1.2 m long break for two break locations (i.e. inlet and outlet). The coolant temperature has been varied over a wide range from 100°C to 250°C (Cases 7 to 14). These results (in Figure 6) show that maximum pressure increases monotonically with decreasing inlet temperature. Another observation is that the pressure pulse width increases with decreasing temperature. The results again confirm the previous observation that an inlet break gives a higher pressure peak than an outlet break for a given set of channel conditions.

Effect of Bearing Clearance

The effect of bearing clearance has been studied, by varying the valve discharge area in the valve simulating the clearance, for a 1.98 m outlet break in channel H19. The results obtained for three different bearing clearances (Table 3) indicate that, with increased bearing clearance, the peak pressure predicted decreases due to the increased...
discharge of coolant through the bearings. A
doubling of the nominal bearing clearance area (from 
2.75 to 5.5 cm$^2$) reduces the peak pressure by 
about 5 percent while a three fold increase in the
bearing clearance area decreases the peak pressure 
by 13 percent. These results demonstrate that the
peak pressure is very insensitive to any increase in
the bearing clearances within a desirable range.

<table>
<thead>
<tr>
<th>Clearance Area (cm$^2$)</th>
<th>Peak Pressure (MPa)</th>
<th>Pulse Width (ms)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.75</td>
<td>15.5</td>
<td>30.0</td>
</tr>
<tr>
<td>5.50</td>
<td>14.8</td>
<td>30.3</td>
</tr>
<tr>
<td>8.25</td>
<td>13.5</td>
<td>34.4</td>
</tr>
</tbody>
</table>

**Effect of Header Pressure**

In order to study the effect of header pressure
on the predicted peak annulus pressure, an outlet
end break (1.98 m long) in channel H19 is again
considered. Three inlet header pressures ranging
from 9.73 to 11.4 MPa, are considered. The results
of the predicted peak pressure in Table 4 indicate
that header pressure has negligible effect on the
annulus over-pressure transient. Thus, as the
header pressure is increased, the incremental
overpressure remains approximately constant.

<table>
<thead>
<tr>
<th>Inlet Header Pressure (MPa)</th>
<th>Peak Pressure (MPa)</th>
<th>Pulse Width (ms)</th>
</tr>
</thead>
<tbody>
<tr>
<td>9.73 (nominal)</td>
<td>15.5</td>
<td>30.0</td>
</tr>
<tr>
<td>10.3</td>
<td>15.87</td>
<td>34.4</td>
</tr>
<tr>
<td>11.4</td>
<td>16.8</td>
<td>35.0</td>
</tr>
</tbody>
</table>

**Results and Discussion**

Based on this sensitivity study, the influence of
various parameters on the peak pressure can be
generalized in terms of coolant subcooling and flow
rate in the channel as follows:

(a) An inlet break causes a larger waterhammer
pressure than the same size break at the outlet
of the fuel channel due to higher coolant
subcooling at the inlet end.

(b) Increasing the break length (for a given break
area) causes an increase in the peak
waterhammer pressure. The maximum waterhammer
pressure is obtained for a full length (5.94 m)
break in the PT.

(c) A reduction in the inlet temperature causes an
increase in the waterhammer pressure again due
to higher coolant subcooling.

(d) An increase in the channel mass flow rate
(e.g., by decreasing the feeder resistance)
increases the cross flow into the annulus and
reduces the coolant outlet temperature. Hence
it causes an increase in the waterhammer
pressure in the annulus.

(e) Alternatively a reduction in the channel power
at same nominal flow and inlet temperature will
have the same effect as increasing mass flow rate
and thus it will cause a larger water
hammer pressure transient.

(f) As the channels in the Pickering reactors are
flow-power matched, the peak pressure predicted
is almost identical for all channels for a
given break length and location.

(g) The peak waterhammer pressure is insensitive to
any increase in the end-fitting bearing
clearance areas within a practically desirable
range.

(h) The header pressures do not affect the
incremental overpressures significantly. The
peak waterhammer pressure is thus obtained by
adding the incremental overpressure to the mean
header pressure.
For this break size, the margin to failure exists for both lower bound and nominal irradiated calandria tube strengths for a unirradiated material. The results of these calculations, given in Table 6, show that a large margin for a full length break is obtained for the steady state annulus pressure is not dependent on break length and hence the margin to failure is identical for both breaks shown in Table 6.

For a full length break, the failure margins for a lower bound strength tube is seen to be negative based on 0.1 percent failure criterion, which implies that the tube strains by 0.1 percent at about 20 MPa pressure. The corresponding failure margin for a nominal tube is positive but small. Thus for the low ductility failure criterion (0.1 percent plastic strain) the limiting failure margin for a full length break is obtained for the transient rather than the steady-state loading phase. For the higher ductility failure criterion (1.0 percent plastic strain) large margins to failure are again obtained.

It is reiterated that the lower bound strength is unrealistic since tube strengths approach approximately the same nominal strength after several years irradiation. Also, a low ductility failure can be considered unlikely or of low probability. The expected failure margins are denoted by asterisks in Table 6 for the transient and steady state. Thus large margins to calandria tube failure are expected in Pickering NGS A.

REFERENCES


FIGURE 1
Schematic Nodalization of Channel G16 with an Inlet Break

FIGURE 2
Schematic Nodalization of Channel G16 with an Outlet Break
FIGURE 3
Schematic Nodalization of Channel G16 with a Full Length Break

FIGURE 4
Pressure Transient in the Event of a Sudden Pressure Tube Break in Channel G16 with an Outlet Break of 1.98 m

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FIGURE 5
Pressure Transient in the Event of a Sudden Pressure Tube Break at the Outlet of 1.20 m in Channel G16

FIGURE 6
Effect of Inlet Temperature on the Pressure Peak in the Event of a Sudden Pressure Tube Break in Channel H19
DESIGN FEATURES OF THE FUEL CHANNEL
TO BE USED IN THE RE-TUBING OF PICKERING 3 AND 4

BY

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Mississauga, Ontario L5K 1B2

1.0 ABSTRACT

Increasing deuterium levels in pressure tubes, axial bearing limits and system planning constraints have necessitated the re-tubing of Pickering Units 3 and 4. The fuel channel is the key component to be replaced. Design changes which were introduced were done to primarily improve installability and reduce installation time. In addition however, some design changes were also made to incorporate improved features without jeopardizing the schedule.

2.0 INTRODUCTION

Pickering Units 3 and 4 have been in commercial operation since June 1972. After approximately 16 years of operation and after the successful re-tubing of Pickering Units 1 and 2, Ontario Hydro decided to re-tube these reactors. The primary reason for this decision, as for Units 1 and 2, was a concern that a pressure tube could contact the calandria tube resulting in hydride blistering and failure. There was a high probability of pressure tube to calandria tube contact occurring since these units have only 2 spacers per channel and, therefore, an increasing number of pressure tubes would prematurely contact the calandria tube. It was also decided to re-tube these units because of axial bearing limits (due to creep elongation) and system planning constraints since Bruce 1 and 2 may also require re-tubing.

The retubing operation will start on June 1989 for Pickering 3 and two years later for Pickering 4. Unit 3 and Unit 4 are scheduled to return to service approximately 23 and 19 months respectively after being shut down. The fuel channel, as shown in Figure 1, is the main component to be replaced.

Fuel channel designers were somewhat limited in the extent of improvements which could be made to the existing P3/4 design due to schedule restrictions. Emphasis was placed on introducing design changes and modifying installation procedures to improve installability and reduce installation time. Following is a description of the major changes which were made to the existing design or to the P1/2 LSFCR* design in order to achieve the above objectives.

* Large Scale Fuel Channel Replacement
3.0 DESIGN DESCRIPTION

3.1 Annulus Spacers

The annulus spacer arrangement is shown in Figure 2.

Four spacers will be used because this prevents contact between the pressure tube and calandria tube and therefore the occurrence of blisters and delayed hydride cracking. The spacers will also be the snug fitting type to ensure that they remain in their design location during plant operation. Furthermore, the spring material will be inconel instead of zirconium to prevent hydriding and blistering of the spring. This material was also used since the tight spacer design requires hooks and the fabrication of hooks leads to the formation of radial hydrides in zirconium. Differences between the two designs are shown in Figure 3.

3.2 End Fittings

In the re-tubing of Pl/2 only one end fitting was removed from each fuel channel. The east end fitting was left in place and the rolled joint was shock heated in order to free the pressure tube from the end fitting. The end fitting bores were subsequently honed so that the end fittings could be re-used and thick ended pressure tubes were then installed. Those pressure tubes were thicker at the east end in order to match the existing (larger) end fitting bore.

The shock heating operation proved to be more time-consuming and problematic than anticipated. In order to reduce the P3/4 outage it was therefore decided to replace both end fittings since this eliminated both shock heating and honing operations and the need for the thick ended pressure tubes.

3.3 Pressure Tube to End Fitting Rolled Joint Configuration

The location of the pressure tube to end fitting rolled joint is shown in Figure 5. Although various rolled joint designs have been used in the past all of the most recent reactors such as Bruce 'B', Darlington and the CANDU 6 have one common feature – the zero clearance rolled joint. To obtain this type of joint the end fitting is first heated and the pressure tube is then inserted prior to rolling. This results in a "zero clearance" fit between the outside diameter of the pressure tube and the end fitting bore. (The fit actually varies from .002" clearance to .007" interference.) Use of a zero clearance joint minimizes the residual stresses which are created in the pressure tube during rolling. Joints with higher clearances have correspondingly higher residual stresses as shown in Figure 6.

The objective of keeping residual stress low is to minimize the probability of delayed hydride cracking (DHC). In order for DHC to occur, the material must be under high stress and contain a high concentration of hydrogen. Therefore, in order to combat DHC either one or both of these conditions must be controlled.
The Pl/2 LSFCR joints were of the low clearance type since all rolled joints were made on the reactor face and tooling was not available to produce a zero clearance joint on the reactor. It is also more difficult and time consuming to produce a zero clearance rolled joint. For P3/4 the tooling is still not available but since both end fittings will be replaced, half of the joints can be made in a sub-assembly area away from the reactor face and, as such, can be of the lower stressed zero clearance type. Low clearance rolled joints will be produced at the reactor face because the tooling is not available and because of schedule limitations.

It should also be noted that in order to further minimize the probability of DHC, consideration was given to installing yttrium metal hydrogen getters in the P3/4 pressure tubes. These yttrium sinks would keep hydrogen levels in the rolled joint area of the pressure tube sufficiently low to prevent DHC from occurring. Since development had not been completed, it was decided not to include this feature in P3/4. However, this concept is being studied further and may be implemented in future reactor designs.

3.4 Positioning Assembly

At the present time the positioning assembly is located inboard on the feeder connection, i.e. between the feeder pipe and the end shield. Accessibility to the positioning assembly is therefore somewhat restricted because of the feeders. In more recent designs such as Bruce 'B' and Darlington the positioning assemblies have been re-located just outboard of the feeder connection. Figure 7 demonstrates this difference.

In order to improve access and therefore reduce installation time, positioning assemblies will only be installed at one end of the reactor face instead of both ends as in P1/2 LSFCR. The fuel channel is therefore allowed to creep at the free (east) end and after approximately 12½ years it is repositioned towards the west end. This will allow continued operation for a further 12½ years.

3.5 Helium Leak Testing of Rolled Joints

The pressure tube to end fitting rolled joint has been helium leak tested for all commercial CANDU reactors built to date (1). This was considered the final test of rolled joint integrity before reactor commissioning and start-up. It was believed that the helium leak test confirmed both the leak tightness of the rolled joint and that rolling procedures were being strictly followed. However, for a retube project, a system helium leak test directly adds at least 3 weeks to the schedule. Consideration was therefore given as to whether this additional schedule/cost could be justified in light of past experience.

The results of the history of rolled joint testing can be summarized as follows:

a) out of approximately 20,000 rolled joints made to date, none has leaked water after installation.

b) out of 11,920 zero clearance rolled joints made, only 6 were rejected for failing the helium leak test. (4 were subsequently retested after removal from the reactor and were acceptable).
c) the reliability of the helium leak test has often been dubious. Some joints that had leak rates in excess of requirements were subsequently retested and deemed to be acceptable. Some joints with deliberate flaws successfully passed the helium leak test.

d) there is no correlation between helium and water leak tightness for low clearance joints which fall within the normal range of helium leak tightness. On the other hand, if a joint passes the helium leak acceptance criteria, tests have shown that any water leakage will be insignificant.

If the helium leak test were the only way to determine the adequacy of the joint, then the continuation of this test would be mandatory. However, a complete dimensional inspection of the joint before and after rolling ensures that the design parameters have been met. Given the adequacy of the dimensional inspection and the past record of helium leak testing, continuation of a helium leak test where there is a resultant schedule penalty could not be justified.

It was decided that while the helium leak test could not be justified if it results in a schedule penalty, it should be continued (in order to obtain additional trending data) where schedule is not an issue (e.g. on the P3/4 subassembly joints and on all new reactor joints).

3.6 High Frequency Ultrasonic Examination of Pressure Tubes

During the past number of years there has been a move towards increasing both the type and the level of inspection carried out at the pressure tube manufacturer. There are three reasons for this:

- it increases the reliability of pressure tubes.
- there are economic incentives since uncovering a flaw at the manufacturer's facility is much less costly than it would be after installation.
- in-service inspection becomes more meaningful and less difficult if a suitable level of inspection goes on at the manufacturer's plant.

Bearing the above in mind, the level of ultrasonic examination was increased for P3/4 LSFGR. Previous ultrasonic testing requirements had included a low frequency normal beam and shear wave inspection. A higher frequency normal beam technique was introduced during pressure tube manufacture after the Bruce N06 incident in order to detect tight lap defects, of the type found in N06. The high and low frequency methods are complementary and both will be used for P3/4 to provide improved detection capability and characterization of a specific type of flaw.
3.7 Modifications to Journal Rings

As in P1/2 LSFCR, the length of both the inboard and outboard journal rings has been increased to extend the creep life of the fuel channel since this increases the time before the end fitting comes off its bearing. This, in turn, increases the operating life of the fuel channel. However, unlike P1/2 LCFCR, minor changes were also made to the geometry of the journal rings to improve installability. This included adding chamfers to the leading edges of the bearings in addition to the existing radii. Figure 8 shows both the existing journal rings and those to be installed in P3/4 LSFCR with the arrows indicating the areas which have changed.

3.8 Shield Plug

The shield plug has been re-designed to improve its reliability and decrease its capital cost. This has been accomplished by reducing the number of parts which comprise a shield plug assembly. The existing 8-jaw design has been modified to a simpler 3-jaw configuration. A further improvement is that the new shield plug can be dismantled in situ if it becomes stuck in an end fitting during re-fuelling. These changes, however, were not incorporated in the P3 design because testing had not been completed in time. Once all tests are completed and schedule permitting, these modifications will be made to the P4 design unless the change cannot be justified on a commercial basis (once contract cancellation costs are considered).

4.0 Future Fuel Channel Designs for Re-Tubed Reactors

As mentioned above, there were a number of innovations such as yttrium sinks which were considered for P3/4 but could not be implemented because development and testing had not been completed in time. However, the use of yttrium sinks and other improvements are currently being evaluated and may be introduced in the Bruce 1/2 fuel channel design. This includes items such as the spring finger detent (instead of positioning assemblies), chromed end-fitting bores and shorter end fittings to increase feeder spacing.

5.0 Summary

The above has described the main new features of the P3/4 LSFCR fuel channel. Although there is no one item which is very novel about this design there are a number of improvements which have been made to both the base design (P1/2 LSFCR) and that which currently exists in P3/4. The P3/4 LSFCR design strikes a balance between the project requirements (improve installability, decrease installation time) and the continuing efforts to increase both the reliability and the operating life of the fuel channel.

REFERENCES


The Evolution of Pressure Tube Sampling Tools for Life Assessment

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Abstract

This paper describes the development of special-purpose tooling to obtain small samples of material from the inside diameter of pressure tubes in CANDU reactors without detrimentally affecting the integrity of the tube. This material is then analyzed for deuterium (D₂) content to assist in assessing the condition of the tube. The evolution of the sampling technology is outlined from the original concept (1986) to the present.

1. Introduction

Pressure tube integrity and life are major long-term concerns in CANDU reactors where recent tube failures have contributed significantly to loss of capacity. Extensive work has been done to demonstrate pressure tube integrity; however, the conventional method of assessing the useful life of pressure tubes in CANDU reactors requires the periodic removal of a tube. Samples are cut from the removed tube and analyzed for deuterium (D₂) content. The D₂ concentration is then used as a measure of the useful life of the remaining pressure tubes. This approach is very costly because of the long shutdown period required to remove and replace a pressure tube.

A method of measuring D₂ concentration without removing pressure tubes would be extremely beneficial. If such a method were available, it would also be possible to determine the condition of many tubes during a shutdown rather than just one.

In recognition of this need, special tooling was developed by Atomic Energy of Canada Limited at Chalk River Nuclear Laboratories (CRNL) to obtain a sample of material from the inside diameter of a pressure tube without removing the tube from the reactor. Only a small amount of material is removed for D₂ analysis (about 50 mg), and the contour of the sample area is carefully controlled so that tube integrity is preserved.

Following bench testing at CRNL, a prototype tool was used at NPD in 1987 May and again in July. One tube was subsequently removed from the reactor and coupons were punched from it, near the positions that had previously been sampled with the prototype tool. Deuterium analyses of the samples were in good agreement with analyses of the coupons. Metallographic sections through the sample grooves showed they were shallow, had very smooth surfaces and no sharp corners that could act as stress raisers. It was concluded that the sample grooves would in no way affect the strength or fracture resistance of the pressure tube.

2. First Generation: The NPD Tool

A prototype tool was designed at CRNL in early 1987 to remove samples from the Nuclear Power Demonstration (NPD) reactor in Rolphton, Ontario. NPD was chosen because scheduled removal of a pressure tube provided an excellent opportunity for in-reactor evaluation of the sampling concept.

The tool consisted of an abrasive flapper wheel for removing oxide, a cutter module to take the sample, a rotating module to interchange the flapper wheel and the cutter for sequential positioning against the inside of the pressure tube and pneumatically actuated rams to lock the tool head in place. A simple manual control system provided for proper sequencing of tasks as a sample was taken.

The abrasive wheel was initially intended to remove loose debris and oxide from the area to be sampled in order to avoid potential contamination of the sample and confusion of the analysis. In field operation, the flapper wheel was found ineffective in removing in-reactor oxide (as opposed to tests on non-irradiated oxide where it was successful). Subsequent comparison of analyses of small samples and throughwall pellets indicated excellent agreement. The D₂ levels in the Zircaloy tubes were sufficiently high that the oxide had no noticeable effect on the value.

This first generation tool served to prove the concept and to demonstrate that suitable samples could be obtained. However, CANDU plants, other than NPD, require larger-diameter tools to fit inside 104 mm diameter pressure tubes. It was also recognized that automatic sequencing of the tool motions would be required for routine use of the tooling by station staff.

Following the success of the NPD tool, work was started immediately to design, fabricate and develop tools for a planned outage at Pickering Unit 4 scheduled for 1987 November.

3. Evaluation of the Effect of the Oxide Layer

In parallel with the design of Pickering tools, samples were taken at CRNL using the NPD sampling tool from a pressure tube removed from service. Adaptors were made to permit the tool to be used in segments of the larger-size irradiated pressure tube stored at CRNL. Samples were taken at four locations, nominally 4.1, 4.8, 5.1 and 5.5 m from the inlet end of the pressure tube. Standard throughwall pellets, as previously used for deuterium analysis, were also punched near the scrape positions.
A comparison of analysis results for the samples and throughwall pellets indicated a severe bias in that the sample deuterium concentrations were significantly higher than concentrations for the pellets taken in the same location of the pressure tube segments. A segment of a second irradiated pressure tube was also sampled with similar results.

Review of micrographs from previous work investigating oxide on Pickering pressure tubes suggested that some concentration of hydrides may occur at the oxide-metal interface both on the OD and ID of the pressure tube. In order to investigate this possibility, three throughwall pellets were taken from the 4.1 m location in the sampled pressure tube and cut into thin sections. Results of deuterium analyses, given in Table 1 below, support the presence of an enriched layer just beneath the pressure tube surface.

<table>
<thead>
<tr>
<th>Pellet Number</th>
<th>Location (Relative)</th>
<th>Thickness (mm)</th>
<th>Weight (g)</th>
<th>Deuterium (A/o)</th>
<th>Error (Est.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Surface</td>
<td>0.25</td>
<td>0.030</td>
<td>0.320</td>
<td>+/- 10.0%</td>
</tr>
<tr>
<td>1</td>
<td>Midwall</td>
<td>0.25</td>
<td>0.039</td>
<td>0.056</td>
<td>+/- 10.0%</td>
</tr>
<tr>
<td>2</td>
<td>Sub-surface</td>
<td>0.38</td>
<td>0.054</td>
<td>0.031</td>
<td>+/- 10.0%</td>
</tr>
</tbody>
</table>

Based on these results, the sampling procedure was modified to equip the tool with a wide cutter to remove the oxide and some tube material (including the enriched layer). The oxide cutter was then replaced with a narrower cutter to obtain a sample in the area previously cleaned of oxide. This approach of surface preparation proved successful.

The oxide scrapes were retained and analyzed separately from the pressure tube samples. The results indicated that the oxide scrapes had very high deuterium concentrations. The samples taken from the same locations following the oxide scrapes showed excellent agreement with the throughwall pellets and followed the general trends for deuterium concentration. Subsequent work verified the credibility of the proposed sampling process (Figure 1) with pressure tubes from several sources.

In order to investigate the location of the deuterium-enriched layer and establish limits for the depth of oxide scrape, a series of oxide scrapes and samples were taken. The depth of the oxide scrape was successively set at 0.025 mm, 0.050 mm and 0.075 mm while the nominal depth of the sample remained constant at 0.10 mm. The deuterium analysis results showed no correlation with the varying depth of oxide scrape, indicating that the enriched layer was within the 0.025 mm band beneath the tube surface.

In performing these and other tests using the oxide scraper, it was found that, because the cutter was relatively wide, the depth and width of cut obtained was very dependent on the geometry of the pressure tube in the region being sampled. Local changes in tube diameter (e.g., ovality, flat spots) were directly reflected in the oxide scrape. This could result in scrapes that were less than 0.025 mm deep, or non-symmetrical scrapes that would produce samples with some amount of oxide covering. Therefore, from a practical point of view, the depth of oxide scrape was set at 0.050 mm to 0.075 mm to cover such eventualities.

4. THE SECOND GENERATION: TOOLS FOR PICKERING UNIT 4

There were two versions of the tool (Figure 2a) which were identical in every way except for the geometry of the cutters and cutter holders. The first tool was equipped with a 19 mm wide cutter set to remove a 0.025 mm to 0.050 mm thick strip, 55 mm long from the pressure tube wall to prepare the surface prior to taking the sample for deuterium analysis. The cutter of the second tool, the sampling tool, was 9.5 mm wide and was set to remove a 0.10 mm thick sample 38 mm long from the pressure tube in the location previously prepared.

A tool consisted of a tool head, a long push rod attached to the tool head to position the tool in the pressure tube, and a controller to sequence the cutter movements.

The tool head was approximately 65 cm long and 102 mm in diameter. It was mainly of aluminum construction, housed in a stainless steel sheath and weighing about 16 kg. Nylon rollers at the front and rear aided easy installation into the fuel channel.

After being located for sampling, the head was clamped to the pressure tube by extending pneumatically actuated pistons at the front and rear of the tool. The pistons were located in the lower half of the tool head and served to lift the head and force it into contact with the top surface of the pressure tube. A pressure of 1.7 MPa was required to ensure no movement of the tool head during the cutting operations.

The sample preparation (oxide removal) tool had a specially ground carbide cutter that was mounted in a shoe to obtain a specified standoff (generally 0.050 mm). The cutter assembly was raised through a window of the tool-head casing and simultaneously driven forward along the axis of the pressure tube. The depth of cut was limited to the amount of standoff set between the end of the tool bit and the shoe. By simultaneous use of the axial drive and lift motors, the tool bit was gradually fed into the tube wall. Near the end of the cut, the cutter was lowered to obtain a gradual exit geometry and to retract the cutter into the tool-head housing.

![Figure 1: Verification of the Sampling Process Deuterium Analysis Results](image-url)
When the sample preparation cut was complete, the pneumatic clamps were released and the tool withdrawn from the fuel channel. The cut material was removed from the cutter shoe cavity by inverting the tool and allowing the material to fall into a shielded container below.

A second tool equipped with the sampling tool bit was then inserted into the fuel channel and positioned in the area previously prepared. The exact axial and circumferential positioning of the sampling tool was made possible by careful calibration of push rods for both the preparation and sampling tools prior to use.

The sampling tool operated in a similar manner to the oxide removal tool, but gave a deeper (0.10 mm), narrower (5 mm) and shorter length (38 mm) cut. The sample obtained was normally in the form of a single curl and was retained in a cavity of the cutter shoe. The sample was retrieved by tool inversion once removed from the channel.

The positioning system was simple and effective. It consisted of a 1 m section of 5 cm diameter aluminum pipe that remained attached to the tool head for handling purposes, and a 4.6 m extension rod. Quick release electrical and air connections at the end of the extension rod served to connect it via a 50 m extension cable to the controller.

5. FEEDBACK FROM USING THE SECOND GENERATION TOOLS

On completion of the Pickering-4 outage, a review meeting was held to present results of the scraping process, discuss operational aspects of the outage and review performance of the sampling tools.

While the concept and performance of the sampling tools were acknowledged as excellent, there were four main problem areas identified:

- Some contamination of the sample surface with oxide occurred due to local irregularities in tube geometry (particularly at the 5.0 and 5.8 m positions) and due to alignment problems associated with sequential use of two tools to first prepare the area and then to take a sample.

- A concern regarding possible scoring of the pressure tubes was addressed by reducing the lift-motor current and routinely checking for pickup on the leading edges of the cutter shoes.

- The push-rod joint was difficult to make and break. The flanged joint caused frustration and deteriorated with use.

- The tools and controllers could not be successfully operated by some of the maintenance personnel.

All of these areas of concern were addressed and corrected in tools designed and subsequently supplied for use at Pickering-3 and Bruce-3.

6. THE THIRD GENERATION: TANDEM SAMPLING DRY TOOL HEAD

In early 1988, the tooling was redesigned to combine the oxide removal and sampling operations in one tool.

The tool head (see Figure 2b) contains two cutter cartridge assemblies mounted in tandem in a common carriage. An arrangement of motors and gears is used in conjunction with a pre-programmed automatic controller to sequentially raise and lower the oxide and sample cutters. The length, width and depth of cuts taken are similar to those obtained previously. The oxide removal cutter was trimmed from 19 mm to 13 mm wide to reduce the effect of tube out-of-roundness on the depth of the cut. New cartridges with quick-release bayonet mounts were designed for the tool cutter bits. This allows operators to have replacement cartridges on-hand with preset cutters so that a speedy exchange is possible if a cutter becomes worn or damaged.

The long push rod was replaced with shorter 2.4 m section rods for ease of handling. A new coupling was also designed for the tool head/push rod/controller system cable connections. This coupling makes the mechanical, electrical and pneumatic connections in a single quick action.

The controller for the tandem sampling tool has two modes of operation: automatic or manual.

In the automatic mode there are two selectable programs for (1) cutting a sample and (2) repositioning of the cutter module to facilitate sample retrieval. Execution of these programs is interlocked with tool pneumatic clamp pressure. Progress is indicated by lights on the front panel. There is no operator involvement required in automatic mode.

In the manual mode, switch indicators, meters showing motor load, and a three-channel strip-chart record give the operator a comprehensive view of system status and performance. The operator has full control over the tool motors.

If abnormal operation occurs during the automatic program, the controller shuts motor operation off. This may be triggered by any of the following:

- excessive cutter load,
- incorrect motor operation during automatic mode,
- blowing of a fuse, or
- activation of a limit switch.

During the design of the controller, particular emphasis was given to the following features:

- operator safety,
- protection of pressure tube, tool and controller against damage through malfunctioning of the controller,
- comprehensive information to the operator to prevent damage to pressure tube, tool and controller through false operation,
- a reduced set of indicators and controls for operators with little training, designed to minimize operator anxiety,
- quick selection and change of current setpoints for automatic control for fine tuning of cutting performance at site, and
- high reliability.

7. THE NEXT MAJOR STEP IN PRESSURE TUBE SAMPLING TECHNOLOGY

The major time-consuming operation during sampling is freezing off the feeders prior to draining a fuel channel. Typically, this operation takes 12 hours. If a tool could be built to operate in an undrained channel, significant cost and man-rem reductions could be achieved.
During the summer of 1987, following the successful sampling of NPD pressure tubes, work on a "wet" sampling tool began. Following proof of concept tests, a prototype wet sampling tool was built and bench tested.

A notable innovation for the wet tool is the use of the fuelling machine to deliver the tool to the sample location. Before the wet tool head can be used with confidence in conjunction with a fuelling machine, a significant amount of testing on mock-ups and in fuelling machines is necessary. The need for extensive testing can be readily seen since, unlike the 'dry' tools that give feedback continuously about the tool operation, there is essentially no feedback with the wet tool.

An interim approach is to mount a wet-type tool head on push rods and use it in an empty channel to obtain pressure tube material samples. For want of a better description, this is called the 'damp' sampling tool. This approach has the advantages of significantly reduced tool complexity and cost, and faster operation.

8. THE FOURTH GENERATION: TANDEM SAMPLING DAMP TOOL HEAD

The damp pressure tube tandem sampling tool (see Figure 2c) consists of a cylindrical tool head approximately 66 cm long and slightly smaller than the inside diameter of a pressure tube. The tool head contains two spring-loaded cutter cartridge assemblies mounted in tandem in a common carriage. In use, a 6.5 cm diameter pneumatic cylinder pulls the carriage in the tool body over a pair of ramps. Each cutter cartridge assembly passes over one of the ramps, causing the cutters to rise in turn and contact the pressure tube. The length, width and depth of cuts taken are similar to those obtained using the tandem dry tool.

Since diametral growth of pressure tubes occurs with time, the tool head is designed to cope with a variation in pressure tube diameter of up to 3.2 mm. This range is readily adjustable by shimming certain tool components. For instance, for relatively new tubes, the tool head could be set to sample tubes with diameters in the range 105.7 mm to 108.9 mm. Resetting the tool head would give access (for example) to pressure tube diameters in the range 105.7 mm to 108.9 mm. This latter dimension is the maximum expected deviation from the design value over the life of the pressure tube.

This tool was used successfully during the 1989 April/May outage at Point Lepreau NGS to obtain 30 samples from 10 channels. Operation of the tool is very quick; after locating the tool in the fuel channel, a sample can be cut in less than 30 seconds, as compared to 30 minutes with the tandem dry tool.

9. THE FIFTH GENERATION: WET PRESSURE TUBE SAMPLING

The wet sampling tool design incorporates many features that were developed and proven in earlier designs, its nearest relative being the damp tool. New features primarily relate to delivery of the tool by the fuelling machine into a flooded channel. The tool is 1 m long, the same length as two fuel bundles, for ease of handling by the fuelling machine. The tool will be used with the reactor shut down but with maintenance cooling flow of about 45 L/min. Accordingly, the cutter cartridges are equipped with retractable plates that ensure the oxide and sample specimens are retained in their respective cartridges. The interface with the fuelling machine allows axial and circumferential positioning of the tool in the fuel channel after some fuel bundles have been removed. The tool is butted against the remaining fuel bundles which maintain the tool axial position. The fuelling machine ram actuates the tool to cut the sample. By operating both fuelling machines in concert, essentially any axial location can be sampled.

10. SUMMARY: EVOLUTION OF PRESSURE TUBE SAMPLING TOOLS

Once the "first generation" tools had proved the validity of the sampling concept at NPD, pressure tube sampling technology evolved to meet the needs of the CANDU power reactors at Pickering, Bruce and Point Lepreau. The major milestones in tool development and application are listed below.

<table>
<thead>
<tr>
<th>Date</th>
<th>Site</th>
<th>Reactor Site</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>'87 May</td>
<td>NPD</td>
<td>First tandem tool</td>
<td></td>
</tr>
<tr>
<td>'87 Dec. 2</td>
<td>Pickering-4</td>
<td>Separate tools for oxide and sample removal</td>
<td></td>
</tr>
<tr>
<td>'88 April 3</td>
<td>Pickering-3</td>
<td>Improved tandem tool</td>
<td></td>
</tr>
<tr>
<td>'88 Oct. 3</td>
<td>Bruce-2</td>
<td>First tandem tool</td>
<td></td>
</tr>
<tr>
<td>'88 Nov. 3</td>
<td>Pickering-4</td>
<td></td>
<td></td>
</tr>
<tr>
<td>'89 April 4</td>
<td>Point Lepreau</td>
<td>First damp tool</td>
<td></td>
</tr>
</tbody>
</table>

Over the past two years, the extensive comparisons that have been made between the D levels of scraped samples and adjacent punched samples in removed pressure tubes have shown excellent correlation between the two approaches. The in-situ pressure tube sampling technology released and been shown to be fast and reliable. It has become the standard method of assessing the D levels in pressure tubes during multi-channel inspections.

11. REFERENCES


FIGURE 2: Various Stages in the Evolution of the Pressure Tube Sampling Tool. From Top to Bottom: (a) Second Generation: First Pickering Tool; (b) Third Generation: Tandem Tool; (c) Fourth Generation: Damp Tool
INTRODUCTION

The CANDU fuel channel assembly, see Figure 1, is made up of a zirconium - 2.5% niobium alloy pressure tube of nominal 103.4 mm inside diameter, 4.3 mm wall thickness, and a length of 6.3 m. Each end of the tube is attached by means of a cold-rolled joint to a stainless steel end fitting which allows for the coolant piping connections, as well as containing a closure plug that can be removed by automated machines to enable on-power refuelling. External and concentric with the pressure tube is a thinner tube of Zircaloy-2 alloy, known as the calandria tube, that separates the hot pressure tube from the cool moderator. Dry nitrogen or carbon dioxide gas fills the annular space between the tubes, and up to four "garter spring" spacers are intended to keep the tubes from touching each other.

Because of the critically important role of the pressure tubes as part of the primary pressure boundary, it is necessary to confirm their structural integrity and collect data related to the long-term materials performance in order to predict maintenance requirements and assist in the design process. In order to address these needs, Ontario Hydro maintains two inspection programs for pressure tubes. The Periodic Inspection Program (PIP) involves inspecting a small sample of tubes in each reactor on a recurring basis to ensure the detection of generic problems. The requirements for the PIP are mandated by the regulatory authority, the Atomic Energy Control Board (AECB). The second program, the In-Service Inspection program (ISI), is short-term in scope and is directed by Ontario Hydro with AECB concurrence.

It aims to provide specific information on known problems or on populations of tubes that are known to be at greater risk. The objectives are changed as the requirements for information change.

In order to satisfy these inspection programs and also minimize both outage time and radiation exposure to personnel, highly automated inspection systems have been developed. The first system that will be described is known as CIGAR - Channel Inspection and Gauging Apparatus for Reactors. CIGAR is used to perform the general surveillance monitoring of pressure tubes in order to meet the needs of both the PIP and ISI programs. The other system that is the subject of this paper is PIPE - Packaged Inspection Probe. This system was developed to perform a limited scope inspection on a large population, in as short a time as possible, in order to comply with the ISI. The description in this paper will concentrate on the hardware and its use, rather than the details of the NDE techniques employed.

CIGAR

Overview

In addition to flaw detection, information is required on tube diameter and wall thickness, tube sag, and garter spring location along the tube. These parameters can all be measured within 6-12 hours per channel.

The reactor must be shut down with the heat transport system cold and depressurized, and fuel channels to be inspected must be defuelled. The radiation fields experienced within the tube are still in the order of 10-6 R/Hr gamma.

The CIGAR system consists of an in-channel package comprising a modified channel closure plug and an inspection head, which is driven axially and rotationally within the tube by means of a drive mechanism mounted on the reactor's fuelling machine support platform. The drive mechanism connects to the inspection head mechanically and electrically by means of drive rods that pass through an O-ring sealed orifice in the closure plug. Electrical power to the drive mechanism is supplied by a "local" control console, situated within containment on the reactor vault floor. This console contains the motor drive units, power supplies, and interface circuitry to communicate with the control computer located at a remote site outside of the containment boundary. The "remote" console contains the control computer - a DEC PDP 11/23, the data acquisition computers, and the inspection instrumentation. Figure 2 shows the overall layout of the system, and Figure 3 the remote consoles.
The inspection head, see Figure 1, is modular in design, allowing it to be configured for different applications by the substitution of various transducer modules. In addition to the transducer carrying modules, the head has two sprung centering modules that act to maintain its axis coincident with the mean axis of the pressure tube while it is being moved. A universal joint prevents radial loading from the drive rod being transmitted to the head, and a tailpiece provides the mechanical and electrical mating configuration to the drive rod. A different configuration of the inspection head that has been used combines the ultrasonic flaw detection and garter spring detection capabilities with a surface profilometry module consisting of a fine tipped stylus connected to an LVDT. This enables accurate measurement of inside surface discontinuities.

The drive rod contains 15 miniature coaxial cables terminated with special connectors at either end. Alignment between the drive rods and the inspection head is ensured by a keyway, and a sprung button acts as a locking outboard end of the drive rod attaches to a ball-screw driven carriage that runs on roundways on the drive mechanism. Rotational drive is provided by a keyed shaft driving spur gears, and the use of a 30-conductor slip ring unit conveys the signals from the drive rod cabling to fixed cabling on the drive system.

Independent d.c. electric motors are used to provide axial and rotary motion to the drive rod thus allowing maximum flexibility for inspection head scanning patterns. Additional a.c. motors are used to give a fine X-Y positioning capability of the drive rod with the fuel channel to be inspected. Two rods are required to inspect the full length of a channel and the second of these is stored in a motor driven, retractable magazine when not required. A single rod design was not feasible due to dimensional constraints of the reactor vault airlocks.

Under normal operating conditions, the equipment once set up is operated entirely from the remote console located in an inspection trailer. Preprogrammed sequences are used to align the rod with the fuel channel, connect the rod to the inspection head, perform the scans, and load and unload the second rod as required.

The data acquisition system contains two computers, a Compaq 386 PC and a Plessey mini-computer, which is based upon a DEC 11/73 processor. The Compaq 386 performs data acquisition and processing for channel gauging, sag measurement, and garter spring location. It generates output on a Hercules monochrome graphics display and hardcopy on an Epson dot matrix printer. The PC contains a high speed 12-bit analogue-to-digital converter card, a 96-line digital I/O interface card, a 40 MByte hard disk, 1.2 MByte floppy disk, and a serial port.

The data acquisition and processing for the ultrasonic flaw detection is handled by the Plessey computer. It contains an analogue-to-digital converter card, digital I/O capability, five serial ports, and both a hard and floppy disk. The Plessey directs its output to two Qume stand-alone graphics terminals and produces hardcopy on the Epson printer.

The Compaq and the Plessey computers operate under the control of the control computer during data collection. The control computer communicates with the data acquisition computers via parallel digital links. During off-line data playback, these computers are under the control of the system operator.
Ultrasonic Flaw Detection System

The purpose of this system is to enable a 100% volumetric inspection of the pressure tube for defects to be performed. During this scan, the inspection head is rotated at 20 rpm and advanced axially at 1 mm/s, generating a helical scan pattern of 3 mm pitch. Four 45° shear-wave transducers are arranged as a circumferential and an axial pair, see Figure 5, with the beams convergent on a 1-2 mm diameter spot on the inside surface. The transducers are 9.5 mm active diameter, 10 MHz, highly damped, and spherically focussed at 33 mm in water. They are pulsed sequentially by a Krautkramer-Branson KB6000 multichannel ultrasonic flaw detector and their responses displayed on the internal CRT. Electronic gates in the instrument allow, for each transducer, analog signals proportional to the ultrasonic response from the interface (ID), half-skip (OD), and full-skip (ID) to be output.

During the scan, eight analogue signals from the KB6000 are recorded directly onto magnetic tape by a Racal FM recorder. These signals are passed from the recorder to the analogue-to-digital converter within the Plessey computer. The operator may select any two signals or two combinations of signals for on-line display. This display takes the form of two isometric data plots of signal amplitude versus axial and rotary position, see Figure 6. The graphics are generated on two Qume terminals connected to the Plessey, which allows one to display data while the other is dumping its screen image to the printer. With this system, no data is lost during the inspection.

Off-line playback capability allows the operator to select different signals or combinations of signals to display each time the data is replayed. In addition to the helical scan, detailed scans are available, which allow the operator to scan small areas of the pressure tube, using an axial or rotary scanning pattern. These scans yield better resolution but are otherwise very similar to the helical scan. Helical scans can also be performed at finer pitches down to about 1 mm and are typically used in the rolled joint regions and other areas of interest.

Ultrasonic Gauging System

The purpose of this system is to acquire information on the pressure tube inside diameter and wall thickness. This is achieved by means of three ultrasonic transducers of the same type used for flaw detection. Two are mounted a fixed distance apart, diametrically opposed, measuring water path distances to the inside wall. The third, known as the reference transducer, is aimed at a fixed target that yields two reflections a known distance apart for velocity calibration. The transducers are connected to a multichannel ultrasonic thickness meter, a Krautkramer WDM-U. Measurements made within a calibration ring of known dimensions enable correction constants to be determined and subsequently applied during the data processing.

Gauging data is collected simultaneously with the flaw detection scan of the pressure tube. This represents approximately 2,000 revolutions of the head, during each of which 20 equally spaced samples of gauging data are collected by the Compaq and stored on hard disk. During the collection process, raw data is continuously displayed on the monochrome monitor which allows the operator to confirm the correct operation of the system.
After data collection is complete, the operator can view the data in both raw and processed form and produce a report. This report contains tables of pressure tube wall thickness and diameter variations. Conditions such as excessive wall thinning or pressure tube ovality are automatically flagged in the report. The information is also displayed graphically in the form of parameter versus axial position plots as shown in Figure 7.

**Sag Measurement System**

The combined effects of neutron flux and high temperature cause the pressure tubes to sag during their lifetime. It is important for reasons of integrity and future design that information be obtained on the extent of this sag deflection. The sag measuring module on the inspection head contains a servo-accelerometer, encased in lead shielding to protect the electronic components. This device is highly sensitive to its angular position with respect to an acceleration force, gravity in this instance, and can detect angular differences as small as 0.1 seconds of arc. The readout instrumentation consists simply of power supplies and a 4 1/2 digit DVM with BCD output capability.

In use, slope measurements are made with respect to the earth's gravitational field at the rate of one pressure tube slope measurement for each 30 mm of pressure tube length. During data collection, raw slope values are displayed, allowing the operator to confirm correct operation of the system. Off-line, slope values are converted into a pressure tube sag profile by computing the integral of the slope measurements made over the length of the pressure tube. Adjustments are made to correct for measurement bias and to remove the effect of overall pressure tube slope. Off-line sag processing requires approximately 2 minutes to complete and produces a table of sag values versus axial position and graphical output. Figure 8 shows an example of the graphical output.

**Spacer Location System**

The axial position of the "garter spring" spacers is critical in ensuring that contact does not occur between the hot pressure tube and the cool calandria tube, a condition that is now known to have serious metallurgical consequences for the pressure tubes. These spacers are made of a helically wound coil through which is threaded a wire, the girdle wire, to maintain the shape. The function of this inspection system is to accurately locate the spacers along the tube. Use is made of a send-differential receive eddy current coil that couples with the girdle wire of the spacer. The coil is connected to a Forster Defectomat F eddy current test instrument with the presence of spacers showing as Figure 8 responses on the instrument's impedance plane display.

Data to locate these spacers is collected at 1 mm intervals, while the inspection head is propelled axially through the pressure tube. Each data point consists of an x and y eddy current signal, and these values are displayed on-line during data collection.

Off-line playback of garter spring data is an interactive, graphically oriented process carried out by the system operator. Raw eddy current data is first presented to the operator as plots of x and y eddy current signals versus axial position. The operator identifies a signal which may represent a garter spring and then expands the x and y plots of the data containing this signal. The operator then positions a narrow window over the suspected garter spring, and the computer plots the windowed data as y versus x. If the operator...
confirms the signal to be that of a garter spring; the signal location and y versus x plot are recorded. This process is repeated until all of the spacers are located and the output shown in figure 9 is produced.

![Diagram of spacer location](image.png)

**FIGURE 9: SAMPLE SPACER LOCATION PLOT**

**Operational Experience**

The CIGAR equipment has been in-service use since September 1985 and has contributed considerably to Ontario Hydro's inspection programs since then. The extent and frequency of pressure tube inspections have increased dramatically since a tube ruptured in 1983 and another in 1986. The equipment was originally designed around late 1970's inspection program requirements of some 20 tubes/year. Despite the much higher utilization, 270 tubes to March 1989, reliability has still been reasonably good. Many changes have been made to the hardware and software in nearly 4 years of use, to improve both capability and durability.

Significant reductions in both outage duration and personnel radiation exposure have been realized with the use of this equipment, and the amount and quality of NDE information obtained has also increased substantially over previous equipment.

**Future Developments**

Within the next 3 months, it is planned to completely upgrade the flaw detection data acquisition and processing by switching to a system based on another Compaq 386 based-PC equipped with a high speed analogue-to-digital converter, high resolution colour graphics capability, and high capacity optical disk drive. Using this system, up to 16 channels of analogue information can be digitized, and the output displayed either in isometric form, as with the current system, or as colour-coded C-scan plots. Hardware to allow the collection of individual A-scans and their presentation as B-scan plots will also be included. The remote console will be replaced at this time with one that will better accommodate the new hardware and all the additions since 1985.

Further related to the problems of pressure tube/calandria tube contact, it is highly desirable to directly measure the annular gap between the two tubes, rather than relying on modelling based on spacer positioning and sag results. In order to make these measurements from within the tube, an eddy current technique was developed. The send coil of the spacer location system is again utilized to generate a large energizing field of 4 KHz, while a small receive coil is displaced approximately 70-80 mm axially into the weak part of the field. This design gives equal sensitivity to both inside and outside effects, and phase adjustment allows response to changes in gap to be aligned with the Y-axis of the complex impedance plane display.

To predict pressure tube/calandria tube gap at a given location, it is necessary to record pressure tube wall thickness and the response of the eddy current sensor at that location. Wall thickness data is collected by the gauging system as described above, and while this is occurring, eddy current data is simultaneously collected and stored in a data file.

Due to limitations of the number of individual conductors leading to the inspection head, incorporation of this system is dependent on the inclusion of a cable sharing on-head multiplexer that has been developed. These enhancements are planned within the next 12 months.

Work is also underway to include eddy current flaw detection capability to complement the ultrasonic system for certain types of defects, thus enabling better characterization and depth estimation of detected flaws. The addition of an underwater television camera inspection head to interface with the CIGAR drive unit is also being actively pursued to add visual information about a defect to aid in the interpretation of the other NDE signals. Other specialized modules containing different arrangements of ultrasonic transducers are also envisaged to optimize the detectability of potential hydride blisters on the outside surface of the pressure tube.

**Pipe Overview**

In certain reactors, pressure tubes were inadvertently over rolled during installation, resulting in high residual stress and some subsequent cracking by a hydride precipitation mechanism. These Delayed Hydride Cracks (DHCs), as they are known, are of concern since if they propagate, eventually leakage can result. In order to test a large population of these rolled joint regions for the presence of DHCs, an inspection system utilizing the reactor fuelling machines was developed.

The fuelling machines are robotic devices designed to enable on-power refuelling of the CANDU reactors. They have the capability to home on to an individual fuel channel and make a high pressure seal to it. Mechanically, there are a number of differences between the fuel handling systems for the Pickering style reactors to those at Bruce. Although hardware was designed for both stations, the system has to date only been used at Bruce, and
so the following description will be specific to that configuration. The machine has a mechanically driven charge tube and an inner ram to enable it to handle components and push fuel. A pair of these machines work in parallel, clamped to either end of the channel. By designing an ultrasonic inspection head mechanically compatible with the fuelling machine and making some minor modifications to the machines themselves, they became inspection tools.

Because of temperature and radiation limits for the ultrasonic transducers and cabling, the inspection must be performed with the reactor shutdown and in a cold, depressurized condition.

The inspection is targeted at finding axially oriented, inside surface defects over a 100 mm long section of the tube at the rolled joint region. Both ends of any tube can be inspected within 2 hours, including the time to open and close the channel.

The transducer carrier has an outer casing, the transducer block, and a push rod with electrical connector. The outer casing mates mechanically with the fuelling machine charge tube at the rear and contains a calibration tube at the front. The push rod mates electrically with the fuelling machine ram which has been modified to carry an electrical connector and cabling to the rear of the machine. The transducer block, shown in Figure 10, is mounted on the front of the push rod and carries 11 ultrasonic transducers similar to those used on CIGAR. Eight of these are arranged as four circumferential looking pairs with each transducer separated axially by 1 mm from each other. In this way, a single 360° rotation of the head scans an 8 mm band of the tube. The three remaining transducers are arranged as an axial-looking pair, and a normal beam probe for characterization purposes only.

The electrical cabling is conveyed to the rear of the machine where it is stored on a motor-driven winch assembly. No slip rings are used, the cable passing through a potted seal and then stored on a contra-rotating drum on the outside. The cable on the drum is in the form of a flat assembly of fifteen miniature coaxial cables. Cables from the rear of the fuelling machine then lead from the reactor vault via a penetration, to the station control room, where the data acquisition console is located adjacent to the station fuelling machine control panels. The fuelling machine computers are loaded with modified software to permit the execution of the inspection scan sequence.

The PIPE data acquisition system is controlled by a Compaq 386 PC with a 40 MByte hard disk and a 1.2 MByte floppy disk. To this PC have been added a 16-channel, high-speed analogue to digital converter card, a digital I/O card, and an optical disk drive sub-system. The Compaq synchronizes its operations with the fuelling machine control computers through a digital interface consisting of four signal lines.

The Compaq is connected to both a Hercules monochrome display and an EGA colour graphics display. The monochrome display is used for the operator interface which is completely menu-driven. An area of the display is reserved for error and status messages, the balance being used for menus and data entry panels. The colour display is used for the graphic presentation of inspection data and by several utility functions. Hardcopy, an example of which is shown in Figure 11, is produced on an Epson LQ2500 colour dot matrix printer.

System Calibration

Initially, the responses of each of the eight circumferential transducers to the calibration notch are equalized by means of variable attenuators. During this operation, the computer displays the signal strength of each transducer in the form of a histogram to aid the operator in making the adjustments.

Before and after each rolled joint is inspected, a system check scan is performed by rotating the transducers over the calibration notch and recording the signal levels. These scans are very similar to the inspection scans and produce similar displays. After completion of a system check scan, the computer analyzes the data collected for acceptance against threshold values.
Rolled Joint Inspection

The rolled joint inspection is performed by rotating the inspection head back and forth in a raster pattern over the rolled joint area. The collected ultrasonic data is displayed in real-time as a pair of colour encoded isometric plots which represent the interface and full-skip responses. Hardcopy is usually produced between inspections.

Mass Storage Sub-System

A typical rolled joint inspection produces approximately 2 Mbytes of data, and an inspection program may call for 150 rolled joints to be scanned during one outage. To deal with these huge volumes of data, the Compaq is equipped with a Write Once Read Many (WORM) optical disk sub-system capable of storing 230 Mbytes on a removable cartridge. During the data collection, data is written to hard disk because of its high access speed. Then after the collection is complete, the data is transferred to the optical disk.

Operational Experience

PIPE has proved to be extremely successful, going from concept to first use in just over 1 year. Inspection rate averaged 10 channels, 20 rolled joints, per day once commissioning was completed. The best inspection rate that could be achieved for this type of inspection with CIGAR would probably be two channels per day and thus a significant saving in outage duration has been realized.

An extension to the PIPE concept, known as BLIP - Blister Inspection Probe, has been developed, although not yet used, for the Pickering configuration. BLIP is targeted toward a large-scale inspection for outside surface hydride blisters. Inspection using six line-focused ultrasonic transducers, arranged as three axial looking pairs, is confined to a 60° arc centered on the bottom of the tube. Standard CIGAR-type eddy current garter spring detection capability is also included on the transducer carrier. At present, a single modified fuelling machine can inspect half the entire length of the pressure tube.

CONCLUSION

In conclusion, during the past 10 years, Ontario Hydro has devoted a lot of effort in developing remote inspection capability for one of the most inaccessible and critical components of its CANDU nuclear units. The investment has paid off many times in reduced employee radiation exposure, unit outage duration, and quality of NDE information obtained.

ACKNOWLEDGEMENTS

The design, development, and utilization of the equipment described above is the work of many groups of people, both within Ontario Hydro and in other companies. M.P. Dolbey et al of Research Division were responsible for the NDE systems, D. Booth and D.W. Murray of GE Canada designed the CIGAR drive mechanism, and D.M. Hayter and E. Di Stanislao were responsible for project coordination of CIGAR and PIPE respectively.

Atomic Energy of Canada Ltd have made significant contributions to Pickering/600 MW applications of both CIGAR and PIPE. Finally, the field application of both systems has been aided greatly by S. Clements et al at Bruce NGS and H. Underhill et al at Pickering NGS.
COMPARISON OF MEASURED AND CALCULATED DOSE RATES AND ACTIVITIES DURING PRESSURE TUBE REPLACEMENT IN PICKERING UNITS 1 and 2

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ABSTRACT

The shielding codes ORIGEN, DOT3.5, and QAD-CG have been used to calculate specific activities in and the dose rates from fuel channel components removed during the Pickering Units 1 and 2 Large Scale Fuel Channel Replacement (LSFCR) program. The calculated activities and dose rates have been compared with available measurements. The results give confidence that the methodology can be applied to radiation dose assessment associated with LSFCR program for Units 1, 2 and 3, 4 of Pickering reactors or other CANDU nuclear power stations.

CALCULATION OF SPECIFIC ACTIVITY

The isotope generation and depletion code ORIGEN (1) was used in the flux mode to calculate the specific activity of various fuel channel components including pressure tubes for Pickering Units 1, 2 and 3, 4. These calculations are described below:

Pressure Tubes

An ORIGEN calculation for the Zr-Nb pressure tubes in Units 3 and 4 was run for an irradiation time of 6400 days. The irradiation was made for an average bundle power (viz., 1653 MW/(390 channels x 12 bundles per channel) = 353kW) and at 80% capacity factor.

The fluxes were taken from a WIMS(2) calculation of fluxes in a 28-element lattice cell for a burnup of 3500 MW.d/MgU. The capacity factor was included in the thermal flux, 8.62 x 10¹³ n/cm².s, used in the irradiation. The values of the neutron spectrum indices, THERM, RES and FAST used were 0.831, 0.052 and 0.228 respectively.

The composition of the Zr-Nb alloy was the average of six randomly selected test certificates. The cobalt impurity level was reported to be less than 5 ppm, but taken to be 5 ppm in the calculations.

Specific activities of a Zircaloy-2 pressure tube were estimated from those of the Zr-Nb pressure tube by correcting for the chemical compositions. Table 1 presents the calculated specific activities for a Zircaloy-2 pressure tube associated with an average power bundle 269 days after shutdown.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Specific Activity (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr-51</td>
<td>3.77E7</td>
</tr>
<tr>
<td>Mn-54</td>
<td>3.48E8</td>
</tr>
<tr>
<td>Fe-55</td>
<td>1.58E11</td>
</tr>
<tr>
<td>Fe-59</td>
<td>2.84E8</td>
</tr>
<tr>
<td>Co-60</td>
<td>4.55E10</td>
</tr>
<tr>
<td>Zr-65</td>
<td>6.90E9</td>
</tr>
<tr>
<td>Sr-89</td>
<td>4.65E7</td>
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<tr>
<td>Y-91</td>
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<tr>
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<td>6.00E11</td>
</tr>
<tr>
<td>Sr-113</td>
<td>2.19E10</td>
</tr>
<tr>
<td>Sr-124</td>
<td>1.10E10</td>
</tr>
<tr>
<td>Sr-125</td>
<td>1.57E11</td>
</tr>
<tr>
<td>Hf-181</td>
<td>2.13E9</td>
</tr>
<tr>
<td>Ta-182</td>
<td>8.20E9</td>
</tr>
</tbody>
</table>

NOTE: 3.7E10 Bq = 1 Ci

End Fitting, Pressure Tube Stub and Shield Plugs

The specific activities of the type 403 stainless steel end fittings and type 410 shield plugs, including the pressure tube stub were calculated using ORIGEN. The compositions of the type 403 and 410 stainless were based on measured chemical contents supplemented by typical levels of impurities. The cobalt concentration in the stainless steel was assumed to be 120 ppm.

The calculations were made for three locations in the shield plug and end fittings and one location in the pressure tube stub. The exact locations are shown in Figure 1. The irradiation was at 80% capacity factor for 6400 days and decay times from 1 h to 5 years.

Neutron Flux in Components

Under irradiation the pressure tube undergoes several centimetres of elongation or creep. The irradiation flux in the end fittings and shield plugs is a complex function of time, coolant flow
direction, reactor power and channel power. However, to gauge the sensitivity of the specific activities to these variables, the fluxes from DOT 3.5 (3) analyses made for the REFAB (Repositioning End Fittings and Bearings) program for a central fuel channel were reviewed.

In the Pickering A reactors, the pressure tube creep is taken up at one end of the reactor by clamping all the end-fittings at one end of the reactor, the so-called fixed end, and accommodating the creep at the free end. Consequently, with the bi-directional flow of coolant through the channels, there are four types of fuel channel penetrations.

![Diagram of fuel channel penetrations](image-url)
The behaviour of these four types of fuel channel penetrations is shown in Figure 2. The six sketches in the upper half of the figure show the changes in end shield arrangement for a coolant flow from right to left and in the lower half for a coolant flow from left to right.

At the beginning-of-life positions, the fuel channel/shield plug interface is at the calandria side tubesheet. This is shielding configuration A in Figure 2.

As the pressure tube creeps, the fuel channel components and the fuel string are pushed into the end shield at the downstream and free end of a fuel channel (configuration B), and the fuel string moves into the core for the upstream and fixed end of the channel (configuration D). The latter produces a coolant gap between the end shield and the fuel string. This is configuration C in Figure 2. The downstream and fixed end of the channel, the shield arrangement is unchanged (configuration A).

After re-positioning fuel channel components, configuration A develops for the downstream and free end of fuel channels, while configuration C develops for the upstream and the fixed end of the channels. Conversely, configurations B and D develop for the downstream and fixed end of channels, and for the upstream and free end of channels respectively. With further irradiations, configuration D develops into a fifth configuration E in which the coolant gap opens up between the end bundle and the shield plug (as the pressure tube, end-fitting and shield plug creep into the end shield).

Since the pressure tube elongation in Units 3 and 4 pressure tubes will range from 4.5 cm to 6.0 cm over the irradiation period, the fluxes in the end fittings and shield plugs will range from those with configuration A to the second set of configuration B fluxes. Table 2 shows neutron fluxes in three groups, viz., fast ($E_n > 0.82$ MeV), intermediate ($0.6337 < E_n < 0.414$ eV) and thermal ($E_n < 0.0$) for several of these configurations.

For the activity calculation purposes, the configuration A fluxes were chosen after multiplying by 0.667. This is the ratio of average bundle power to the central channel bundle power for the end bundles, obtained from a time-averaged reactor physics calculation.

Values of THERM, RES and FAST derived from these fluxes and the thermal neutron fluxes used in ORIGEN calculations are given in Table 3. The calculated activities for Unit 2 channel G-16 shield plug are given in Table 4.

DOSE RATE CALCULATION AND COMPARISON WITH MEASUREMENTS

The activities obtained from the ORIGEN calculations and the decay schemes of the radionuclides (4) were used to find the decay gamma sources grouped in seven or more energy groups. The gamma sources were incorporated into the QAD-CG (5) code which is a three-dimensional point kernel integration code, to calculate dose rates from a) the end fitting and its pressure tube stub under water, b) four pressure tubes and eight shield plugs inside the fuel channel flask and c) 80-100 pressure tubes inside various dry storage modules (DSM's).
TABLE 4: ACTIVITIES CALCULATED BY ORIGEN FOR FIGURING UNIT 2 CHANNEL J-15 WEST END FITTING AND ITS PRESSURE TUBE STUB 130 DAYS AFTER SHUTDOWN

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>End Fitting</th>
<th>Pressure Tube Stub</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tip of E/F</td>
<td>Rear of Heavy Wall Section</td>
<td></td>
</tr>
<tr>
<td>Cr-51</td>
<td>1.37E10</td>
<td>5.85E9</td>
</tr>
<tr>
<td>Mn-54</td>
<td>7.71E9</td>
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<td>Fe-55</td>
<td>4.06E11</td>
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<td>Fe-59</td>
<td>1.95E9</td>
<td>1.02E9</td>
</tr>
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<td>Co-58</td>
<td>1.45E9</td>
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<td>Co-60</td>
<td>2.02E10</td>
<td>1.46E9</td>
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<td>Ni-59</td>
<td>3.98E5</td>
<td>1.71E5</td>
</tr>
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<td>Zn-65</td>
<td>9.29E5</td>
<td>3.05E3</td>
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<td>Sr-89</td>
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<td>Y-91</td>
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<td>Zr-95</td>
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<td>6.03E5</td>
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<td>9.27E5</td>
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<td>Ag-108m</td>
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<td>Ag-110m</td>
<td>2.17E5</td>
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<td>Sr-124</td>
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</tr>
<tr>
<td>Tc-182</td>
<td>-</td>
<td>6.59E10</td>
</tr>
</tbody>
</table>

Dose Rates from End Fitting and Its Pressure Tube Stub

The dose rate on contact with and at various distances from the end fitting and its pressure tube stub under water were calculated using the QAD-CG code. The pressure tube stub was represented by a tubular source 2.34 cm long (12.7 cm of which sticks out of the end fitting) while the end fitting was represented by three source regions in the QAD-CG calculation (see Figure 3).

Pressure Tube Stub Source Specification.

Taking into account a) the pressure tube stub mass of 2.6 kg and b) the axial source distribution (i.e., thermal flux distribution from DOT 3.5 calculations), the effective source mass which should be multiplied by the specific gamma source was obtained. The product gave the total gamma source in the pressure tube stub.

End Fitting Source Specification.

The specific activities for the tip of the end fitting in Table 4 were used for the first and second source region taking into account exponential attenuation along the latter. The gamma source distribution along the third source region was taken to be the gamma source at point X in Figure 3 with exponential attenuation along the end fitting. The exponential attenuation was taken from DOT 3.5 calculations.

Dose Rate Comparison

Figure 4 shows the comparison of the measured dose rates on contact with the end fitting and its pressure tube stub with those of calculations. The agreement is good. Note that the measured dose rate towards the tail end of the end fitting, although relatively small, is greater than the calculated dose rate. This was attributed to the deposited corrosion product activity on the end fitting surfaces. Figure 5 shows the comparison of calculated dose rates away from the end-fitting axis along direction 'A' and 'B' with those of measurements.

Dose Rates Around Fuel Channel Flask

The dose rates around the fuel channel flask (see Figure 6) containing four pressure tubes and eight shield plugs from row Q of fuel channels on Unit 1 were calculated using the QAD-CG code. The hot ends of shield plugs are about 1.25 m from the loading end of the flask. These components were taken to be decayed for two years.

Pressure Tube Source Specification.

Taking into account a) the maximum rated fuel channel ($P_{channel} = 5.34$ MW), b) the pressure tube mass of 168 kg, c) the axial source distribution along the channel (i.e., bundle power distribution), the specific activities of nuclides in Table 1 (after correcting for the extra decay time of two years) were converted into eleven group gamma sources.

Shield Plug Source Specification.

The simplified sketch of the shield plug used in the QAD-CG calculations is shown in Figure 7. Only the tip of the shield plug is active and the activity drops very rapidly along the shield plug.

![Figure 3 QAD-CG Model for End Fitting and Its Pressure Tube Stub](image-url)
For the dose rate calculation purposes, the plug was represented by four source regions. These regions are 2.4 cm, 32 cm, 18 cm and 36 cm long and weigh 0.24 kg, 4.3 kg, 8.6 kg and 20 kg respectively.

The gamma sources used in the calculations were based on the specific activities given in Table 5 corrected for the two-year decay period.
NOTE: ALL DIMENSIONS IN CENTIMETRES

FIGURE 7 SHOWING SKETCH OF SHIELD PLUG AND ITS FOUR SOURCE REGIONS
USED IN QAD-CG CALCULATIONS

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Tip of Shield Plug (Bq/kg)</th>
<th>Front of Main Body of Shield Plug (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr-51</td>
<td>3.81E9</td>
<td>1.85E7</td>
</tr>
<tr>
<td>Mn-54</td>
<td>2.46E10</td>
<td>9.21E7</td>
</tr>
<tr>
<td>Fe-55</td>
<td>1.73E12</td>
<td>8.55E9</td>
</tr>
<tr>
<td>Fe-59</td>
<td>1.44E9</td>
<td>8.73E9</td>
</tr>
<tr>
<td>Co-58</td>
<td>2.12E8</td>
<td>8.25E8</td>
</tr>
<tr>
<td>Co-60</td>
<td>7.99E10</td>
<td>6.59E8</td>
</tr>
<tr>
<td>Ni-59</td>
<td>1.96E7</td>
<td>9.99E4</td>
</tr>
<tr>
<td>Zr-65</td>
<td>1.09E7</td>
<td>5.25E2</td>
</tr>
<tr>
<td>Nb-95</td>
<td>1.67E6</td>
<td>2.25E4</td>
</tr>
<tr>
<td>Ag-103m</td>
<td>7.18E5</td>
<td>1.55E2</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>3.46E8</td>
<td>7.51E5</td>
</tr>
<tr>
<td>Sr-113</td>
<td>5.59E6</td>
<td>9.73E4</td>
</tr>
<tr>
<td>Sr-124</td>
<td>1.15E5</td>
<td>2.1</td>
</tr>
<tr>
<td>Sr-125</td>
<td>3.74E7</td>
<td>7.03E5</td>
</tr>
</tbody>
</table>

Dose Rate Comparison

The total dose rate on contact with the fuel channel flask at a point opposite the hot ends of the shield plugs adds up to about 7.0 mSv/h (4.2 mSv/h from pressure tubes and 2.85 mSv/h from shield plugs). This does not agree well with the measured dose rate (6) of 120-150 mSv/h made on 18/19 November 1985 (i.e., two years after shutdown).

Inspection of the activation fluxes in the shield plug tip area in the channel after creep showed that the activation flux used in the ORIGEN calculation before creep could be low by a factor of 2. Since all the dose rate received from the shield plug is coming from the tip of the plug, this translates to an increase in dose rates from 2.85 mSv/h to 5.7 mSv/h.

The cobalt impurity level used (i.e., 120 ppm) in the shield plug material could be as high as 500 ppm. If this is the case, then the calculated dose rates would be higher by a factor of 4.2. Now, the dose rate received from the shield plugs increase to 24 mSv/h, plus the pressure tube component brings the total dose rate to 27 mSv/h. This still leaves a discrepancy of a factor of 4 to 5. We have speculated that the fuel channel flask had some weak spots and the higher dose rates were measured opposite these locations. It is worthwhile to note that the attenuation achieved in reducing dose rates around the flask with 2.5 cm additional lead seemed too much, and implied that there was a larger contribution to the measured dose rate from gammas below 1.25 MeV (average energy for 60Co gammas).

Calculated & Measured Activities of Pressure Tubes

The activities of the Zircaloy-2 pressure tubes calculated by ORIGEN were compared with the measurements of activity in the pressure tube from channel G-16 of Unit 2.

Note that the ORIGEN activities were prorated by the factor 1.5 which is the ratio of the average bundle power used in ORIGEN to that of the time-averaged maximum bundle power in channel G-16. This is given in Table 6. ORIGEN calculations agree well with the measured activities (eight out of nine isotopes) in the pressure tube. Note that the measured activity for 181Hf in the pressure tube seems low.

Dose Rates Around Dry Storage Modules

The dry storage module shielding arrangement is shown in Figures 8 and 9. They contain 80-100 pressure tubes each. The dose rates outside the DSM Prince Charming containing 87 pressure tubes cooled for 136 years were calculated with the
QAD-CG code. The dose rates were also calculated at two additional points inside and at the end of the DSM cavity.

SHIELDING THICKNESSES: 1.3 cm STEEL (TOTAL) 
5/8 cm CONCRETE (ρ = 367 g/cm³)
DECAY TIME: 3.6 YEARS

FIGURE 8 DRY STORAGE MODULE SHOWING SHIELD THICKNESS, DOSE POINT LOCATIONS AND DOSE RATES (ALL POINTS AT MID-LENGTH OF DSM)

For the dose rate calculation purposes, a homogenized volume source region was specified for the bottom half of the DSM cavity. The pressure tube activities were taken from ORIGEN and the resulting gamma sources were grouped in six energy groups.

Dose Rate Comparison

The calculated dose rates (Figures 8 and 9) and their comparison with those of various measurements outside/inside the DSMs are given in Table 7. These are explained below:

<table>
<thead>
<tr>
<th>Dose Point (Fig. 8, 9)</th>
<th>Calculations (µSv/h)</th>
<th>Measurement (µSv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>97</td>
<td>NA</td>
</tr>
<tr>
<td>2</td>
<td>69</td>
<td>NA</td>
</tr>
<tr>
<td>3</td>
<td>91</td>
<td>NA</td>
</tr>
<tr>
<td>4</td>
<td>56</td>
<td>NA</td>
</tr>
<tr>
<td>5</td>
<td>17</td>
<td>NA</td>
</tr>
<tr>
<td>6</td>
<td>32</td>
<td>NA</td>
</tr>
<tr>
<td>7</td>
<td>21</td>
<td>NA</td>
</tr>
<tr>
<td>8</td>
<td>5</td>
<td>NA</td>
</tr>
<tr>
<td>9</td>
<td>17</td>
<td>NA</td>
</tr>
<tr>
<td>10</td>
<td>0.1</td>
<td>NA</td>
</tr>
<tr>
<td>11</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td>12</td>
<td>5</td>
<td>2</td>
</tr>
<tr>
<td>13</td>
<td>4</td>
<td>1.5</td>
</tr>
<tr>
<td>14</td>
<td>4.7 x 10⁴</td>
<td>NA</td>
</tr>
<tr>
<td>15</td>
<td>7.7 x 10⁴</td>
<td>7.5 x 10⁴</td>
</tr>
</tbody>
</table>

FIGURE 9 DRY STORAGE MODULE SHOWING DOSE POINT LOCATIONS AND DOSE RATES (DECAY TIME = 3.6 YEARS)
First, the comparison between the calculated dose rates at points 1 and 3 was made with those of measurements at two elevations along the DSM Glacier on April 5th, 1986 (i.e., 2.4 years after Unit 2 shutdown). These measurements were made at the mid-height (point 1) and at 0.9m up from the bottom (point 3) along the DSM. The measured dose rates were in the order of 70 μSv/h (max.). The corresponding calculated dose rates from Table 7, but with 2.4 years decay were 100 μSv/h and 110 μSv/h. The agreement is good.

Second, the comparison between the calculated dose rates at points 6, 7, 8, 9 and 10 was made with the three measurements outside the end wall of the DSM Prince Charming containing 87 pressure tubes on June 10th, 1987 (i.e., 3.6 years after Unit 2 shutdown). Points 6, 7 and 8 are on contact with, 0.6 m and 3.7 m from the end wall at 0.9 m up from the bottom of DSM. The calculated dose rates at these point were 32 μSv/h, 21 μSv/h and 5 μSv/h respectively. The three measurements were reported to be 1 μSv/h, 2 μSv/h and 1.5 μSv/h. However, it wasn't quite clear where the exact measurements were made. It was stated that higher dose rates could have been measured if the measurements were made at higher elevations. This is correct and is shown by the rise in the calculated dose rate at points 10, 9 and 6, i.e., on contact with the end wall at bottom, 0.6 m and 0.9 m up. The calculated dose rates were 0.1 μSv/h, 17 μSv/h and 32 μSv/h. To demonstrate the rise in the measured dose rate at 0.6 m away from the end wall (i.e., from 1 μSv/h on contact to 2 μSv/h), the dose rates were calculated at three more points outside the end wall, viz., points 11, 12 and 13 in Figure 9. These points are located on contact with, 0.6 m and 3.7 m away from the end wall, 0.3 m from the bottom of the DSM. The dose rates were 2 μSv/h, 6 μSv/h and 4 μSv/h. Thus, the trend in the measured dose rates was demonstrated.

The dose rates were also calculated at points 14 and 15 inside the DSM cavity. Point 14 is located at the end wall 0.9 m from the bottom of the DSM, and point 15 is located 0.9 m from the end wall on inspection port axis. The calculated dose rates were 4.6 x 10^8 μSv/h and 7.7 x 10^7 μSv/h. The dose rate at point 15 was measured to be 7.5 x 10^7 μSv/h for the DSM Scugog. The measured dose rates agree well with the calculated dose rates.

Comparison of the measured activities from the pressure tubes and dose rates from the end fitting and its pressure tube stub under water and the pressure tubes inside the DSM’s show good agreement with the calculations. The dose rates calculated outside the multibarrel flask show poor agreement with the calculated dose rates. We await the results from Pickering 3 and 4 in the hope we can reconcile our calculations with measurements.

Overall, the results give confidence that the methodology can be reliably applied to radiation dose rate assessment associated with LSFCR program for Units 1, 2 and 3, 4 of Pickering reactors or other CANDU nuclear power stations.

ACKNOWLEDGEMENT

We would like to thank Dr. V. Fidleris, Metallurgical Engineering Branch of CRNL for providing us with the measured activities for the pressure tube and dose rates from the end fittings.

REFERENCES

(3) RSIC Computer Code Collection, "DOT 3.5 Two Dimensional Discrete Ordinates Radiation Transport Code" 1975 November.
(6) KENNEDY, J.E., "Large Scale Fuel Channel Replacement (LSFCR) Radiological Hazards and Conditions During Reactor Disassembly" Safety Services Department, Ontario Hydro Report SSD-1R-87-13, 1987 December.
Session 5:

Public Information Needs for Waste Management

Chairman:

A.V. Manchee, Ontario Hydro
SESSION #5: Public Information Needs for Waste Management

June 6 p.m., Room - Provinces I

Chairman: Mr. A.V. Manchee - Ontario Hydro

This special session on Information Needs for Public Acceptance of Waste Management Programs recognizes that although waste management has become one of the most important issues of the late twentieth century, the design and building of effective storage/disposal facilities is not the main challenge facing the technical experts. No matter whether the waste is domestic garbage, toxic chemicals, or radioactive fission products in used nuclear fuel, public acceptance is critical if a disposal facility is to be built. What the public needs most in order to make an informed decision is INFORMATION.

Successful public affairs programs depend on proper positioning, packaging, and provision of the right information. This session looks at identifying and meeting this information need from the corporate, government, media, and public opinion industry perspectives.

Five Speakers (Panel Discussion):

14:00 Terry Young
"Public Communication"
Ontario Hydro Media Relations

14:20 Brian Finlay
Government Information Needs
Ontario Ministry of Energy

14:40 Derek Nelson
Media Perspectives
Queens Park Columnist
Thomson News Service

15:00 Margaret Buhlman
Public Attitudes Toward the Environment
Decima Research

15:20 Michael Scott
The Role of Public Consultation
Ontario Waste Management Corporation
Director of Communications

15:40 Open Discussion
Public communications is a function often misunderstood and/or disregarded by the technical community. Corporate media relations staff often encounter attitudes among technical people such as "no news is good news" particularly in the area of waste management, "we know what we’re doing and the field is far too complicated for the media and public to understand anyhow". This attitude has fostered media distrust of the nuclear industry. The public needs to understand the issue, and information must be in a form comprehensible to them. Management of nuclear waste is going to be a major issue in the near future. The industry should take the initiative and try and educate the public now while it can still influence the information flow. The CNS can play a significant role by making its members accessible and responsive to the media. The CNS should position itself as a credible alternative voice on nuclear waste management and other industry issues.
GOVERNMENT INFORMATION NEEDS

Brian Finlay
Ontario Ministry of Energy

The government has unique information needs relating to the nuclear industry. This presentation will look at these needs, and how successful the industry is in meeting them.
MEDIA PERSPECTIVES

Derek Nelson
Queens Park Columnist
Thomson News Service

There is no one media perspective on nuclear waste, but there are certain constants in media interests concerning the issue and not all deal just with information. These will be examined as well as some alternative possibilities as to how the media will handle on-going waste disposal stories.
PUBLIC ATTITUDES TOWARD THE ENVIRONMENT

Margaret Buhlman
Decima Research

In times when environmental issues are a top concern among the Canadian public, we can expect more focus on industry and government activities to alleviate these concerns. This presentation will address public attitudes toward the environment, with particular emphasis on perceptions of waste management issues. The emphasis will be to provide an indication of the extent to which people see waste management as an issue, how it should be solved, and what is in store for industry as they deal with the public and the problem.
Public consultation has become an essential part of any major development project, in either the public or private sector. The Ontario Waste Management Corporation has undertaken a major public consultation program, as part of the process it developed for the siting and construction of an industrial waste treatment facility. Mr. Scott will discuss the role public consultation has played in this project and the issues that need to be addressed if waste management projects are to involve the public in the future.
Session 6:

Thermalhydraulics I

Chairman:

V.S. Krishnan, AECL-WNRE
TUF VERIFICATION AGAINST SELECTED RD-14 EXPERIMENTS

by

J. PASCOE, A. TOMASONE, M. WILLIAMS, W. YOUSEF, W. LIU and J. LUXAT

Ontario Hydro, Toronto, Ontario

ABSTRACT

The advanced, two-fluid computer code TUF (Two Unequal Fluids) has been developed as part of a study of the effectiveness of Emergency Coolant Injection Systems for Ontario Hydro reactors in the event of a large Loss Of Coolant Accident (LOCA). Three experiments performed in the RD-14 thermal-hydraulics test facility at AECL-WNRE have been simulated as part of a co-ordinated verification program for the TUF code. The experiments were chosen to demonstrate both the capability of the code to capture the underlying two-fluid phenomena and the robustness and accuracy of the solution algorithms employed in the code.

The following experiments were simulated:

1) A partial inventory thermosyphoning test conducted at high heated section power and high secondary side pressure.

2) A critical inlet header break with rundown of the main coolant pumps.

3) A critical inlet header break without rundown of the main coolant pumps.

The paper presents an analysis of the experimental test results, concentrating on identifying the governing physical phenomena. The results obtained from the simulations of the tests using the TUF code are presented, together with a comparison to the experimental data and an evaluation of the performance of the code.

1.0 INTRODUCTION

In order to provide an advanced, state of the art tool for the use in the thermal-hydraulic analysis of nuclear generating stations, the TUF (Two Unequal Fluids) code has been developed in the Nuclear Safety Department at Ontario Hydro. TUF is a full six-equation code capable of modelling thermal non-equilibrium and non-homogeneity as well as piping elasticity/plasticity and non-condensables.

TUF simulations of three experiments performed on the RD-14 test facility are presented in this paper. The experiments included; a partial-inventory thermosyphoning test (T8512 and repeats T8513 and T8514), a critical inlet header break test with primary pump rundown (B8713) and a critical inlet header break without pump rundown (B8706).

The partial-inventory thermosyphoning test was performed at high primary and secondary side pressures to study the class of transients in which decay heat is removed from the core by two-phase natural circulation. Of particular interest is the point at which thermosyphoning breakdown, brought about by a loss of primary coolant, occurs.

The critical inlet header break experiments B8713 and B8706 were performed at CANDU typical nominal reactor operating conditions. An inlet header break is defined to be critical when a loss of inlet to outlet header pressure differential causes periods of sustained low flow in the heated section downstream of the broken header. These tests were performed to obtain experimental data relevant to blowdown behaviour with emergency coolant injection in a CANDU typical figure-of-eight loop. Of particular interest in critical breaks is the possibility of heater temperature excursions due to sustained low coolant flows in one pass induced by the break.

The simulations discussed here are part of an ongoing co-ordinated verification program for the TUF code.

A description of the RD-14 test facility and experimental procedure is given in the following sections. The TUF code is discussed briefly and the experimental and the TUF simulation results are discussed in more detail in subsequent sections.

2.0 RD-14 FACILITY DESCRIPTION

The RD-14 test facility, shown in Figure 1, is a scaled (heated sections, steam generators, pumps and headers are 1:1 vertical) representation of a typical CANDU reactor. It was designed to provide the capability to simulate behaviour in a figure-of-eight heat transport loop under a variety of conditions representative of those expected under plant upsets and accidents. The primary system (10 MPa nominal) consists of one loop with two 5 MW (nominal), 6 meter long, horizontal electrically heated sections representing 2 passes through a core. The loop contains two centrifugal pumps which provide representative head and flow characteristics and which provide flows of similar magnitude to the flow in individual channels in a reactor. A pressurizer complete with electric heaters provides primary loop pressure regulation.
The secondary system removes heat from the primary loop controlled via a jet condenser which condenses the steam and returns the condensate to the steam generator. At the time of thermosyphoning test (TS812) the steam generators were configured with simple perforated plate steam separators. These were found to be inadequate in providing the desired recirculation and were replaced by a cyclone type separator for the blowdown experiments.

In the partial-inventory thermosyphoning tests, primary coolant is drained from the inlet of heated section 2, passed through a condensing heat exchanger and stored in an inventory tank. By measuring the liquid level in this tank at 25°C and atmospheric pressure, the amount of mass drained can be determined.

The ECI system shown in Figure 1, consists of both a high and low pressure injection stage. High pressure injection is provided by a pressurized ECI tank (5.5 MPa(g)). Low pressure injection is provided by a low pressure ECI pump connected to an auxiliary distilled water tank.

The discharge, as a result of opening the break valve at header 4, is vented to atmosphere via a large vertical blowdown stack which protrudes from the building (see Figure 1).

### 3.0 CODE DESCRIPTION

TUF is a fully UVUT (Unequal Velocities and Unequal Temperatures) two-fluid code. TUF utilizes the flow-rate approach in the solution scheme and ensures mass and energy conservation, while maintaining computing efficiency and numerical stability.

While the code can be used in a full two fluid configuration, it also allows the user the following choices:

1. Homogeneous equilibrium model.
2. A four equation dynamic slip model (equal phasic temperatures)
3. A Drift flux model (algebraic velocity difference with one phase at saturation)
4. A five equation model with unequal phasic velocities and one phase at saturation.

TUF allows the user to specify at which locations of the input model geometry, that a particular model is to be used (for example in the simulations presented in this paper the primary side was modelled as UVUT and since the secondary side behaviour is of little or no interest, it was modelled as homogeneous equilibrium). The user also has a choice of central or non-central nodalization (the simulations presented here were non-central).

The TUF code has detailed models of heat conduction (fuel to fluid, fluid to piping, primary to secondary and piping to environment) and of all system components.

A more complete description of the TUF code is given in Reference 1.

### 4.0 TUF RD-14 NODALIZATION

The nodalization for the RD-14 facility is shown in Figure 2. Note that a full primary, secondary and ECI systems are included in the model. The nodalization consists of 188 nodes and 196 links which forms the complete model. Primary pumps were modelled using full four quadrant homologous pump curves. The value of inertia for the pumps was calibrated such that the rundown characteristic from the experiments were reproduced. Most of the vertical and horizontal piping sections were modelled as separate nodes, where possible, in order to provide a better geometric representation of the loop. A full secondary system including preheaters, external downcomers, steam separators and jet condenser are included. The jet condenser pressure and the feed-water flow were modelled as time varying boundary conditions. The ECI system model includes check valves and motorized ball valves. The low pressure ECI pump delivery characteristic was derived from the measured head/flow characteristic. The high pressure ECI injection mode was modelled as a time varying boundary condition applied to the ECI tank pressure. The ECI injection line head loss coefficients were adjusted to reproduce the pressure and flow distribution within the ECI piping network. Specifically, during sample steady state simulations, the header pressures were fixed (to the experimental values) while the loss coefficients were varied until the correct ECI flows were obtained. The steady state conditions of the loop, before the opening of the break valve, were calibrated so that experimental system pressures, heated section inlet and outlet temperatures, boiler inlet and outlet temperatures, heat loss to atmosphere, flows, and other system parameters were matched (see Tables 1 and 2). The heat transfer from the primary loop piping to the atmosphere was simulated taking into account piping wall temperatures, atmospheric temperature and the natural convection heat transfer coefficient.

The nodalization for the secondary side was done in such a way so as to separate individual components (riser, steam drum and so on) and to preserve orientation.

Modelling of the RD-14 steam generators for the thermosyphoning test proved to be difficult because of deficiencies with some experimental data. The downcomer flow was too low to measure and the top of the downcomer was approximately 5°C hotter than the bottom. The full power recirculation ratio, normally expected to be 5, was found to be less than unity. Clearly the desired mode of recirculation, two-phase into the steam separator, with liquid returning to the top, did not occur. The steam generators apparently operated
more like a kettle - two-phase moving up the centre and liquid falling near the walls. The net result is an effective reduction in the heat transfer area. The centre tubes will see a two-phase mixture while the outer tubes will see liquid only. The use of the correct heat transfer model becomes increasingly important. This mode of recirculation in the steam generator cannot be modelled properly with a one-dimensional code, without extremely detailed nodalization.

Also, in the thermosyphoning test the feed-water was found to vary periodically (Figure 3) within a range of approximately 60 C.

This variation in temperature was found to have a direct and significant effect on such loop parameters as boiler outlet temperature. Figure 4 shows the variation in boiler 2 outlet subcooling and the rate of change of feed-water temperature. To confirm the influence of feed-water temperature variability on boiler outlet subcooling, the feed-water temperature and boiler outlet subcooling were cross correlated and the result is shown in Figure 5. The Cross Correlation Coefficient (CCC) is a measure of how well two sets of numbers are related. A CCC of one means a change in one set coincides with a like change in the other, while a CCC of minus one means a change in one set causes an opposite change in the other. A zero CCC means no correlation. The x-axis in Figure 5 is a measure of the shift or lag, between one set of numbers and the other. For example the sine and cosine functions are 100% cross correlated with a lag of π/2. From Figure 5 it appears that there is no significant correlation between the boiler outlet subcooling and the feed-water temperature in experiment T8514. The results from T8512 indicate a relatively high level of agreement and from T8513 indicate moderate correlation. At a lag of zero, the subcooling, in experiments T8513 and T8514, is negatively correlated with the feed-water temperature. An increase in feed-water temperature increases the secondary temperature (albeit marginally) resulting in a decrease in subcooling. Experiment T8512 does not show a similar effect as the initial temperature was high and increased only a few degrees before peaking. The period from A to B in the feed-water variation (Figure 3) is one of increasing temperature. The corresponding region of the correlation curve is one of an increase from -.15 to +.1, indicating that, as the feed-water temperature increases the subcooling decreases. As the temperature of the feed-water decreases (from B to C in Figure 3) the CCC decreases to a value of -.13. Thus, as the feed-water temperature decreases the subcooling increases. The slight increase from C to D in Figure 3 is reflected in the stable period from C to D in Figure 5. The feed-water temperature decreases again from D to E with a subsequent decrease in the CCC to -.15 (implying an increase in the subcooling). At point E the cycle repeats.

The situation is further complicated by the fact that the TUF steam generator model has an implicit steam separator, which forces the liquid into the downcomer and steam into the steam drum.

Since the steady state downcomer and steam flows were essentially unknown in the thermosyphoning test, care was taken to obtain the correct primary side mass flow and heat balance and to keep the downcomer flow small by making the flow resistance high. Such a procedure was thought to be sufficient since the details of the secondary side transient is of little interest in thermosyphoning tests where the secondary side pressure and temperature are relatively constant.
The feed-water temperature variation, clearly has a direct effect on the primary side. The feed-water variation in the temperature was modelled in detail by imposing a functional fit to the experimental data (Figure 3), as a boundary condition.

To further facilitate modelling of the secondary side in the thermosyphoning test, an extra node was added upstream of the feed-water tee (node 185 in Figure 2).

Before the start of an experiment, instrumentation was zeroed and adjusted. Steady, single phase natural circulation was established at the desired power level and primary and secondary pressures. The pressurizer was then isolated and data gathering was initiated. The initial loop conditions are summarized in Table 1.

At approximately 25 seconds the first draining of roughly 2% of the initial loop volume of 951.46 litres, was carried out. After allowing the loop sufficient time (approximately 750 seconds) to stabilize, another draining of 2% of the initial loop volume was carried out (Figure 6). Three more like drainings at 750 second intervals were done. The final drains were of approximately 10% of the initial loop volume at roughly 1500 second intervals. The drainings were carried out until interruption of continuous thermosyphoning occurs. As a result of stratification in the heated sections, heater sheath temperatures in excess of 600°C were reached. These drainings were simulated in TUF by extracting liquid at a controlled flow-rate (Figure 6) from the inlet of heated section 2.

The first three drainings cause an increase in the volumetric flow-rate at the inlet of the heated section (Figure 7) that is seen in both the experimental and predicted values. During this phase void appears in the outlet of the heated sections (Figure 8) with a resultant increase in the density gradient between the inlet and outlet. The agreement between the TUF predictions and the experimental values is generally good up to about 2200 seconds. However, the TUF predicted pressure (Figure 9), heated section 2 outlet temperature (Figure 10) and heated section 2 outlet sheath temperature (Figure 11) are slightly higher than in the experiment. As pointed out in Table 1, the initial values for the TUF simulation were based on averages of the three experiments.

The inlet void at boiler 2 (Figure 12) and the boiler outlet plenum temperature (Figure 13) further illustrate this point. While the predicted and experimental outlet voids (Figure 12) follow the same trend, the predicted outlet plenum temperature lags behind the experimental values by several degrees. As well, the sudden drops induced by the drop in feed-water temperature do not show up in the predicted values. Such a finding is not surprising since the TUF steam generators are well mixed (high recirculation ratio) unlike the actual steam generators (which behaved as "kettles").

5.0 RESULTS

5.1 Partial-Inventory Thermosyphoning Test T8512

The thermosyphoning experiment simulated, T8512, had a secondary side pressure of 4.6 MPa and a power input of 160 kW (nominal). Also two repeats of this experiment, T8513 and T8514, were carried out and with the possible exception of the effect of feed-water temperature variation, the results of all three tests were similar. Of the three experiments, T8513 exhibited the earliest trip time (on fuel element simulator overheat), and was therefore chosen to be the test simulated.
After 2200 seconds the TUF heated section 2 volumetric flows (Figure 7) continue to increase with subsequent drainings up to 4800 seconds. The experimental values, however, begin to decrease due to increasing two-phase pressure drops.

FIGURE 7. VOLUMETRIC FLOW RATE AT TEST SECTION 2 INLET.

FIGURE 8. TEST SECTION 1 OUTLET VOID

FIGURE 9. PRESSURE AT HEADER 9

FIGURE 10. TEST SECTION 2 OUTLET FLUID TEMPERATURE

FIGURE 11. TEST SECTION 2 OUTLET SHEATH TEMPERATURE - TOP PIN

FIGURE 12. BOILER 2 INLET VOID

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The drain at 4800 seconds results in void at the boiler outlet (Figure 14), increased void at the heated section outlets (Figure 8) and saturation temperatures at the boiler outlets (Figure 4). The heated section volumetric flow rates (Figure 7) drop to the threshold of stratification, with a transient rise in the outlet sheath temperature (Figure 11).

The TUF predictions, while following this general trend do not capture all of the details. The heated section outlet voids (Figure 8) remain lower than the experimental values and consequently the flow-rates are too high.

The draining at 6100 seconds induces wide spread temperature excursions (Figure 10 and 11) brought upon by flow stratification (Figure 7) and void carry-over at the steam generators (Figures 4, 13 and 14). It is assumed that the threshold for stratification is 250 kg/m²s - approximately 1 l/s in the test section.

The decrease in boiler 2 inlet void (Figure 12), heated section 1 outlet void (Figure 8) and increase in subcooling at boiler 2 outlet (Figure 4) indicate liquid moving from the outlet side of the steam generator back to the heated section which leads to a decrease in the density driving force and flow reversal.

The TUF predictions lag behind and although the timing of the increase in the boiler 2 inlet void at 6300 seconds is captured, there is no subsequent increase in fluid (Figure 10) or sheath temperature (Figure 11). There does appear to be an indication of saturated fluid at the boiler outlet (Figure 13) and an increase in the void at the heuned section 1 outlet (Figure 8). It appears that the volumetric flow-rate which has been high because of the high density gradient resulting from the lower boiler outlet temperature, will certainly drop to the stratified level before or during the next draining.

The overall agreement between the TUF simulations and the experimental data is acceptable. The difficulties experienced in reproducing the experimental data relating to test section flows is due primarily to modelling uncertainty related to heat transfer to the steam generators. It is concluded that the TUF steam generator model, with its greater recirculation, tends to act as a slightly more effective heat sink. This in turn acts to maintain a large density gradient and higher thermosyphoning flow during the earlier part of the transient.

The drain at 4800 seconds results in void at the boiler outlet (Figure 14), increased void at the heated section outlets (Figure 8) and saturation temperatures at the boiler outlets (Figure 4). The heated section volumetric flow rates (Figure 7) drop to the threshold of stratification, with a transient rise in the outlet sheath temperature (Figure 11).

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**TABLE 1. Initial Loop Conditions and TUF Steady State Values**

<table>
<thead>
<tr>
<th>Partial-Inventory Thermosyphoning</th>
<th>RD-14*</th>
<th>TUF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power input (kW)</td>
<td>160</td>
<td>160</td>
</tr>
<tr>
<td>Header 3 Pressure (kPa)</td>
<td>6595.7</td>
<td>6595.7</td>
</tr>
<tr>
<td>Steam Drum Pressure** (kPa)</td>
<td>4615.6</td>
<td>4615.6</td>
</tr>
<tr>
<td>Boiler 1 In Plenum Temp.*** (C)</td>
<td>278.46</td>
<td>278.46</td>
</tr>
<tr>
<td>Boiler 1 Out Plenum Temp.***(C)</td>
<td>257.49</td>
<td>250.17</td>
</tr>
<tr>
<td>Initial Feed-water Temp.**** (C)</td>
<td>26.82</td>
<td>27.27</td>
</tr>
<tr>
<td>Heated Section 2 Inlet Flow (l/s)</td>
<td>1.27</td>
<td>1.29</td>
</tr>
<tr>
<td>Heated Section 2 Inlet Temp. (C)</td>
<td>249.33</td>
<td>249.35</td>
</tr>
<tr>
<td>Heated Section 2 Outlet Temp. (C)</td>
<td>279.83</td>
<td>279.54</td>
</tr>
<tr>
<td>Boiler 2 Inlet Plenum Temp. (C)</td>
<td>278.46</td>
<td>278.64</td>
</tr>
<tr>
<td>Boiler 1 Downcomer Flow (kg/s)</td>
<td>N/A**</td>
<td>1.32</td>
</tr>
<tr>
<td>Boiler 1 Steam Flow (kg/s)</td>
<td>N/A**</td>
<td>0.054</td>
</tr>
</tbody>
</table>

* Average of T8512, T8513 and T8514 values.
** Flow too low to measure.
*** Average of Boiler 1 and 2.
**** Value from T8513.

5.2 Critical Inlet Header Break Tests B8713 & B8706

The RD-14 facility was brought up to reactor-typical full power conditions, summarized in Table 2.

**TABLE 2. Initial Experimental Conditions**

<table>
<thead>
<tr>
<th>Primary System: Power per Heated Section</th>
<th>5 MW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Outlet Header Pressure</td>
<td>10.1 MPa</td>
</tr>
<tr>
<td>Mass Flow Rate</td>
<td>21 Kg/sec</td>
</tr>
<tr>
<td>Outlet Temperature of the Heated Sections</td>
<td>310 °C</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Secondary System: Steam Drum Pressure</th>
<th>4.6 MPa</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feed-water Temperature</td>
<td>187 °C</td>
</tr>
<tr>
<td>Steam Drum Level</td>
<td>50.0 %</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>ECI System: High Pressure ECI Tank</th>
<th>5.6 MPa</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low Pressure ECI Pump 8 Discharge Pressure</td>
<td>1.6 MPa</td>
</tr>
</tbody>
</table>

---

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The system was allowed to achieve steady state conditions before the transient was initiated. Table 3 illustrates the sequence of events for both critical break experiments.

<table>
<thead>
<tr>
<th>TABLE 3 Sequence Of Events</th>
</tr>
</thead>
<tbody>
<tr>
<td>EVENT</td>
</tr>
<tr>
<td>-------</td>
</tr>
<tr>
<td>Data Recording Started.....</td>
</tr>
<tr>
<td>Surge Tank Isolated........</td>
</tr>
<tr>
<td>Blowdown Valve Opened......</td>
</tr>
<tr>
<td>Channel Power Reduced</td>
</tr>
<tr>
<td>Outlet Header Pressure</td>
</tr>
<tr>
<td>Reached 6.1 MPa(abs) ECI</td>
</tr>
<tr>
<td>Injection Valves open......</td>
</tr>
<tr>
<td>High Pressure ECI Began....</td>
</tr>
<tr>
<td>Primary Pumps Tripped to</td>
</tr>
<tr>
<td>Over-voltage................</td>
</tr>
<tr>
<td>HP ECI Tank Reached low-level Set-point. HP ECI Stopped, Low Pressure ECI Started</td>
</tr>
<tr>
<td>Primary Pumps Shutdown.....</td>
</tr>
<tr>
<td>Data Recording Stopped.....</td>
</tr>
</tbody>
</table>

The break is initiated by a fast acting 6" ball valve, downstream of a 30.0 mm (5% header area) orifice connected to inlet header 4. The experiments with this break size were selected because they produced the longest periods of sustained low flow in heated section 1. The discharge is vented to atmosphere via a large vertical blowdown stack shown in Figure 1. The primary pumps in experiment B8713 were forced to follow a controlled exponential rampdown representing a typical CANDU primary pump trip with a free-wheeling rundown. The primary pumps in experiment B8706 were intended to continue running throughout the transient, however, after the pump suctions voided, they tripped due to over-voltage at 22 seconds.

Following the isolation of the surge tank, the blowdown valve at inlet header 4 is opened 4 seconds later. This event initiates a degradation of the positive pressure differential across heated section 1. Figures 15 and 16 show the TUF calculated and the experimental differential pressures for heated section 1 in tests B8713 and B8706, respectively. The agreement is found to be good, both the sustained low pressure differential (between 0 and 22 seconds) and the large negative pressure differential (between 23 and 50 seconds) are well predicted. The resultant reduction in the flow of heated section 1 is sustained for about 15 seconds. The fuel element simulator power was dropped to 200 kW per heated section at 12 seconds in both tests simulating reactor shutdown and decay power levels.

In test B8713 the primary pumps are ramped down starting at 12 seconds, while in test B8706 they are not. At about 17.5 seconds the pressure in the outlet headers dropped below 6.1 MPa at which time the ECI injection ball valves were opened allowing a path for ECI flow should the pressure drop further (i.e. to the ECI tank pressure of 5.6 MPa). During this time the fuel element simulators at the inlet to heated section 1 experience a temperature excursion in which the top element sheath temperatures experience a larger increase than the bottom element temperatures due to stratification effects in heated section 1. Figure 17 shows the top element predicted, experimental and saturation temperatures for test B8713 (shown in Figure 18 for test B8706). Although the inlet of heated section 1 experiences a longer period of sustained low flow, the fuel element simulator sheath temperatures at the outlet of heated section 1 experience higher temperatures. Agreement between the TUF calculated and the experimental sheath temperatures is good. The onset of the element sheath temperature excursion and cool-down is well predicted, as well as the maximum sheath temperature.
After significant voiding in the primary pump suction of test B8706, the pumps tripped due to over voltage (contrary to the original intent). As a result of this significant voiding, the head for both pumps drops off very quickly and hence little difference in the general trends of these blowdown experiments is observed. The pressure transient for header 4 is shown in Figure 19 for test B8713 and Figure 21 for test B8706. The pressure in header 1 is shown in Figure 20 for test B8713 and Figure 22 for test B8706. These are the two headers for the broken pass. The comparison between the TUF prediction and the experiment is seen to be good up to 45 seconds. In order to predict the temperature excursions well, the void transient must be correct. Between 45 and 80 seconds TUF predicts slightly lower pressures than observed. Heated section 1 inlet void is shown in Figure 23 for test B8713 and Figure 24 for test B8706. The agreement is found to be quite good especially with regard to timing of onset of voiding and refill at that location. The ECI injection flows are shown in Figure 25 for test B8713 and Figure 26 test B8706. Only header 1 ECI flow is shown since the experimental measurement for the flow in header 4 was in excess of the maximum range of the instrument. The agreement between the TUF prediction and the experimental results is good. The ECI tank reached its low level set-point and low pressure injection initiated at about 87 seconds. Good agreement between TUF and the experimental data is obtained at the later times of the transient.
6.0 CONCLUSIONS

The Advanced thermal-hydraulic code TUF has been described and its features summarized. The TUF nodalization of the RD-14 test facility was presented along with the details of the modelling procedures.

TUF was used to simulate the following experiments:

1) Partial-inventory thermosyphoning test T8512.
2) Critical inlet header break with primary pump rundown test B8713.
3) Critical inlet header break without primary pump rundown test B8706.

It was found that:

- The overall agreement between the TUF predictions and the RD-14 partial-inventory thermosyphoning test T8512 was acceptable. The major disagreement, the magnitude of the flow reduction following draining, it is thought, can be rectified by accurate modelling of the steam generator flow dynamics. Particularly in tests, such as T8512, where the steam generator was run in a off normal mode with severely reduced recirculation.

- Predictions obtained using the TUF thermal-hydraulic code were compared to experimental data from RD-14 critical inlet header break tests B8713 and B8706. The agreement with the data was found to be quite good with only a slight under prediction of pressure during the refill phase of ECI injection. Little difference between the test with primary pump rundown (B8713) and without (B8706) was observed in both the experimental results and the simulations.

7.0 ACKNOWLEDGEMENTS

The experiments presented in this paper were funded by the CANDU Owners Group (COG).

8.0 REFERENCES

VERIFICATION OF CHAN-II(MOD 6) AGAINST EXPERIMENTS

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ABSTRACT

The CHAN-II series of codes is used in the licensing analysis of CANDU reactors. The basic code provides a fast calculational tool for the prediction of fuel channel heatup during a postulated loss-of-coolant accident (LOCA) with loss of emergency coolant injection (LOECI). It is the objective of this paper to discuss verification of the CHAN-II(MOD 6) computer code against two seven-element experiments performed at WNRK under COG/CANDKV, designed to represent a multi-rod geometry. The main models and assumptions used in the code will be briefly described.

The objective of the experiments is to produce data for code verification under integrated conditions with significant hydrogen production and flow rates similar to the LOCA/LOECI scenario. Agreement between prediction by CHAN-II(MOD 6) and experimental results is reasonably good.

INTRODUCTION

The CHAN-II series of codes is used in the licensing analysis of CANDU reactors. The basic code provides a fast calculational tool for the prediction of fuel channel heatup during a postulated large break loss-of-coolant accident (LOCA) with and without a loss of emergency coolant injection (LOECI) in which the transport of heat by convection is reduced. The code models the progression of postulated LOCA and LOECI events including fuel channel geometry changes due to severe overheating.

The CHAN-II code is based on an approach which replaces the complex pin geometry by an equivalent cylindrical ring geometry [1] (see Figure 1). The fuel channel is divided into several axial segments. Each of these segments is further subdivided into fuel rings, flow ring subchannels and the pressure tube and calandria tube. Fuel elements laying on the same pitch circle radius are represented by a fuel ring, subdivided into two annular masses, one inner and one outer, for the purpose of carrying out the thermal-hydraulic and heat transfer analyses for each flow ring subchannel.

The inlet flows to the subchannel rings (i.e. flow split) may either be input or calculated by the code. No lateral mixing of the subchannel flows is assumed to take place within a bundle, however, various mixing assumptions at the bundle end plates are available. Usually, subchannel flows are assumed to be constant during the transient except when there is a change in the bundle or pressure tube geometry. Various models which focus on determining the physical, thermal and chemical state of the fuel channel during an accident scenario are used to formulate a set of ordinary differential equations governing the temperature distribution. Interactions of the various thermal-hydraulic, thermomechanical and chemical models are also accounted for. The FORSIM code package [2] is used to solve the resulting set of ordinary differential equations explicitly to obtain the temperature distribution.

The Ontario Hydro version of the CHAN computer code has undergone extensive modification, improving and adding new models, resulting in CHAN-II(MOD 6) for single phase steam flow. (The latest licensing version, CHAN-II(MOD 7), includes two phase cooling). Other versions of CHAN have been used to compare with the CHAN verification experimental series [3]. It is the objective of this paper to show results of CHAN-II(MOD 6) for two seven-element, steam cooling experiments performed at WNRK under COG/CANDKV.

DESCRIPTION OF EXPERIMENTAL APPARATUS AND INSTRUMENTATION

The test apparatus is shown in Figure 2. Steam, supplied by the steam generator, is regulated via a 1-inch gate valve and monitored by an orifice flow meter. The steam leaving the test section is condensed and any hydrogen produced is collected in a hydrogen tank. The rate and total volume of hydrogen produced is monitored.

The test section consists of a centre heater (fuel element simulator) surrounded by a ring of six heaters with a pitch circle diameter of 38 mm. This geometry simulates the centre pin and inner ring of pins in a 37-element CANDU fuel bundle. Figure 3 shows a schematic of the seven-element test section. Each fuel element simulator consists of a graphite rod heater, a Zircaloy Pickering-type fuel sheath, and annular alumina pellets to electrically insulate the graphite rod from the sheath.

Figure 1: Schematic Drawing of Fuel Bundle Representation

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A zirconium flow tube, representing the pressure tube, surrounds the fuel element simulators. The flow tube is separated by a CO2 gap from an aluminum cooling jacket in which water is circulated, representing the moderator. The cooling jacket is an annulus, the inner wall representing the calandria tube.

For instrumentation purposes, the test section is subdivided into 12 equal axial nodes. Twenty-four platinum/13% rhodium (Type R) thermocouples were spot welded to the inner sheath surface of the centre and surrounding six heaters at different axial and azimuthal locations to monitor the fuel element simulator temperatures. Twelve thermocouples were spot welded on the outer surface of the flow tube to monitor the axial and circumferential temperature distribution. Four shielded Chromel-Alumel (Type K) thermocouples were used to monitor the steam inlet and outlet temperatures. Duplicate thermocouples were used at each location to increase reliability. Two thermocouples (Type K) were used to monitor the cooling water inlet and outlet temperatures. Thermocouple locations are shown in Figure 3. The test apparatus is described in more detail in [4].

**EXPERIMENT PROCEDURE AND RESULTS**

The general test procedure for both experiments is identical [4]. First, the test section is filled with argon and heated to approximately 300°C to avoid steam condensation. Then steam is introduced and the argon is purged. The steam flow rate is measured with an orifice meter. A predetermined power curve is then followed. The power is raised in steps to achieve steady state as well as various temperature rates of change on the sheath. The steam leaving the test section is condensed and any hydrogen produced is collected in a hydrogen tank. Heater failure defines the end of the experiment.

In the first seven-element experiment (Test 1), steam was introduced into the test section after the initial heat-up period. The power history curve is shown in Figure 4. Steam was introduced at time zero. As power was increased, the fuel sheath and pressure tube temperatures slowly started to increase. The power was slightly reduced to 10 kW at 300 s to obtain steady state conditions in the test section, and was maintained at this level for 700 s until the system reached equilibrium. The power was then raised to 43 kW and maintained at this level for 200 s until termination of the test due to overheating of the steam outlet pipe. The power was switched off at 1212 s and steam flow discontinued at 1289 s.

The measured steam mass flow rate was nearly constant at 11.8 g/s. The steam pressure in the test section was 428.4 kPa. The maximum centre pin sheath temperature recorded at axial segment 11 was 1213°C. The temperature difference between sheath temperatures corresponding to axial segment 2 and 11 increases from about 200°C at the start of the test to a maximum of about 500°C at the peak temperatures. The maximum ring pin and pressure tube temperatures were approximately 1161°C and 1075°C respectively. Hydrogen was not collected for this experiment.
In the second seven-element experiment (Test 2), the steam flow was 5.7 g/s. The steam inlet temperature was 242°C and the test section steam pressure was 203.4 kPa. The power history curve for the test is shown in Figure 5. Time zero in the Figure corresponds to 1500 s from the start of the experiment. The power was raised in steps from 10 kW at time zero to 40 kW at 900 s. At 900 s, the centre pin thermocouple readings became erratic and the power was reduced to 11 kW. It was maintained at that level for 50 s and then raised to 34 kW for 100 s.

The code was modified to accommodate the seven-element heater geometry including the material properties of the annular alumina pellets and graphite rod. Light water steam properties were added to the code.

The following assumptions are used in the simulation of the seven-element tests:

1. The seven heater elements are modelled by one centre pin and one ring representing the outer six elements.
2. No flow mixing occurs in the channel between the inner and outer flow subchannels. The flow split is input and assumed to be proportional to the subchannel areas.
3. The thermal capacitance of the graphite rod, alumina pellets and sheath were lumped together into the fuel. The thermal resistance between the graphite rod and sheath is due mainly to the alumina pellet and the air trapped in the alumina.
4. Thermodynamic and transport properties are for light water.
5. The convective heat transfer coefficient is obtained using the Hadaller Banerjee correlation [6]. The correlation assumes a Nusselt number of 2 in the laminar flow regime. A value of 6 was also used for comparison. The enhancement of heat transfer due to disruption at the entrance of the developing hydrodynamic and thermal boundary layer (thermal entrance effect) [8] is also accounted for, for comparison.
6. The emissivities of the fuel sheath, the pressure tube inner and outer surfaces were assumed constant and equal to 0.8. The emissivity of the inner surface of the calandria tube was set to 0.2.
7. The ring radiation heat transfer model is used.
8. Axial heat conduction and radiation is neglected.
9. The thermodynamic properties at each axial segment are evaluated at the segment centre temperature.
10. The temperature of the cooling water jacket is assumed to be constant at the measured average between inlet and outlet.
11. The Urbanic Heidrick equation [5] is used to calculate the Zircaloy-steam reaction rate.
12. A flat axial power profile was assumed in the simulations due to the relatively flat variation of the graphite rod resistance with temperature.
13. The inlet steam temperature was 240°C for Test 1 and 225°C for Test 2.

All centre pin thermocouples in this test failed prematurely. A post-test examination of the test section showed that many centre pin thermocouples were no longer attached to the sheath. The failure is attributed to the repeated heating and cooling of the test section. Most of the ring pin thermocouples on the surface facing the hot centre pin also failed.

The maximum centre pin sheath temperature recorded was 1343°C. The minimum axial temperature difference between the ring pin temperature at axial segments 2 and 11 was 200°C; the maximum was 500°C. The maximum ring pin and pressure tube temperatures were approximately 1276°C and 1330°C respectively. The cumulative hydrogen production was 3.3 g-mole.

**Comparison between Chan-It Predictions and the Experimental Results**

**CHAN-It Modifications and Assumptions**

The Chan-It computer code accepts the initial axial and radial temperature distribution in the channel as input and calculates the channel transient response. The code accepts CANDU fuel bundle geometry, i.e., 37- or 28-element. The normal computational grid consists of 13 axial subdivisions (one per bundle), three radial fuel rings and a centre pin for the 37-element bundle, and 12 axial subdivisions and three radial fuel rings for the 28-element bundle.
14. The power was set equal to the measured voltage times the measured current squared. Power losses were assumed negligible.

15. The moderator temperature was 13°C for Test 1 and 12°C for Test 2.

Comparison With Experiments

Five parameters are chosen to show the comparison between the CHAN-II predictions and experimental results, namely, the central pin sheath temperature, the outer facing ring sheath temperature, pressure tube temperature, outlet steam temperature and hydrogen production.

Test 1

Figures 6 and 7 show the comparison between prediction and experiment for Test 1 using a Nusselt number of 2 for the laminar region in the Hadaller-Banerjee correlation and not accounting for the thermal entry effect. There is a prolonged period of steady state from 500 s to 1000 s followed by a transient increase in temperatures between 1000 s and 1500 s.

Figure 6 shows the steam temperature at the outlet of the test section (average of the inner and outer subchannels for the predicted and average of the two thermocouples for the measured). At steady state (around 1000 s), the temperature is underpredicted by about 40°C which is within experimental error considering the spread in thermocouple readings. However, the temperature is underpredicted by about 100°C at the peak (1300 s). During the relatively low-temperature steady state region, the outlet steam temperature is primarily a function of the channel power and steam flow and independent of the convective heat transfer coefficient between the sheath and the steam. It can therefore be concluded that the assumed power and steam flow are reasonable for this test.

The underprediction of the outlet steam temperature during the transient, however, indicates that heat transfer from the sheath to the steam is being underpredicted. This is born out in Figure 7 which shows that fuel sheath temperatures are overpredicted, during both steady state and transient, at the front end of the test section. At the outlet of the test section, the fuel sheath temperatures are in excellent agreement during the steady state but overpredicted during the transient period.

Examination of the Reynolds number along the channel reveals that during steady state, the steam flow is turbulent at the upstream end of the test section but changes to laminar as steam temperature increases towards the downstream end. This is due to the increase in steam viscosity and decrease in steam density as the steam temperature increases. As overall temperatures increase throughout the transient, the whole channel becomes dominated by laminar flow. In the laminar regime, the Hadaller-Banerjee correlation assumes a Nusselt number of 2 based on flow through tubes. A more appropriate correlation based on flow through annuli gives a Nusselt number of 6 for this geometry. The simulation was repeated with this value and results are shown in Figures 8 and 9.

Figure 8 shows the steam temperature at the outlet of the test section (average of the inner and outer subchannels for the predicted and average of the two thermocouples for the measured). There is little effect on the outlet steam temperature (Figure 8) during steady state, however, during the transient the agreement is much better. The predicted sheath temperature in the outer ring (Figure 9) is much closer to experiment at the downstream end of the test section. Although the agreement is better, heat transfer from the sheath to steam is still underpredicted at the upstream end.
The thermal entrance effect has little effect on outlet steam temperatures (Figure 10). Sheath temperatures near the inlet to the test section (Figures 11 and 12) are still slightly overpredicted indicating that enhancement due to the thermal entrance effect could be greater than predicted by the model. Downstream sheath temperatures are well-predicted. The pressure tube temperatures (Figure 13) are also slightly overpredicted at the upstream end, however, the agreement is good downstream. In general, the agreement between prediction and experiment is excellent for Test 1.

In laminar flow, a hydrodynamic, thermal boundary layer develops next to wetted surfaces. A certain distance, referred to as the thermal entry length, is required downstream from a region of turbulence, such as the inlet of the test section, for this boundary layer to become established. The heat transfer coefficients are enhanced in this region over what would be expected for fully developed laminar flow [8].

A model to calculate the enhancement of heat transfer coefficients due to this thermal entry effect was incorporated into CHAN-II specifically for application to these tests. Enhancement factors close to 2 were obtained for the first few axial segments. Results including both the correction to the laminar Nusselt number and the thermal entrance effect are shown in Figures 10 to 13.
The same steps of simulation were repeated for Test 2. Figures 14 and 15 show results assuming a Nusselt number of 2 in the laminar regime and no thermal entrance effect. There is a brief period of steady state around 450 s followed by the transient. Examination of the Reynolds number indicates that the flow regime is laminar for the entire experiment.

The outlet steam temperature (Figure 14) is again underpredicted in the steady state region (although within experimental error) but substantially overpredicted for the transient, which is opposite to Test 1 results. Sheath temperatures (Figure 15) are overpredicted both in steady state and transient, but the discrepancy is substantially greater than in Test 1.

Results assuming a Nusselt number of 6 and accounting for the thermal entrance effect are shown in Figures 16 and 17. Both of these effects should apply to Test 2 as well as Test 1. The predicted sheath temperatures (Figure 17) move in the right direction, although still high, however, the outlet steam temperatures (Figure 16) are substantially overpredicted, now for both steady state and transient periods. It is apparent from these results, in particular, considering the effect of Nusselt number and the steady state steam temperatures in the second case, that the assumed steam flow is too low for this experiment. The uncertainty in the flow
measurement technique can be quite high for these low flows. It was therefore decided to increase the flow to match the outlet steam temperature during steady state. An increase of 40 percent was required. Results are shown in Figures 18 to 21 for an increased flow of 7.95 g/s. The outlet steam temperature in the transient period is still somewhat overpredicted, however, results for the sheath and pressure tube temperatures are very good. Test 2 was plagued with thermocouple problems and some of the measurements look somewhat erratic.

![Outlet Steam Temperatures for Test 2](image1)

![Centre Pin Sheath Temperatures](image2)

![Outer Ring Sheath Temperatures](image3)
Figure 22 shows the predicted and measured cumulative hydrogen production for Test 2. Predictions for the three sets of assumptions are shown. The hydrogen production rate is very sensitive to sheath temperature. The case which gives the best agreement for sheath temperatures (Nusselt number of 6, thermal entrance modelled, flow increased by 40 percent) also gives the best agreement for hydrogen production. This indicates that the Zircaloy-steam reaction model (Urbanic-Heidrick) used in the CHAN code is valid.
CONCLUSIONS

1. The CHAN-II computer code was used to model two seven-element CHAN verification experiments. The effects of assumed Nusselt number in the laminar flow regime, enhancement of the convective heat transfer at the upstream end of the test section and uncertainty in the measured flow were studied.

2. In general, excellent agreement was obtained for Test 1 (11.8 g/s), however, for Test 2 (5.7 g/s) the flow had to be increased by 40 percent from the measured value. A significant uncertainty in the flow measurement and thermocouple readings is suspected for Test 2. With the flow increased, the agreement was very good.

3. Both tests were dominated by laminar flow. The Hadaller-Banerjee correlation for convective heat transfer to steam assumes a constant Nusselt number for laminar flow, set equal to 2, based on experiments on flow in tubes. Using the more appropriate value of 6 for this geometry, based on flow in annuli, gives much better results. It appears that convective heat transfer in the laminar regime could be somewhat greater than currently modelled in CHAN-II.

4. The experiments show evidence for a thermal entrance effect where convective heat transfer near the inlet of the test section is enhanced over a distance required to establish the hydrodynamic, thermal boundary layer under laminar flow conditions. Other appendages that disturb the flow (e.g., bundle end plates) could disturb this boundary layer and enhance the heat transfer. This effect is currently not modelled in the licensing version of CHAN-II.

5. The Zircaloy-steam reaction model appears to be valid considering the good agreement between calculated and measured sheath temperature and hydrogen production for Test 2. The conduction, radiation and convection models appear to be adequate for the conditions experienced in the tests.

REFERENCES


ABSTRACT

A numerical code called FIDAS (film dryout analysis code in subchannels) has been developed with the main objective of simulating dryout and post-dryout heat transfer in rod bundles. This code features three-fluid representation of two-phase flow, in which the third fluid is liquid droplets entrained in the vapor phase. Since the code is able to predict the dryout occurrence by the film dryout criterion, no empirical CHF correlation is required. FIDAS gave good predictions of the onset of dryout when compared with experimental results from full-scale dryout tests on a 36-rod bundle. Furthermore, FIDAS gave better accuracy in predicting a wide range of thermal-hydraulic conditions when compared with other subchannel codes.

INTRODUCTION

FIDAS has been developed at the Power Reactor and Nuclear Fuel Development Corporation (PNC), mainly to predict onset of dryout and post-dryout heat transfer in rod bundles.[1] Validation studies of the developed code and models have been performed against experiments conducted under the conditions covering water-cooled reactor operation.[1][2][3] However, sufficient validation has not yet been performed for dryout in rod bundle geometry.

An especially challenging problem for codes like FIDAS has been the prediction of flow quality in subchannels and the prediction of dryout power of rod bundles. In the validation process, it is also useful to compare predictions with those generated by similar codes. Hence, as a first step of validation for rod bundle geometry, examinations were carried out on subchannel quality, by comparison with experimental data and with predictions by similar codes. Moreover, as a second step, the prediction capability is examined for dryout power of a 36-rod bundle to be used in the Japanese pressure tube type heavy water reactor; comparison will be made with full-scale dryout experiments in which dryout power has been measured in pressure tubes with varying inside diameter.

THE FIDAS FORMULATION AND CALCULATIONAL FEATURE

FIDAS provides a three-fluid, three-field representation of two-phase flow in rod bundles. The three fields are specified by continuous liquid film, continuous vapor and entrained liquid droplets suspended in vapor as shown in Fig.1(a). Thus a set of 12 basic field equations is built into the code, namely three continuity, three energy, six momentum equations. The 12 field equations together with the volume fraction conservation relation among three fields enable us to obtain analytical information for the following 13 parameters: three axial and three lateral velocities, three volume fractions, three specific enthalpies, and pressure. In order to predict dryout occurrence and post-dryout temperature behavior of the heated surface, the code provides the capability to simulate flow regime evolution in subchannels; i.e. single-phase liquid flow at the entrance, bubbly-slug flow formulation by evaporation, transition to a liquid-vapor-droplet annular flow, a further transition to vapor-droplet mist flow due to liquid film disappearance on the heated surface, and superheated single-phase vapor flow formulation due to complete evaporation of all droplets. In this evolution process, the onset of dryout is defined as the disappearance of liquid film adhering to the heated surface as shown in Fig.1(b). The liquid film flow rate is determined by mass transfers of droplet deposition, entrainment and vaporization from the liquid film.[1],[2]
As the inter-subchannel mixing model between adjacent subchannels, three types of mixing model are assumed: diversion cross flow, turbulent mixing and void drift. Diversion cross flow is caused by radial pressure gradients. The diversion cross flow can be obtained from the force balance of the lateral pressure gradient and the lateral friction force at the wall-fluid and fluid-fluid interfaces. Turbulent mixing results from the natural eddy transport between subchannels. Turbulent mixing is based on Prandtl's mixing length theory. Void drift is a phenomenon in which vapor tends to diffuse toward unobstructed regions. It was originally introduced by Lahey[4]. In FIDAS, these three types of mixing have been considered.

The conservation equations, as well as the constitutive relations for various models, are described in detail in Ref.[1]

CODE VALIDATION I: Prediction of Flow Quality in Subchannels

Experiments and Calculational Procedure

Two sets of experiments were compared with the FIDAS code predictions: the General Electric (GE) 9-rood bundle measurements by Lahey et al.[4], and the ISPRA 16-rood bundle measurements by Herkenrath and Hufschmidt [5]. In both sets, flow quality was measured at the exits of various subchannels of a square lattice bundle, in which electrically heated rods simulated nuclear fuel rods. Measurements were made using channel splitters and an isokinetic probe technique. The flow conditions were those of a normally operating boiling water reactor, in which the coolant reaches bubbly and slug flow. The main differences between the GE and ISPRA test sets were the bundle size (9 and 16 rods respectively) and the bundle length (1.80 and 3.66 m). The cross-sections for the test bundles are shown in Fig.2, as well as the subchannel configurations used in the code analysis.

The conditions of pressure, total mass flux, power, and inlet enthalpy under which the test were performed are listed in Table 1.

Results and Discussions

The flow quality for the corner subchannel is the most difficult to predict accurately. Since the corner has the smallest size and the highest power density, it is most sensitive to flow conditions. Consistent experimental measurements were difficult for the same reason; corner data for both GE and ISPRA test sets are characterized by large scatter, as shown in Fig.3 and 4.

THERMIT-II and COBRA-IV were compared here with FIDAS in the prediction of flow quality in subchannels. COBRA-IV [6] assumes a homogeneous vapor/liquid mixture. THERMIT-II [7] is similar to FIDAS, but it is a two-fluid code which uses empirical correlations for dryout, and does not simulate entrained droplets as a third fluid.

As shown in Fig.3, FIDAS and THERMIT trace well the experimental data of corner subchannel quality. For the corner of the 9-rood bundle, COBRA fails to represent the quality accurately. This is due to the absence of void drift modeling in which turbulent

Table 1: Experimental conditions for GE and ISPRA tests

<table>
<thead>
<tr>
<th>Condition</th>
<th>GE 9-rood bundle (uniform heating)</th>
<th>ISPRA 16-rood bundle (uniform heating)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure (MPa)</td>
<td>7.0</td>
<td>7.0</td>
</tr>
<tr>
<td>Mass Flux (kg/m²s)</td>
<td>725</td>
<td>1000</td>
</tr>
<tr>
<td>ROD POWER (kW)</td>
<td>1450</td>
<td>1500</td>
</tr>
<tr>
<td>Subcooling (kJ/kg)</td>
<td>530 - 1600</td>
<td>320 - 1600</td>
</tr>
<tr>
<td>AVERAGE EXIT QUALITY (%)</td>
<td>7.0 - 600</td>
<td>40 - 110</td>
</tr>
<tr>
<td>QUALITY (%)</td>
<td>2.9 - 31.8</td>
<td>2.0 - 31.0</td>
</tr>
</tbody>
</table>

Fig. 2 Test sections for uniformly heated rod bundles
mixing dominates completely and the subchannel quality approaches average bundle quality. For the 16-rod bundle (Fig. 4), FIDAS also agree well in predicting flow quality. In general, FIDAS gives better predictions for subchannel quality comparing with other codes under normal reactor operating conditions. It is considered that this improvement depends on suitable value of mixing length for each phase and void drift parameters. Indeed, the fact that FIDAS considers mass flux droplets as a third fluid, and offers a mechanistic (rather than empirical) dryout model, may make it a better code for simulations of dryout and post-dryout heat transfer in rod bundles.

CODE VALIDATION II: Prediction of Dryout Power

Experimental Apparatus

Over ten thousand points of full scale dryout data for several kinds of ATR (a pressure tube type HWR developed by PNC in Japan) fuel bundles have been accumulated by using the 14MW Heat Transfer Loop (HTL) in PNC over the last decade.[8][9][10]

As shown in Fig. 5, HTL consists of a test section housing a heater-rod bundle, an electric power supply system connected with the bundle, a steam drum, a high pressure condenser, a subcooler, a pressurizer, circulating pumps, etc. The maximum operating condition of HTL is 10MPa of pressure, 310°C of temperature, and 22.2 kg/s of flow rate.

The objective of this series of experiments was to investigate the effect of pressure tube inner diameter on dryout power. A 36-rod test bundle was used for this series of experiments. And the bundle was settled in the pressure tube vertically. In the 36-rod test bundle, heater rods are arranged in three concentric circular layers surrounding a non-heated center-rod, with ring type spacers which have the same shape as those used in an actual bundle. The rod diameter is 14.5mm, and the average pitch-to-diameter ratio is 1.17. The bundle has a heated length of 3.7m and heat transfer area of 6.06m². Electric DC current was supplied to the heater rods. The heat is generated by the Joule effect due to electric resistance inherent in the heater tube. Heat flux distribution is axially uniform and laterally non-uniform such as 1.11/1.01/0.65 from outer to inner layer.

Surface temperatures of heater rods were measured with chromel-alumel thermocouples of 0.5mm outer diameter. As shown in Fig. 6, thermocouples were welded immediately upstream of spacers on heater rods of No.21, No.24, No.27, No.30 and No.33 which face to narrow subchannels, since previous burnout experiments showed that dryout usually occurs at these locations.

Simulated pressure tubes used in the experiment were made of Al₂O₃ ceramic and had different inside diameters. The inside diameters chosen in the
experiments were 117.3 mm as a nominal case, 120.0 and 122.4 mm for enlarged cases as shown in Fig. 7. Tolerances of pressure tube I.D. in production were between -0.1 mm and +0.1 mm. The test bundle is centered in the pressure tube accurately within 0.1 mm eccentricity.

Experiments were carried out under the following conditions:

<table>
<thead>
<tr>
<th>Pressure</th>
<th>Mass flow rate</th>
<th>Inlet subcooling</th>
<th>Pressure tube I.D.</th>
</tr>
</thead>
<tbody>
<tr>
<td>7 MPa</td>
<td>2.78 - 11.1 kg/s</td>
<td>42 kJ/kg</td>
<td>117.8, 120.0, 122.4 mm</td>
</tr>
</tbody>
</table>

Thermal-hydraulic conditions were decided from normal operating conditions of the ATR plant. Measuring accuracies were estimated as ±0.5 K for inlet temperature, ±0.5 % for system pressure, ±2 % for inlet flow rate, and ±1 % for electric power supplied to the heater bundle.

After thermal-hydraulic conditions including system pressure and coolant temperature at the inlet of test section and coolant flow rate in the test section were sufficiently stabilized, bundle power was increased so gradually that steady state condition could be kept. The onset of dryout was recognized by the rapid rise in temperature of the heater rod surfaces.

Calculational Procedure

A 1/12 sector subchannel model was used in the present study, as shown in Fig. 7. Subchannel layout is combined-fine type, and the cross section was divided into 39 cells. Each cell has only one wet-wall surface. Hence, liquid film flow rates adhering to heater or cold wall surfaces which have different heat fluxes can be calculated separately. In enlarged cases of pressure tube I.D., the flow area in outermost subchannels of No.1, No.2 and No.3 were varied. The heated length of 3.7 m was divided into 37 axial nodes, as a result, axial node length is 100 mm. Dryout power has a tendency of decreasing with increase of axial node length. The axial node length had been confirmed to be short enough from the result of parametric study done to examine numerical error.

In FIDAS, flow rates of liquid film and entrained droplets are calculated based on the deposition and entrainment correlations which were developed and validated on the basis of experimental data in smooth circular tubes[2]. In bundle geometries, however, there are spacers which are considered to affect entrainment and deposition phenomena. In order to take into account the spacer effect, the droplet deposition rate was tuned as approximately 80% of the value for smooth circular tubes.

Results and Discussion

As shown in Fig. 8, experimental data show that the dryout power decreases clearly with the increase of pressure tube inside diameter. Measured dryout power for pressure tube I.D. of 120.0 mm were approximately reduced to 97.5% of the value for nominal pressure tube I.D. of 117.8 mm. Furthermore, in a case of 122.4 mm, dryout power data were reduced to 90% of the value for the nominal case. Presumably, the enlargement of flow area in outer subchannels leads to

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Fig. 8  Effect of pressure tube I.D. on dryout power

Fig. 9  Prediction accuracy of FIDAS on dryout power

Fig. 10  Comparison between FIDAS and COBRA
the enhanced imbalance in quality and mass flux distributions between subchannels, since coolant tends to flow in broader subchannels. Consequently, this enhanced imbalance in quality and mass flux distribution is considered to reduce the dryout power.

FIDAS prediction indicates by comparisons with experimental data that FIDAS can predict well the measured dryout power without any empirical CHF correlations over the entire range of experimental flow rate, and that FIDAS satisfactorily traces the decrease of dryout power with increase of pressure tube inside diameter.

Comparisons between experiments and predictions by FIDAS on dryout power is summarized in Fig.9. The prediction accuracy by FIDAS on dryout power could be estimated to be ±5%, since almost all of the predictions within ±5% error band which is indicated by broken lines in the figure.

Finally, comparisons between FIDAS predictions and similar subchannel predictions using the COBRA-IV[6] code are shown in Fig.10. As shown in the figure, the prediction by FIDAS traces well dryout power decrease with the enlargement of inner diameter of the pressure tube. On the other hand, COBRA-IV[6] failed to simulate the measurement, although COBRA had been carefully fine-tuned for the 36-rod bundle geometry in advance of this analysis. This is attributable to the fact that the inter-subchannel mixing including void drift is not taken into account adequately in COBRA, which formulates two-phase flow on the basis of the homogeneous mixture model.

CONCLUSIONS

FIDAS was found to be generally successful in predicting flow quality in subchannels, under actual reactor operating conditions.

Comparison between FIDAS prediction and the full-scale dryout experiment shows that FIDAS provides sufficient prediction capability of dryout power for the rod bundle geometry, a capability that exceeds that of COBRA-IV.

It is concluded that the FIDAS code is a powerful tool for nuclear reactor design and safety evaluation.

ACKNOWLEDGMENT

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ABSTRACT

A moderator test facility has been designed as a two-dimensional physical model of a CANDU calandria, capable of reproducing the important thermalhydraulic phenomena in the moderator flow. The purpose of the facility is to provide an extensive experimental database for the verification of moderator computer codes used in reactor safety analyses, and to enhance the understanding of the complex moderator thermalhydraulics. A vital step in computer code validation was to compare computer predictions against data collected during isothermal tests; tests with heat input will be studied later.

Predictions from the MODTURC (MODerating TURbulent Circulation) computer code, which is currently used for licensing calculations, and the next generation of moderator circulation code, named MODTURC-CLAS (CO-Located Advanced Solution), are compared to data collected for several isothermal tests with the moderator test facility. The comparisons show that both the MODTURC and MODTURC-CLAS codes are fundamentally sound, and able to predict flow patterns which are both qualitatively and quantitatively close to observations and measurements. The analysis of the comparisons has already provided some indications of what aspects of the flow modelling could use improvement.

INTRODUCTION

An important safety-related feature of the moderator system in CANDU reactors is its heat-sink capability. In certain Large-Break Loss-of-Coolant Accident (LOCA) scenarios, the moderator is credited with the ability to remove stored and decay heat from fuel in the channels if coolant flow stagnation has resulted in pressure tube deformation (ballooning and/or sagging) into contact with calandria tubes. Predictions of temperature and subcooling distributions in the moderator are required as part of the analysis to ascertain fuel channel integrity in such events.

Moderator circulation within the calandria of a CANDU reactor is complex. It combines the effects of high-speed jets, internal heat generation, and a tube bank. The calandria (Figure 1) is a horizontal cylindrical vessel typically about 8 m in diameter and about 6 m long. Several hundred calandria tubes occupy about 17% of the space in the central region of the calandria. This central region, referred to as the core, is surrounded by the reflector region within which the moderator inlets are located. The heavy water (D$_2$O) moderator fluid is circulated by a system of pumps at a flow rate of about 1 m$^3$/s. It is drawn from the calandria through outlets located near its bottom, cooled in heat exchangers, and returned to the calandria through banks of inlet nozzles located on the sides and/or near the top of the vessel. Normally, the moderator is heated primarily by nuclear radiation, with heat generated directly in the fluid at rates up to 1000 kW/m$^3$ in the core region and about 100 kW/m$^3$ in the reflector region. A small contribution to the total heat load comes from conduction from the vessel walls, calandria tubes, reactivity mechanisms and guide tubes that are also heated by nuclear radiation. In large-break LOCA scenarios, especially those with a postulated failure of the Emergency Coolant Injection System, many pressure tubes may deform into contact with calandria tubes, and the convective heat transfer from these fuel channels would become the dominant part of the total moderator heat load following a reactor trip.

The heat generated in the moderator gives rise to buoyancy forces whose interactions with inlet jet inertia and resistance to flow in the core region determine the flow pattern in the calandria. The flow is generally highly turbulent in all regions of the calandria. Modelling the flow which develops in the calandria under these conditions requires powerful computational tools.

Since the early 1980's, a specialized, internally-developed code named MODTURC (1) has been in use at Ontario Hydro for licensing calculations. MODTURC is a finite difference code which solves the coupled three-dimensional partial differential conservation equations for mass, momentum, energy and two quantities characterizing turbulent diffusion. The equations, formulated in cylindrical coordinates, are discretized and solved on a mesh consisting typically of about 10,000 nodes.

More recently, a computer code with enhanced capabilities,
named MODTURC-CLAS (based on the methods described in References 2, 3, 4, and 5), has been under development jointly by Ontario Hydro and Advanced Scientific Computing Ltd. of Waterloo, Ontario. This code employs the same mathematical formulation of the conservation equations as does MODTURC. The enhancements in this code relate to the use of a non-orthogonal mesh, the incorporation of more accurate equation discretizations, and the implementation of more efficient equation solvers.

The equations solved by both MODTURC and MODTURC-CLAS contain a number of modelling assumptions and idealizations. The moderator is assumed to be single-phase liquid which is treated as incompressible apart from small density variations in the buoyancy terms of the momentum equations. The presence of the fuel channels in the core region is modelled utilizing the concept of isotropic porosity, defined as a homogeneously distributed porosity characterized by the average ratio of fluid volume to total volume in a control volume. Tube resistance to the flow is represented by pressure loss terms (distributted drag) in the momentum equations, based on empirical correlations for flow past a circular cylinder. Heat generated in the moderator and heat transferred to the moderator from the calandria tubes and all other solid surfaces are combined and represented by a volumetric source term in the energy equation. Turbulent diffusion is modelled using the k-ε formulation.

The inclusion of these assumptions and simplifications into the modelling of moderator circulation entails prediction uncertainties whose precise magnitude cannot be assessed a priori. Empirical verification of the predictions is desirable. To aid in achieving this objective, a moderator test facility has been built and is presently in operation at Stern Laboratories Inc. (SLI) in Hamilton, Ontario.

The main objectives of the moderator test facility are to:

(a) reflect the important features of moderator circulation and heat transfer, and be capable of simulating certain real reactor conditions;

(b) yield detailed and accurate qualitative and quantitative data pertaining to the flow pattern and temperature field through flow visualization, velocimetry, and thermometry;

(c) be modifiable to admit separate-effect tests needed to verify individual assumptions and component models in the code in order to pinpoint areas for improvement;

(d) be characterized by a predominantly two-dimensional flow pattern in order to economize the design, fabrication, and operation of the apparatus, and to facilitate data collection and simulation on a computer.

Although basic similarity considerations were taken into account in the design of the facility in order that the behaviour of the apparatus does not deviate significantly from an actual CANDU moderator system, the facility is not intended as an accurate small scale model. It should be stressed that the main objectives of the facility are computer code verification and enhancing the understanding of phenomena pertinent to moderator thermalhydraulics.

The experimental program is funded by the Moderator Circulation Working Party of the CANDU Owners Group (COG), with whose permission these results are presented.
Figure 3. Moderator Test Vessel.

that the jets issue into the vessel at the horizontal centreline. The angle of the nozzles is adjustable from 0° (vertically upwards) to 15° (inward from vertical). The width of the nozzle slot is variable from 8 mm to 16 mm. The outlet port at the bottom of the test vessel is a perforated section which covers the full thickness (0.2 m) and is about 15 mm wide.

The tube heaters are fabricated from Inconel 600 tubing with a 0.25 mm wall, an outside diameter of 0.033 m, and a heated length of 0.2 m. They are designed to provide 340 W at 1.65 volts (DC) each when used as resistance heaters. Alternatively, they can be used as electrodes with an AC power supply to induce heat generation in the fluid itself. All the power connections to the heaters are made from the back face of the test vessel, leaving the front face free for instrumentation and flow visualization.

The following nominal flow and power parameters were chosen to be the most representative of the operating parameters of a typical CANDU reactor operating at full power:

Total inflow rate = 2.4 kg/s
Heat input from electrical heaters = 100 kW
Inlet water temperature = 55 °C
Outlet water temperature = 65 °C

These parameters were arrived at by a similarity analysis of the governing differential equations of motion and energy of the moderator, so as to achieve as closely as possible a dynamic similarity between the water flow in the test vessel and the moderator flow in a typical CANDU calandria.

Instrumentation

The total flow rate into the test vessel is measured using a Herschel type venturimeter, and the flow rate through each of the two inlet lines is measured using square-edged orifice meters with flange taps. The pressure drop across these flowmeters is measured using Rosemount Model 1151 DP differential pressure transmitters. The flowmeters were calibrated using a weigh tank system, and the pressure transmitters were calibrated using a dead weight tester. The estimated uncertainty in the flow rate measurements is ± 0.5 % (2o).

Fluid velocity measurements in the test vessel are achieved by Laser Doppler Anemometry (LDA). The LDA system employed in this study consists of a Lexel 2W Argon-ion laser and TSI Inc. optics and electronics, configured for dual-beam back-scatter operation. The argon laser itself remains stationary, while the optics can be aimed from any desired location on the front face of the test vessel by a two-dimensional traversing mechanism. The optics is connected to the laser via a flexible fibre optic link. The system is equipped with a 250 mm focusing lens, which was found to be optimum in reducing signal noise. The present system can measure only one velocity component at a time, but the in-plane perpendicular component can be measured by subsequently rotating the optics through 90°. The system is capable of making velocity measurements in the range of 0.05 m/s to 100 m/s, at an estimated accuracy of ± 1 % (2o).

A significant source of error is determining the precise location of the measuring volume which, due to refraction of the laser beam through the polycarbonate wall of the test vessel, is known to ± 1 mm of the traversing mechanism location. A TSI Bragg cell frequency shifter has been added to the system, permitting the measurement of very low velocities and reversing flows. The Bragg cell permits measurement of velocities as low as 0.001 m/s at an estimated resolution of 2 per cent.

All power metering, pressure, differential pressure, and LDA instrumentation signals are transmitted to a HP 1000 minicomputer-based data acquisition system. Up to 256 data channels are available and the computer is able to scan, record, convert units, display, and (if necessary) process any of these data channels. All channels are scanned and recorded at the rate of 2 to 3 scans per second. The system is capable of storing, processing, and presenting transient histories of the scanned channels, or time-averaged values of the readings. Typically, the reported instrument readings for steady state test conditions are 6 minute averages.

The qualitative data acquisition is a tracer technique using a pH indicator for flow visualization. In this method, a pH indicator called Bromothymol is dissolved in the water circulating through the test vessel. The indicator is a transparent amber colour when the pH is about 6.0, and is an opaque blue colour when the pH is around 7.6. An injection system is used to inject either an acid or base solution at the pump inlet (see Figure 2) into the circulating water/indicator solution. Acid injection causes the blue indicator to be "marked" by locally changing the pH, thus producing a sharp colour change to amber. When the amber colour permeates the test vessel, base injection is used to locally mark the fluid back to its original blue colour. The reversible process can be repeated a number of times. Flow visualization consists of both direct visual observation and photographic recording (both still and videotape photography).
TEST DESCRIPTIONS

Overview

The guiding principle in designing the test sequence for the first phase of the moderator circulation experiments was that it begin with the simplest configuration, evolving gradually through a series of well-defined steps to the full complexity associated with the moderator flow. It was decided to attempt to isolate certain phenomena through what is termed "separate effect" testing:

(a) Turbulent flow without internal objects or heat addition. This examines the interaction of opposing turbulent jets, and the impact on stability and symmetry of the flow pattern.

(b) The influence of the tube bank on jet development, turbulence generation, and momentum losses without heat input.

(c) The interaction between natural and forced convection in the presence of heat input from the tube bank.

All of the above configurations are readily attainable with the test facility described here. These include testing: with and without the tube bank, with and without heat addition, and with and without the vertical dividing wall which enforces symmetry in the flow pattern.

Separate effect testing is important for identifying and investigating the controlling factors in the moderator flow pattern and temperature field. A major goal of this experimental program, moreover, is the verification of predictions from moderator circulation computer codes. Separate effect testing is useful for verifying various sub-models in the codes in relative isolation of one another, which is essential if compensating errors exist.

Table 1 shows the sequence of isothermal tests in the first phase of moderator circulation experiments which are under consideration here. A similar sequence of tests with heat input also has been designed. Some of those tests have been performed and the data are awaiting analysis. The description and evaluation of those tests will be reported later.

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Tube Bank?</th>
<th>Enforced Symmetry?</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>N</td>
<td>Y</td>
<td>Isothermal 65°C</td>
</tr>
<tr>
<td>2</td>
<td>N</td>
<td>N</td>
<td>-</td>
</tr>
<tr>
<td>2a</td>
<td>N</td>
<td>N</td>
<td>Unequal Inflows</td>
</tr>
<tr>
<td>3</td>
<td>Y</td>
<td>Y</td>
<td>-</td>
</tr>
<tr>
<td>4</td>
<td>Y</td>
<td>N</td>
<td>-</td>
</tr>
</tbody>
</table>

Table 1. Isothermal Test Sequence.

Symmetry can be enforced in the test vessel by inserting a dividing wall at the vertical centreline. These tests are useful as an independent validation of computer codes as well as an investigation into the effect of flow symmetry on moderator subcooling.

A test with enforced asymmetry also appears in the test sequence. Test 2a looks at isothermal flow in the test vessel, with no tube bank, no dividing wall, and with unequal mass flow rates at the two inlet jets.

It is appropriate to define the coordinate system which is used in specifying locations in the test vessel. A Cartesian frame is employed with origin (0,0,0) at the geometric centre of the inside surface of the front face of the test vessel. The x-axis corresponds with the horizontal centreline and is positive directed towards the right. The y-axis corresponds to the vertical centreline and is positive directed upwards. The z-axis is the axial direction (or depth direction, into the plane of the paper) and is positive directed towards the back face of the test vessel. Thus, the front face, mid-axial, and back face planes are defined by z=0 m, z=0.1 m, and z=0.2 m, respectively.

The inlet nozzle slots extend from x=-0.954 m to x=0.946 m (right nozzle) and from x=0.946 m to x=-0.954 m (left nozzle) at the horizontal centreline (y=0 m). They extend approximately the full depth (0.2 m) of the vessel and, in the base configuration, issue jets vertically upwards. The circumferential wall of the vessel is at radius 1 m from the origin.

Occasionally, it is convenient to use "clock terminology" to describe the radial location of various point of interest in the test vessel flow pattern. Briefly, the twelve o’clock position corresponds to the top-most point in the test vessel (x=0 m, y=1 m), the three o’clock position corresponds to the right-most point in the vessel (x=1 m, y=0 m), and so on.

Isothermal Flow, No Tubes, Enforced Symmetry (Test 1)

This test represents the least complicated and most boundary-constrained flow pattern that has been examined in the moderator test vessel. The complicating factors of the tube bank and the interaction between the inlet jets do not play a role.

The flow was isothermal (65 °C), at the nominal total inflow mass rate (2.4 kg/s), and constrained to be symmetric by the installation of the dividing wall at the vertical centreline of the vessel. Two counter-rotating vortices of equal size were set up, with the vortex centres close to the horizontal centreline and approximately midway between the inlet nozzles and the vertical centreline. A relatively wide region of parallel vertical downflow existed.

Since the LDA system was not operational at the time this test was performed, the only data to have been collected so far is flow visualization. In the interest of efficiency, it was decided to proceed with the subsequent tests and to return to this test later, when maintenance dictates that the tube heaters be removed from the test vessel. The LDA velocity measurements planned for this test are to be conducted mainly in the region where the inlet jets develop. This test is believed to be the best for obtaining verification data in the turbulent “reflector” region since the inlet jets are steadier than in Test 2 (see below), and the tube bank shear is not a complicating factor.

Isothermal Flow, No Tubes (Test 2)

This test was designed to examine the influence of inflow jet turbulence on the circulation pattern without the
complicating factors associated with the tube bank, such as: momentum losses, turbulence generation, and buoyancy interaction. Without the vertical dividing wall, the inlet jets were allowed to interact. Hence, this test provides a strong overall test of the turbulence model in moderator circulation computer codes.

In this test the total inflow rate was 2.4 kg/s and the water in the test vessel was isothermal at 65 °C. The flow pattern was very similar to that of Test 1: two counter-rotating recirculation zones of nearly equal size. However, the stagnation point of the two colliding inlet jets fluctuated between the twelve o’clock and midway between the twelve and one o’clock positions on the upper circumferential wall. This slight asymmetry was likely due to small unintended differences between the two inlet nozzles. A confirmation of this was obtained by interchanging the two nozzles, which resulted in a mirror-image flow pattern.

Figures 4 and 5 show LDA measurements of horizontal and vertical velocity components at various x,y locations at the mid-axial plane of the test vessel. Positive horizontal velocities indicate motion to the right; positive vertical velocities indicate motion upwards. These profiles, coarse as they may be, agree qualitatively with the near-symmetric dual-recirculation pattern mentioned earlier.

Isothermal Flow, No Tubes. Unequal Jet Inflow (Test 2a)

Operation of a CANDU calandria with grossly unequal flow rates in the inlet jets is highly atypical, but periodic calandria inlet valve testing in Pickering NGS A has resulted in short term flow imbalances. The primary motivation for this test, however, is to provide a graphic test of the ability of moderator circulation codes to accurately model momentum losses in the reflector region of the calandria. The location of the stagnation point at the collision of the two inlet jet streams is a good flow feature to compare with predictions.

In this test a steady isothermal (65 °C) flow was set up with unequal inflow mass rates at the two inlet nozzles: the left nozzle issued 1.0 kg/s while the right nozzle issued 1.4 kg/s. Hence, the total inflow remained at the nominal level of 2.4 kg/s. A very asymmetric flow pattern was set up, with two counter-rotating recirculation zones, of which the left was much smaller than the right. The centre of the right vortex appeared to be close to the geometric centre of the test vessel. The stagnation point between the two inlet jets established itself at the ten o’clock position, with less apparent fluctuation than in the balanced flow case.

Additional tests were conducted in which the level of imbalance in the inlet mass flow rates was varied, while keeping the total inflow mass rate at the nominal level. As expected, the lower the imbalance, the closer the jet stagnation point moved towards the twelve o’clock position. The displacement of the stagnation point appeared to be a continuous function of the inlet flow imbalance.

Figure 4. Test 2 : Measured Horizontal Velocity Component.

Figure 5. Test 2 : Measured Vertical Velocity Component.

Isothermal Flow, With Tubes. Enforced Symmetry (Test 3)

The only difference between this test and Test 1 was the presence of the tube bank in the flow. Isothermal conditions (65 °C) existed for the nominal flow rate (2.4 kg/s total). The dividing wall at the vertical centreline enforced symmetrical recirculation zones.

From the flow visualization it was evident that the major differences between this flow and Test 1 were the reduced fluid velocity of the downward flow in the core and the higher location of the vortex centres. The less vigorous recirculation left regions of low flow below the inlet nozzles, at the 4 and 8 o’clock positions. Unlike Tests 1 and 2, there was no wide region of parallel vertical downflow in the centre of the vessel; the flow tended instead to spread outwards from the vertical centreline.

Since preliminary flow visualizations of this test and Test 4 indicated no significant difference in the flow patterns, the LDA velocity measurements were restricted to Test 4.
Isothermal Flow. With Tubes (Test 4)

This test is identical to Test 2, except for the presence of the tube bank. Isothermal conditions (65 °C) and nominal inflow (2.4 kg/s) were sustained in a steady flow. There was no qualitative difference between this test and Test 3 (unlike Test 2, which displayed a slight asymmetry when compared to Test 1).

LDA velocity measurements were obtained along the vertical centreline in the tube bank and in the jet development region at two radial locations. These data appear in Figures 15, 16 and 17 in the comparisons with code predictions.

Future Tests

The isothermal tests described here are the first step of the first phase of the moderator circulation experiments. A series of non-isothermal tests is nearing completion; the analysis of those tests will complete the first phase.

Tests in the second phase of the experimental program will deal with different electrical heating modes applied to the tube heaters and modifications to the inlet nozzle width and orientation. Tests to investigate the transient behaviour of the moderator following an abrupt change in operating conditions will also be examined.

The third phase of tests will investigate more specific phenomena of concern to safety analysis. Such phenomena include: non-uniform power distributions in the core, localized boiling in the core, low moderator level effects, and the dispersal of ingressing chemical solutions, such as neutron "poison".

COMPARISONS WITH PREDICTIONS

Overview

The LDA velocity measurements of the isothermal tests without enforced symmetry at the vertical centreline, namely Tests 2, 2a, and 4, are compared to code predictions. No LDA measurements are yet available for Test 1, or are planned for Test 3.

Figure 6 shows the 21 X 31 mesh used by the MODTURC computer code to simulate these tests. This mesh was shown to give rise to grid-independent predictions. As can be seen in the figure, the mesh is defined by orthogonal cylindrical coordinates. The inlet nozzles are approximated by "blocking off" the arc-shaped control volumes corresponding to the nozzle locations. The blocking off is also extended to the small passages between the nozzles and the circumferential wall, so that no throughflow is permitted; this is considered to be realistic since the bottom of the inlet nozzles are almost in contact with the circumferential wall. A staggered arrangement is employed, whereby the liquid temperatures and pressures are stored at different nodal locations than the velocities.

Figure 7 shows the 29 X 27 mesh used by the MODTURC-CLAS code for these test problems. Based on experience with MODTURC, and the improvements inherent in the MODTURC-CLAS approximations, this mesh size should be sufficient for obtaining grid-independent predictions.

A mesh refinement procedure will be undertaken to verify this.

The mesh in Figure 7 is non-orthogonal and the inlet nozzles have been approximated by "notches" in the sides of the computational domain. A co-located mesh arrangement is employed, whereby all solution variables are stored at the same mesh location.

The flow predictions from both MODTURC and MODTURC-CLAS are sensitive to the magnitude of the inlet momentum, for the same inflow mass rate. Due to the presence of wire meshes on the inlet nozzles, which were designed to improve the inflow distribution but which also serve to partially obstruct the jet flow, there is some uncertainty as to the precise magnitude of the inlet.
momentum. This must be resolved by further detailed LDA measurements close to the nozzle openings. The best available estimate of the inlet momentum has been used in all of the code predictions which follow.

In addition to the comparisons reported here, MODTURC-CLAS predictions have been compared to solutions of various standard benchmark problems which have appeared in the literature. These comparisons will be reported elsewhere.

Comparisons with Test 2

MODTURC and MODTURC-CLAS predict virtually symmetric flow fields for this test, as shown in the streamline patterns in Figures 8 and 9. Both predictions show recirculation vortex centres slightly above the horizontal centreline of the vessel and close to the points midway between the inlet nozzles and the vertical centreline, as was observed in the flow visualization of this test.

In both predictions, the inlet jets display a tendency to bend towards the circumferential wall, but this effect is slightly under-predicted by MODTURC. The strong curvature in the MODTURC-CLAS inlet jet prediction is close to what was observed.

Figure 10 shows a comparison of both the MODTURC and MODTURC-CLAS predictions of the vertical velocity component at the horizontal centreline of the test vessel with those measured in Test 2. An estimate of the ±2σ error bound on the measurements appears in the figure. The MODTURC-CLAS prediction is in good overall agreement with the measurements. The MODTURC prediction agrees reasonably well with the measurements in the outer region of the vessel. However, in the region close to the vertical centreline, MODTURC over-predicts the velocity gradient. This effect is compensated for in the presence of a tube bank by over-estimating the flow resistance of the tubes, for the reason given in the comparisons with Test 4.

Comparisons with Test 2a

Figures 11 and 12 show the flow pattern for Test 2a predicted by both MODTURC and MODTURC-CLAS, respectively. The MODTURC prediction displays a large region of essentially horizontal flow and a larger left recirculation vortex than was observed. In this respect, the MODTURC-CLAS prediction, which shows a much smaller left recirculation vortex, is more realistic. The vortex centre of the right recirculation zone is predicted by MODTURC-CLAS to be close to the geometric centre of the vessel, which is in good agreement with observation.

The jet stagnation point, where the momentum of the stronger right jet is finally overcome by the momentum in the weaker jet was observed to be approximately at the 10 o'clock position (30° from the horizontal centreline). The stagnation point in the MODTURC-CLAS prediction is at approximately 43°. The highest point that the left jet in the MODTURC prediction reaches is approximately 23° from CNS 10th ANNUAL CONFERENCE, 1989, 6-31
horizontal. However, there is no stagnation point on the circumferential wall above the left jet in the MODTURC prediction; rather than clinging to the wall, the left jet is pushed away from it towards the vessel interior by the prevailing right jet. This behaviour is consistent with Test 2, in which the MODTURC-predicted inlet jets did not cling closely enough to the outer circumferential wall.

Comparisons with Test 4

Figures 13 and 14 show the flow patterns predicted by MODTURC and MODTURC-CLAS, respectively. The essential qualitative features of the flow patterns are the same: symmetry about the vertical centreline, vortex centres high in the test vessel, and a fanning out of the streamlines in the central region of the vessel. The "fanning out" feature was very apparent in the flow visualization, in which the tracer could be seen to form a "V"-shaped front as it travelled down the vertical centreline of the vessel. It is apparent from the MODTURC-CLAS predicted flow pattern that the regions of low flow observed in the visualization at the 4 and 8 o'clock positions are in the vicinity of stagnation points.

Figures 15 and 16 show the tangential velocity profiles in the "reflector" region of the vessel along radial lines displaced 30° and 60° (respectively) from the horizontal. Both the MODTURC and MODTURC-CLAS predictions of velocity are compared to LDA measurements. An estimate of
CONCLUSIONS

The comparisons of MODTURC and MODTURC-CLAS flow predictions with measurements for several isothermal tests have demonstrated the fundamental soundness of the codes. Some areas for improvement have been identified.

The comparisons between code predictions and Test 4 measurements indicate that the empirical tube drag model used by both MODTURC and MODTURC-CLAS over-estimates the flow resistance. Further refinement of this model will be undertaken.

The Test 2 and Test 4 comparisons showed excess jet momentum in the MODTURC predictions when compared both with measurement and with MODTURC-CLAS predictions. The reason for this is not entirely clear and further comparisons between MODTURC and MODTURC-CLAS are necessary to resolve this difference in the two codes.

Both MODTURC and MODTURC-CLAS under-predict the radial gradient of the tangential velocity in the jet development region. The fact that the jet predictions are more diffuse than the observations likely is due to the absence of wall curvature effects in the turbulence model. The development of appropriate modifications to the k-ε turbulence model has already been initiated.

FUTURE WORK

In addition to resolving the model verification issues presented here, the validation effort for MODTURC-CLAS and MODTURC will continue with the non-isothermal tests of the first phase of the experimental program, for which complete data sets have only recently become available.

The testing, which has been somewhat ahead of computer code verification, is currently in the second phase of the program. The next major series of tests involve investigating the transient response of the test vessel to abrupt changes in flow rate and heat input.
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DETERMINATION OF BLOWDOWN HEAT TRANSFER
BASED ON EXPERIMENTAL INVESTIGATION OF
RD14 EXPERIMENTS. ANALYSIS WITH THE CODE SLLOH

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ABSTRACT

This paper presents the results of analysis performed to better understand blowdown cooling phenomena associated with a rapid depressurisation in a high pressure, high enthalpy system. Large conservatisms are used for the heat transfer coefficients during the blowdown phase of a large loss-of-coolant accident (LOCA) analysis and it is believed that higher heat transfer coefficients can be used if they can be experimentally supported. Detailed analyses of the experimental results obtained in the RD14 facility during rapid depressurisation transients serve this purpose. The SLLOH code (1) was employed to quantify the heat transfer rates during blowdown.

A large amount of data has been gathered during the performance of experiments B8607 to B8610 which has to be analysed in such a way that it can be used for model development and model verification purposes. A particular effort has been made to better understand the behaviour of the system and to relate it back to anticipated behaviours.

The area of interest was the channel thermal-hydraulic behaviour during blowdown and the modes of heat transfer which take place during that period. It is important to be able to assess the amount of water that is left in the channel since it can be credited for fuel cooling before the Emergency Cooling Injection (ECI) reaches the core. In general, system codes using the homogeneous approximation such as SOPHT predicts a very sustained channel voiding leading to high fuel and fuel sheath temperatures.

Because the experimental data indicated stratified conditions in the heated sections toward the end of the initial blowdown period, it was decided to employ the SLLOH code to analyse these tests. This proved successful and serves as a validation of the models employed in this code.

EXPERIMENTS DESCRIPTION

A detailed description of the RD-14 loop can be found in other papers presented at this conference. The RD-14 loop is a full elevation representation of a CANDU primary heat transport system. The RD-14 heated section consists of two 6 metres long horizontal channels with a 5500 kW maximum power each, simulating reactor fuel channels. Each channel contains 37 electrically heated fuel elements simulators, which have uniform heat flux distribution with about the same heat capacity as reactor fuel.

Two pumps are used to circulate the coolant around the loop and their flow rate through the channel is similar to flows through a CANDU reactor channel. Under normal operating conditions, the heat generated in the channels is removed by two steam generators.

Figure 1 gives a schematic of the RD14 test facility with the identification of the broken header and the critical and broken pass. The critical pass is the pass upstream of the break and the driving force of the flow through the channel is limited during a large LOCA. The broken pass is the pass that has the break on one of the headers.

![Figure 1: Simplified Diagram of the RD14 Facility](image)

In this paper we analyse the large header breaks experiments numbered B8607 to B8610. Those experiments have been performed under different conditions, with or without pumps after trip and with or without ECI. The qualitative information we were interested in, was independent of these particular system conditions and we could therefore use all the experiments as one single set.

All the experiments analysed are large inlet header breaks. The experimental procedure consisted of bringing the loop to the desired operating conditions and opening the fast acting relief valve to simulate a break. The power was then tripped shortly after the beginning of the transient (about 2 seconds) and the power of each simulated channel was reduced to 385 kW.

We have been particularly interested by the heater elements sheath heat-up rates and heat-up times since these two parameters when interpreted can provide a direct indication of element cooling and flow regimes.
Figure 2 gives a cross section of the RD-14 heater bundle indicating the heater element numbers.

**FIGURE 2: RD-14 HEATER BUNDLE CROSS SECTION**

**EXPERIMENTS ANALYSIS**

The description of the analysed experiments will concentrate on experiment B8607. Nevertheless all experiments have been analysed and have shown very similar transients to B8607 and the conclusions we have reached are based on all available data.

Figure 3 gives the pressure transient at the inlet and outlet headers of the critical pass for experiment number 8607.

**FIGURE 3: INLET AND OUTLET HEADER PRESSURE TRANSIENT**

It can be seen that the inlet header pressure (IH1 on Figure 1) which is higher than the outlet header pressure at time 0, falls almost instantaneously below the outlet header pressure after the opening of the fast acting relief valve. Due to that, good reverse flow cooling of the heater elements in the broken pass is expected after shutdown.

In all the experiments, in the early part of the transients, a sharp increase in sheath temperature is observed in the broken pass as shown in Figure 4 for experiment number B8607.

**FIGURE 4: BROKEN PASS TEMPERATURE & POWER TRANSIENTS**

This increase is due to the almost instantaneous flow reversal that takes place at the time of the break, leading to a transient period of low flows. For two seconds after the break, the heater power is maintained at 100% full power and is reduced to 20% in about 10 seconds. That combination of low flow and high power induces sheath dryout and explains the temperature excursion which is suppressed as soon as the flow is reversed.

Figure 5 gives the experimental sheath temperature transients for the top, middle and bottom rod (positions number 1, 37 and 10 in Figure 1) at the central location of the critical pass channel and the calculated saturation temperature corresponding to the measured channel inlet pressure for experiment B8607.

**FIGURE 5: TOP, MIDDLE AND BOTTOM SHEATH TEMPERATURES**

At the opening of the break valve, the flow acceleration provides a good heater element cooling and from 10 to 80 seconds into the transient, the sheath temperature follows the saturation temperature.

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At the end of the ECI phase flow stratification occurs immediately as shown by the first heat up of the top rod, followed by the heat up of the lower rods. Some slug flow can also be observed as displayed by the heatup behaviour of the centre rod for which the sheath first heats up under steam cooling conditions and is then cooled down by a larger flow rate. The delays in heatup are very noticeable in the experiments and show that the channel is slowly voided through steaming.

All experiments show extremely good heater element cooling during the rapid blowdown phase (from 10 to 30 seconds after the start of the experiment). This shows that significant flows occur in the heated sections and the heat removal capability of the system is adequate. At 100 seconds the ECI injection is terminated due to lack of water in the injection tank. At that point the heater elements heat up as displayed in figures 5 starting with the top elements first. The whole transient shows that the water present in the channel is not flushed out at the onset of the break and that only steam or a very high quality mixture is leaving the channel during the blowdown phase.

The heater element temperature transients show clearly that the channel is cooled by steam flow under stratified conditions and that the heat up rate of the heater elements is much lower than the adiabatic heat up rate. The observed heat up rate is about 3 C/s for all experiments while the calculated adiabatic heat up rate is about 7 C/s.

Figure 6 shows for the four experiments the time to heat-up for the top, middle and bottom rods.

The difference in time to heat-up the bottom rod after the start of the heat-up of the top rod corresponds almost exactly to the time it takes to vaporise all the water contained in the channel taking into account the channel residual power. This tends to show that the driving force that empties the channel is the channel power itself through boiloff.

Figure 7 gives the calculated and measured sheath temperature transients for a top element at the center of the channel for experiment number B8007.

It can be seen that the calculated heater element temperature is very close to the experimental heater element temperature. At the beginning of the transient the temperature is overpredicted due to the fact that in the calculation we assume only turbulent steam cooling. Nevertheless even in that period the calculated temperature remains low due to the large steam flow generated by the important header to header pressure gradient. These observations tend to show that the models used in SSLOH represent the physical phenomenon and that additional information can be inferred from the calculation. As previously deduced from the experiments, the void fraction in the channel is quite high after ECI interruption and channel stagnation. The average channel void is less than 20% and reaches 50% after about 100 seconds for all experiments. These values have been determined by analysing the experimental temperatures profiles. Very similar values were observed in the simulation.

By analysing the experimental results, we have shown that during the blowdown phase of the experiment and before ECI actuation the heat removal capability of
the system is quite high and no sheath temperature excursions are observed in the critical pass. After ECI interruption and even without ECI actuation, heat-up of the sheath is half the adiabatic heat-up rate and the channel heat up is gradual with the top elements heating up first followed by the lower elements up to 80 seconds later for the bottom elements.

CONCLUSION

Using the experimental results available from RD14 tests, we have shown that during a large inlet header break LOCA the channel does not empty at the time of the break and that large flows cool the core. The temperature excursion appearing at the beginning of the transient is temporary and short lasting. The critical pass experiences a sustained low flow, but remains well cooled by steam generation due to the water that stays in the channel after the blowdown phase.

In case of ECI interruption or loss of ECI, channel heat-up is limited since the water remaining in the channel cools the sheath through steam production and steam cooling. The driving force of the steam flow is the steam production itself and the header to header pressure drop due to the remaining of the system.

The predictions shown with the SSLOH code are quite accurate when compared to the experimental values and can be used to validate the model represented in the code.

The development of two fluid codes such as TUF, should enable the calculation of such phenomena as phase separation in the channel and heater element cooling through steam and water at saturated temperature.

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CATHENA SIMULATION OF THERMOSIPHONING IN A MULTIPLE-CHANNEL PRESSURIZED-WATER TEST FACILITY

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ABSTRACT

The purpose of the work reported in this paper is to assess the ability of the computer code CATHENA to simulate the two-phase thermosiphoning behavior of the RD-14M multiple-channel facility. This paper presents the CATHENA idealization of the RD-14M facility and briefly discusses some of the unique modeling features that were required to represent it. Simulation results are compared with experimental results, and it is demonstrated that CATHENA correctly simulates many of the events and phenomena, in particular bidirectional flow.

INTRODUCTION

During some postulated accident conditions, decay heat is removed from a CANDU reactor core by two-phase natural circulation or "thermosiphoning" of the primary coolant. A series of experiments were performed in the RD-14M test facility at the Whiteshell Nuclear Research Establishment (Whiteshell) to gain a better understanding of two-phase natural circulation in a CANDU-typical heat transport system. These tests also provide a database that can be used to assess and validate computer codes, such as CATHENA.

CATHENA is a one-dimensional two-fluid thermal-hydraulics computer code developed at Whiteshell primarily to analyze postulated loss-of-coolant accident scenarios for CANDU nuclear reactors (1). It uses a staggered-mesh, semi-implicit, finite-difference solution method that is not transit-time limited (2). It has an extensive wall heat transfer package that includes radial and circumferential conduction, thermal radiation and the zirconium-steam reaction. Component models representing pumps, valves, pressurizer, break discharge, and steam separators are included. CATHENA continues to be validated against a number of separate effects, component and integral tests (3). RD-14M test results represent one basis for the integral tests used for validating CATHENA.

In this paper the RD-14M facility and the natural circulation test procedure are briefly described. The CATHENA nodalization of RD-14M is presented and discussed. Experimental and CATHENA simulation results from an RD-14M partial inventory thermosiphoning experiment conducted at high primary power (160 kW/pass) and high secondary side pressure (4.6 MPa) are compared.

FACILITY AND TEST PROCEDURE

The CANDU Owners Group (COG) are conducting experiments to provide a better understanding of the transient behavior of reactor cooling systems and to provide the data needed to revise and validate computer models. In the past, this data was obtained from small-scale pressurized-water loops (4). However, to study nonequilibrium and multiple-channel effects, the large-scale RD-14M facility was built.

The RD-14M primary circuit is shown schematically in Figure 1. It has the essential geometric and physical characteristics of a typical CANDU heat transport system. It incorporates the basic "figure-of-eight" geometry of a CANDU reactor and maintains a 1:1 vertical scaling of a typical reactor. It has two full-height recirculating U-tube-type steam generators, complete with internal preheaters and cyclone steam separators. Primary fluid flow is provided by two high-head single-stage centrifugal pumps. The facility has ten full-length channels complete with end-fitting simulators. The channel inlet and outlet feeder piping arrangements are designed to represent
Darlington NGS feeders and are trace heated to minimize heat losses. Each channel contains seven electrically heated rods with similar heat capacity and surface heat flux as reactor fuel. The emergency coolant injection system, the blowdown system and the header interconnects were not used for the test considered in this paper and are not shown on Figure 1.

In the test presented in this paper, the loop was initially brought to conditions of stable, single-phase, natural circulation (primary pumps off) of the primary fluid at preselected heated section powers, feeder trace heating powers, primary circuit pressure and secondary circuit pressure. After isolating the pressurizer, controlled intermittent draining of primary fluid was begun to introduce void into the primary circuit. Five draining operations of approximately 2% of loop inventory were followed by drains of approximately 10%. The experiments were terminated when two heater sheath temperatures on any heated section exceeded 600°C.

The RD-14M test facility and test procedure are described in more detail in a companion paper entitled "Two-Phase Natural Circulation Experiments in the RD-14 Multiple-Channel CANDU Thermalhydraulics Test Facility".

CATHENA IDEALIZATION

The idealization presented here was developed to simulate the various types of tests to be conducted in the RD-14M facility. These tests include primary heat transport system breaks with and without emergency coolant injection and decreasing primary system inventory tests. To enable two-fluid nonequilibrium effects encountered in these various tests to be represented, horizontal and vertical sections of piping were modelled separately. Since the purpose of this work was to assess the predictive capabilities of the code, a detailed idealization was used to reduce the possibility of unconvolved solutions or inadequate modelling. The idealization used for these simulations, shown in Figures 2 and 3, contains 422 thermal-hydraulic nodes, 423 links and 869 wall heat transfer surfaces. For clarity the primary and secondary side nodalizations are shown separately.

Primary Circuit Nodalization

Figure 2 shows the primary circuit nodalization. The portion of the primary circuit below the headers is comprised of two identical passes of five heated channels. All ten channels are modelled separately, but for clarity, only one set of channels is shown. The seven fuel element simulators (FES) were modelled as 3 pin groups at a lower, middle and upper elevation within the channel. This allows for the heat transfer from the FES to the liquid and vapour during stratified flow conditions to be correctly represented. Heat losses to the environment from all piping, including the feeders, were modelled using imposed heat transfer coefficients and an ambient temperature of 20°C. In the thermosiphoning test considered in this paper, power was applied to the inlet and outlet feeders and end fittings (via heat tracing tape) to reduce heat losses. In the simulations, both the power input to the pipe walls and the heat losses were modelled. Each end-fitting simulator was represented by two branches, one representing the moving volume of fluid in the annulus and the other representing the stagnant volume of fluid behind the shield plug. These volumes are separated by an orifice whose flow area is equal to the clearance between the shield plug and the liner tube. During depressurization of the primary circuit, it is expected that fluid in the stagnant volume behind the shield plug will be forced through the thin clearance between the shield plug and the liner tube, into the primary circuit.

The RD-14M headers are divided into four sections to capture the effect of stagnant volumes in the ends of the headers and any effects resulting from the axial distribution of feeder connections. Each outlet header has a pressure relief valve connected to it for over-pressure protection. The piping leading from the headers to the relief valves represent a significant volume and were included in the idealization.

Secondary Circuit Nodalization

To date all simulations of RD-14 and RD-14M thermosiphoning tests, using CATHENA, have been predicted using imposed conditions of temperature and heat transfer coefficient to model the secondary circuit. This was done to simplify input preparation and reduce computation time. Previously, good agreement had been achieved for some thermosiphoning cases using this simple method (5). However, other cases indicated that more sophisticated modelling of the secondary side was required to capture the primary-secondary side interaction (6). The idealization that was developed to simulate RD-14M steam generator secondary side (7) is shown in Figure 3. Only the inlet feed
primary flow inlet and outlet. All heat transfer coefficients and temperatures applied to the outside of the primary tubes are calculated by CATHENA. Separation of steam and water is accomplished using the "separator" model, which is described later. The secondary side circuit upstream of the steam generators and the secondary side control system were not modelled.

Header Nodalization

Figure 4 shows a cross section of an RD-14M outlet header and the orientation of the feeder connections. Many thermalhydraulic models treat headers as highly mixed vessels, that for many postulated situations, such as rapid depressurization, is adequate (8). However, should stratification occur within the header, different void conditions are expected at the various feeder connection locations. It therefore becomes necessary to capture the behaviour of a header in more detail to simulate channel-to-channel interactions.

Although two-phase flow within a header is inherently multidimensional, many features of two-phase flow within a header may be represented by a one-dimensional model. In the CATHENA code this is accomplished using a "separator" model.

This model determines the void fraction transported to a feeder as a function of the liquid level in the header. In a normal CATHENA calculation, because the code uses an upwind finite-difference algorithm, the phase volume fraction transported between nodes is simply the phase volume fraction of the "upwind" node. In the "separator" model, the upwind node phase volume fraction is replaced by the volume fraction of that phase which would be seen given the feeder diameter and its physical location on the periphery of the header. The effects of vapour pull-through and liquid-entrainment at the junction are included through the use of correlations by Kowalski and Krishnan (9). The "separator" model therefore allows the passage of liquid, two-phase or vapour depending on the "liquid level" in the header and the velocity of the flow in the feeder.

RESULTS

Figures 5 to 16 show the simulated and observed results for the high-power (160 kW/pass) and high secondary side pressure (4.6 MPa) test T8808. Both the simulation and the experiment were terminated by high PES temperature trips on heated section 9 at approximately 4550 s and 5200 s, respectively.

Primary Pressure

Figure 5 shows the primary circuit pressure history with the drain start times and loop inventory at the completion of the drains indicated. Each drain operation results in a reduction in the primary circuit pressure. The rise in primary pressure after 4200 s results from the reduction of heat transfer to the secondary side, caused by flow stagnation above the headers. Generally good agreement can be seen throughout the simulation except for the premature rise in pressure after 4200 s.

Primary Circuit Flow

The primary circuit flow above the headers, measured at the primary pump 1 outlet, is shown in Figure 6. Each drain sequence can be identified by the change (increase or decrease) in flow. It can be seen that good agreement was generally achieved throughout the simulation. After the drain at approximately 2000 s the code slightly overpredicts the flow. The decrease
in flow, at 2600 s in the simulation, results from void passing over the top of the primary tubes, and collecting in the cold leg side, retarding the flow. It is believed that the overprediction of flow starting at 2000 s is due to an overestimation of the heat removal rate in the steam generator. This causes an underestimation in the amount of vapour predicted to accumulate in the cold leg side of the primary tubes. The oscillatory behaviour after 3400 s is not predicted, probably for the same reason. Following the drain after 4000 s sufficient void penetrates the steam generator to stop the primary flow.

Channel Flows

Figures 7 through 11 show the flowrates in heated channels 5 through 9 starting with the highest elevation channel (HS5) and ending with the lowest elevation channel (HS9). Experimental results show that all channels exhibit similar behaviour for the first 4000 s, with the smallest channel flowrates being recorded by the top two channels (HS5 and HS6). These channels had the lowest flows, in this pass, because they have orificed inlet feeders and the lowest channel powers. The simulation generally agrees well with the experimental results except for the greater degree of instability predicted early in channels 7 and 9. Flow instabilities similar to that predicted in HS9 were observed and predicted in HS14, the identical channel location in the other pass. The instability seen after 3400 s was earlier attributed to void penetration into the cold leg section of the primary tubes. This is supported by the fact that all channel flows show the same behaviour (i.e., onset of oscillations and period) as the above header flow, even though independent channel behaviour has been observed. Since all feeders empty into the outlet headers (normally), their flow behaviour is influenced by the flow above the headers. The simulation does not predict the channel oscillations because the above header flow oscillations are not predicted.

After 4200 s, above-header flow stagnation occurs and independent channel behaviour is seen. Flows continue forward in some channels and reverse in others, hence the term "bidirectional" flow. Generally it has been observed that the upper channels reverse, however, in this test, that did not happen. Only the uppermost channel and the middle channel reversed in this pass. The simulation results show that CATHENA predicted the flow directions correctly in all channels, initially. Premature stagnation was then predicted to occur in heated channels 6 and 9. Heated section 7 was also predicted to stagnate, although a large negative flow was observed in the experiment. Both heated sections 7 and 9 are connected horizontally to the headers (see Figure 4). Both are predicted to stagnate because the liquid-vapour interface in the stagnated headers dropped below the entrance to the feeder-header connection. It seems likely that the amount of void predicted in both inlet header 6 and outlet header 7 may have been overestimated. Heated section 7 had a reverse flow and was being supplied liquid from outlet header 7, whereas heated section 9 had a forward flow and was therefore supplied liquid from inlet header 6.

Channel FES Temperatures

An upper FES temperature history near the middle of heated section 9 is shown in Figure 12. In both the simulation and the experimental results, heated section 9 experienced a FES temperature excursion large enough to terminate the experiment. The FES temperatures rise prematurely in the simulation because the channel is predicted to stagnate prematurely.

Void Fraction

Figure 13 shows the void fraction history at the outlet of pump 2. It can be seen that once thermosiphoning breaks down above the headers, the pump dis-
charge lines quickly become steam filled. The inlet headers act as separators allowing void to vent from the top of the header up into the pump discharge lines and liquid to be supplied to those channels that maintain a forward flow direction. CATHENA captures the separation effect in the headers and correctly predicts high quality steam flows into the pump discharge lines.

**Fluid Temperature**

Shown in Figure 14 is the fluid temperature history at the outlet of steam generator 2. The decreases in temperature at 400, 1000 and 1900 s are a result of surges in the feedwater flowrate into the steam generator preheater. The steam generator control systems were designed to operate at high powers and are not sensitive enough at the very low powers used in these natural circulation experiments. Since no modelling of the control systems was included in the secondary

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FIGURE 8: HEATED SECTION 6 INLET VOLUMETRIC FLOWRATE

FIGURE 11: HEATED SECTION 9 INLET VOLUMETRIC FLOWRATE

FIGURE 9: HEATED SECTION 7 INLET VOLUMETRIC FLOWRATE

FIGURE 12: HEATED SECTION 9 TOP FUEL ELEMENT SIMULATOR SHEATH TEMPERATURE, MID-CHANNEL

FIGURE 10: HEATED SECTION 8 INLET VOLUMETRIC FLOWRATE

FIGURE 13: PUMP 2 OUTLET VOID FRACTION
side idealization, the surges in flow were modelled by altering the imposed feedwater flow conditions to the preheater section.

Good agreement was achieved except for the higher predicted temperatures between 2000 and 4000 s. These higher temperatures result from the overestimated primary flowrates. Most of the primary fluid subcooling occurs in the preheater section of the steam generator, and at the higher predicted flowrates the it is unable to maintain the same degree of subcooling. Once stagnation occurs the code underpredicts the temperature slightly.

Effect of 'Separator' Model

Figures 15 and 16 show the primary pressure and loop mass flowrate, respectively, of a previous simulation of this test without the separator models included in the primary side idealization. Comparison of these figures and Figures 5 and 6 show that better pressure and flow agreement was achieved early in the test, with the steam separators included, especially following draining operations. During the simulation varying amounts of fluid separation was predicted to occur within the outlet headers. Surges in the flow of vapour occurred, particularly after draining operations, and varying degrees of fluid separation was predicted in the headers. The pipework leading to the steam generators, which connect to the top of the outlet headers, saw very high quality steam for short periods of time. The overall effect was that the transit time of the vapour from the the heated section where it was generated, to the steam generator where it was condensed, was reduced. This resulted in the more rapid pressure reductions and sharper increases in flow that are seen immediately after draining operations.

DISCUSSION

The mechanism that causes bidirectional flow and what happens to the void that is generated have been the subject of much discussion. Analysis of the CATHENA simulation for test T8808 offers the following explanation. As the primary pressure decreases, due to draining, the temperature of the fluid entering the steam generators decreases. Since the secondary side temperature remains constant, the decreasing temperature difference between primary and secondary sides reduces the effectiveness of the steam generators. Void on the primary side penetrates further into the hot leg side of the steam generator primary tubes, which increases the buoyant head and the overall primary flowrate. Eventually void is able to pass over the top of the primary tubes and collect in the cold leg side. Further inventory reductions, after this time, result in decreases in overall flow, as more void collects in the cold leg side. Finally enough void passes over the top of the primary tubes to cause the flow to stagnate. At this time, phase separation occurs in the outlet headers, the liquid in the feeder rising to the steam generator can drain back into the header and thermosiphoning above the headers stops. Fluid in the heated channels, at this time, is still trying to flow in the forward direction because of the buoyant head generated by void in the outlet feeders. As pressure builds in the outlet headers, the channel flows slow. Steam continues to be generated in the channels and is forced out into the inlet end fitting and into the inlet feeders. Once void reaches a vertical section of inlet feeder piping, that channel reverses flow direction.

Generally, it is the upper channels that reverse flow direction, partly because it is these channels that have vertical feeder sections connected closest to the heated channels and partly because they also have the shortest vertical sections of piping from which to generate the driving head needed to maintain forward flow. It appears to be the amount of void present in the outlet feeders and channels at the time of flow stagnation that tends to introduce randomness.
in determining which channels reverse. Once a channel or channels reverse, the pressure at the outlet header is relieved and the magnitude of the flows in all channels increases temporarily.

Once bidirectional flow is established, void is transported to the inlet headers. The void separates in the headers, and steam vents into the pump discharge line where it condenses. The piping between the pump and steam generators acts like a trap, temporarily preventing void from progressing past the pumps. Void generated in the channels is now transported up to the primary pumps on the inlet side. Since the reflux condensation that occurs in the steam generator inlet side and the pump discharge lines is not as effective as unidirectional flow through the steam generators, the primary pressure begins to rise. Flow stagnation within a channel or channels continues until the high-temperature limits of the FES are reached and the experiment is terminated.

CONCLUSIONS

The CATHENA idealization of the RD-14M facility was presented along with simulation results for a high-power, high-pressure partial inventory test. The results show that CATHENA can simulate most of the experimental features including the bidirectional flow behaviour observed in the RD-14M experiments. One possible explanation, derived from the CATHENA simulation, of the events leading up to the onset of bidirectional flow and the events that cause it, has been presented.

ACKNOWLEDGEMENTS

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REFERENCES


SIMULATION OF "STANDING START" BEHAVIOUR BY THE CATHENA THERMAHYDRAULICS CODE

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ABSTRACT

This paper describes CATHENA simulations of the thermalhydraulic behaviour of a CANDU-type fuel channel and feeder system in the absence of forced flow. To initiate flow through the channel from stagnant subcooled liquid conditions, significant void may have to be generated to imbalance the hydrostatic head in the liquid-filled inlet and outlet feeders. To study this phenomenon, "standing start" tests were conducted in the Cold-Water Injection Test (CWIT) facility at Stern Laboratories Inc. (formerly Westinghouse Canada). Selected CWIT "standing start" tests and the results predicted using CATHENA are presented. The results show that CATHENA is able to accurately represent the behaviour experimentally observed.

EXPERIMENTAL FACILITY

A schematic representation of the CWIT facility is shown in Figure 1. The facility consists of two horizontal heated channels (the bottom heated channel configuration is shown in Figure 1), two headers, two break simulation valves and the interconnecting feeder piping. Only the bottom heated channel was used for the tests that will be described (the inlet and outlet feeder pipes for the upper heated channel were "blanked off" just below the headers). The Zr-2, horizontal channel contains an electrically heated 37-pin Fuel Element Simulator (FES) bundle having a heated length of 6 m. The channel is connected horizontally to the feeder piping through "off-set" end-fitting simulators. The end-fitting simulators are 581 mm in length and to make an annular flow path consisted of an outer 10-in, Schedule-120 pipe surrounding an inner 8-in, Schedule-160 pipe.

For these experiments, the facility was set up in a symmetric double-break, double-injection configuration with no orifices at the break flanges. Tests were performed with various combinations of injection and feeder orifice sizes. Injection orifices of 19.9- and 14.3-mm throat diameter were used for the total flow and orifices of 13.3- and 49.2-mm throat diameter were used at the headers. Injection pressures were held constant at 1.6 or 1.9 MPa and the blowdown tank pressure was varied from 0.1 to 1.0 MPa. The power supplied to the FES varied from 50 to 200 kW.

Experimental Procedure

To preheat the facility, water at 30°C or 100°C was circulated from the injection system through the outlet header break location. After uniform loop temper-
temperatures were established, the inlet header break valve was opened and the desired injection and blowdown tank pressures were established. At data-scan time zero the required FES power was supplied. The experiment was terminated when all of the FES surface temperatures cooled down or a maximum FES surface temperature of approximately 780°C was reached.

CATHENA SIMULATIONS

The CATHENA code was developed at WNRE primarily for the analysis of postulated loss-of-coolant accident (LOCA) events in CANDU reactors. The code uses a staggered-mesh, one-step, semi-implicit, finite-difference solution method, which is not transit time-limited. The extensive wall heat-transfer package includes radial and circumferential conduction, thermal radiation, and the zirconium-steam reaction. The heat-transfer package is general and allows the connection of multiple wall surfaces to a single thermohydraulic node. The CATHENA code also includes component models required for complete loop simulations, such as pumps, valves, a pressurizer, break discharge, separator models, and extensive control system modelling capability.

The results of two CATHENA simulations will be shown. The first, CWIT-937, is a low-pressure (0.3 MPa at the channel) test with a FES bundle power of 100 kW. The second, CWIT-939, is a higher pressure (0.6 MPa at the channel) test with a power of 150 kW. In both cases, the feeder orifices had a throat diameter of 25.4 mm, and the loop preheat temperature was 30°C.

Facility Idealization

The CATHENA idealization of the CWIT facility used for these "standing start" simulations is shown in Figure 2. The idealization comprises 117 nodes and 118 links. Condensation of the steam generated in the channel by the cold metal at the ends of the channel and in the end-fittings is an important phenomena in these experiments. Therefore, careful attention was paid to the characterization of the metal mass within the horizontal test section including the channel-end roll joints, the unheated sections of the FES pins, and the end-fitting simulators. The additional cooling at the FES pin ends caused by the O-ring cooling circuit was not included in the idealization because the information was unavailable. Because the piping was well insulated and relatively low metal temperatures were seen in these experiments, all external surfaces were assumed to be adiabatic.

The 37-element FES was idealized into 10 "pin" groups to represent the heat transfer split between the liquid and vapour phases under the stratified flow conditions expected. To adequately represent the exposed surface area available for condensation in a stratified flow, the unheated sections of the FES were similarly idealized. Both radial and circumferential conduction were used in modelling of the pressure tube, end fitting, end-fitting simulator, the pipe between the channel and the end-fitting simulator, and the horizontal feeder sections. Circumferential conduction was included to more accurately simulate the temperature distribution in the metal subjected to stratified flow conditions.

In the thermohydraulic idealization of the experimental facility, CATHENA "separator" models were used between the end-fitting simulators and the horizontal feeder pipes, and between the horizontal feeder pipe and the first inclined feeder pipe. Between the end-fitting simulator and the horizontal feeder pipe, the model was used to allow the steam to be retained by the large diameter annular passage in the end-fitting simulator for liquid levels above the entrance elevation of the feeder. Between the horizontal and inclined feeder pipes, the "separator" model was used to represent the escape of vapour around the vertically inclined elbow.

Test 937

In the "standing start" test CWIT-937 the injection and blowdown tank pressures were 1.6 MPa and 0.2 MPa, respectively. The FES electrical power was 100 kW. The injection orifices used had a 14.3-mm throat diameter for the total flow and a 13.3-mm throat diameter at both headers. The feeder orifices in the experiment were 25.4-mm in diameter. Initially, the channel pressure was 0.3 MPa and the water temperature was 30°C.

The results of the CATHENA simulation of CWIT-937 are shown in Figures 3 through 7.

Outlet Feeder Void Fraction. The calculated void fraction in the vertical outlet feeder is compared with experimental fluid-density measurement in Figure 3. The conversion from the densitometer voltage output to void fraction was unavailable. Therefore, only a comparison of the timing of events in the CATHENA simulation and in the experiment could be made. In the simulation the vapour reached the vertically inclined outlet feeder, which started the venting process, at approximately 840 s. This time is in excellent agreement with the experimental venting time of 860 s.

Channel Upper FES Temperatures. Figure 4 compares the measured and the simulated FES temperatures for the upper pin near the inlet and outlet ends of the channel. The maximum FES surface temperature of 620°C calculated by CATHENA is in excellent agreement with the 630°C maximum measured. The slower temperature increase seen in Figure 3 for the measured FES temperature near the heated section inlet is believed to be the result of either a local non-uniformity in the
heater, or a bad thermocouple. This slower temperature increase was consistently observed, at this location, in other experiments in the test program.

Although the venting time was accurately simulated, the time period required to refill the channel was underestimated in the simulation. This is attributed to an underestimation of vapour generation in the channel and end-fitting simulator as subcooled water entered the channel after venting began. The underestimation of vapour generation in the simulation was attributed to a combination of two processes that were not modelled. First, axial conduction between the shell of the end-fitting simulator and the feeder pipe, and second, liquid natural circulation near the beginning of the experiment between the end-fitting simulator and the channel. Both these processes tend to preheat the liquid entering the channel as venting began. Inlet liquid preheating decreases condensation and increases vapour generation in the channel, thereby slowing the refill process.

Channel Bottom FES Temperatures. In Figure 5, the CATHENA calculation of the FES temperatures for the bottom pin near the inlet and outlet ends of the channel are compared with the experiment. The timing of the arrival of subcooled liquid at these locations in the simulation, indicated by the decrease in FES temperature, is in good agreement with experiment. At the start of the experiment, the initial temperature rise calculated for the bottom pin is in good agreement with experiment. However, after approximately 120 s the measured decrease or "leveling-off" in the bottom pin FES temperature was not captured in the simulation. This phenomenon was attributed to single-phase liquid convection between the channel and the large diameter annular passage in the end-fitting simulator at both ends of the channel. As the water is heated in the channel it rises and moves toward the unheated ends of the channel because of the density gradients induced by heating. When heated liquid reaches the entrance to the end-fitting simulator, it rises into the upper part of the annular passage displacing colder liquid into the bottom of the channel. This liquid natural circulation cannot be represented in a one-dimensional model such as CATHENA because only one liquid velocity is available at any spatial location. This liquid natural circulation may not affect the time required to vent the channel because the sensible energy transferred to the metal in the upper part of the end-fitting is small compared to the latent heat available after vapour generation begins in the channel.

Feeder Temperatures. Figures 6 and 7 show the fluid temperatures measured in the horizontal feeder pipes near the inlet and outlet end-fitting simulators, respectively. The calculated liquid, vapour and the upper sector end-fitting outer shell temperatures at the respective ends of the channel are also shown in Figures 6 and 7. The vapour temperature is only shown in the figures during the time when void existed at this location. In the simulation, the liquid temperature in the outlet feeder did not reach saturation until approximately the time channel venting began and remained subcooled in the inlet feeder. In the experiment, however, the measured fluid temperature reached saturation at both locations at approximately 400 s.

The difference between the calculated and measured fluid temperatures is attributed, in part, to the described single-phase natural circulation between the channel and the end-fitting simulator. After approximately 400 s the measured fluid temperatures at these locations exceeded saturation indicating the presence of superheated vapour. Figure 7 shows the good agreement obtained between the calculated vapour temperature and the measured fluid temperature in the outlet feeder until venting began. However, in the inlet feeder, vapour was calculated to be present for a much shorter time (see Figure 6) than was experimentally indicated. This difference is attributed to heat conduction (axial) between the massive shell of the end-fitting simulator and the feeder pipe. Axial conduction was neglected in the CATHENA modelling. This is also presumed to be the reason for the late arrival and disappearance of vapour at the outlet feeder in the simulation.

In summary, the CATHENA simulation of the "standing start" test CWIT-937 shows good agreement with experimental observation.

Test 939

In test CWIT-939 the injection and blowdown tank pressures were 1.6 MPa and 0.5 MPa, respectively. The injection and feeder orifices used were the same as in
the previously described experiment (CWIT-937). The FES electrical power was 150 kW. Initially the channel pressure was 0.6 MPa and the initial water temperature was 30°C.

The results of the CATHENA simulation of CWIT-939 are shown in Figures 8 through 12.

Outlet Feeder Void Fraction. The calculated void fraction in the vertical outlet (inlet) feeder is compared with fluid-density measurement in Figure 8. The conversion from the densitometer output voltage to void fraction was unavailable, therefore, only a comparison of the timing of feeder void fraction in the CATHENA simulation and in the experiment could be made. In the simulation the vapour reached the vertical inlet feeder, which started the venting process, at approximately 605 s. This time is in excellent agreement with the experimental venting time of 600 s, however, in the experiment the venting occurred through the outlet feeder. In this series of experiments, venting was observed from either the inlet or outlet feeder, however, a preference for venting from the outlet feeder was exhibited. This indicated some geometrical asymmetry in the test facility. Perfect symmetry was assumed in the CATHENA idealization and therefore venting through either inlet or outlet feeders should be determined randomly.

Channel Upper FES Temperatures. In Figure 9, the CATHENA calculation of the FES temperatures for the upper pin near the inlet and outlet ends of the channel is compared with the experimental results. The timing of the arrival of subcooled liquid at these locations in the simulation, indicated by the decrease in FES temperature, is in good agreement with experiment. The phenomenon of bottom pin temperature decrease, attributable to single-phase thermosiphoning in the channel, was again observed.

Feeder Temperatures. Figures 11 and 12 show the fluid temperatures measured in the horizontal feeder pipes near the inlet and outlet end-fitting simulators, respectively. The calculated liquid, vapour and end-fitting outer shell temperatures at the respective ends of the channel are also shown in Figures 11 and 12. The effect of the single-phase natural circulation between the channel and the end-fitting simulator is clearly seen as the early fluid temperature increase measured at both inlet and outlet feeders.

In summary, the CATHENA simulation of the "standing start" test CWIT-939 shows good agreement with experimental observation.

FIGURE 6: INLET FEEDER FLUID TEMPERATURE IN CWIT-937

FIGURE 7: OUTLET FEEDER FLUID TEMPERATURE IN CWIT-937

FIGURE 8: OUTLET FEEDER VOID FRACTION IN CWIT-939

FIGURE 9: CHANNEL UPPER PIN TEMPERATURES IN CWIT-939

FIGURE 11: CHANNEL BOTTOM PIN TEMPERATURES IN CWIT-939

FIGURE 10: CHANNEL BOTTOM PIN TEMPERATURES IN CWIT-939
CONCLUSIONS

It is concluded that CATHENA is able to model accurately the "standing start" behaviour in a horizontal CANDU-type channel. In particular, channel FES temperatures and venting times are predicted very well in comparison to those obtained in the experiments. Small differences between the experiments and the simulation results are believed due to a period of single-phase natural circulation at the beginning of the experiment, and axial conduction between the end-fitting simulators and the feeders. These phenomena were not modelled in the CATHENA simulations.

ACKNOWLEDGEMENT

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REFERENCES


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REPAIR OF END SHIELD OF PRESSURISED HEAVY WATER REACTOR

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ABSTRACT

Rajasthan Atomic Power Station (RAPS) has two units of 220 MWe Pressurised Heavy Water type reactors. Each reactor has 306 fuel channel assemblies which are designed to permit their replacement. In the first unit of RAPS, during a reactor shut down, it was observed that cooling water (light water) was leaking from one of the end shields. The lattice locations which were the sites of leakage were identified and the concerned channels, three in number, were removed. Inspection and non-destructive examination indicated that the leakage was occurring from the inaccessible side of the end shield. Remote handling tools were developed for removal of the coolant channels and lattice tubes. Repair of leak was carried out by mechanical gasketing arrangement using indium metal as the sealing gasket material. This paper briefly describes the activities leading to successful repair of the component.

DESIGN DESCRIPTION OF REACTOR COMPONENTS

The reactor consists of a calandria which is a large horizontal cylindrical vessel containing heavy water moderator having end shields on both the sides. 306 fuel channel assemblies placed in a square lattice traverse through the calandria and project through the end shield. Each end shield is a horizontal cylindrical vessel closed by flat tube sheets at the ends. 306 lattice tubes arranged in a square lattice are welded to both the tube sheets and provide the penetrations for the 306 fuel channel assemblies. A massive steel slab assembly is shrink-fitted into the shell of the end shield and provides shielding. Figure 1 shows the general arrangement of the calandria and end shields. The end shield is cooled by circulating water which is fed through multiple inlets. Suitable baffles are provided so that all the lattice tubes, tube sheets and bores in slab assembly are properly cooled.

Each fuel channel which penetrates through the calandria and end shields, is provided with independent inlet and outlet connections. Feeder pipes which connect the fuel channels to the reactor headers are routed through the space available between the end fittings of channels in front of the face of the end shield.

INSPECTION AND MACHINING OF LATTICE TUBES

Leakage of cooling water from the south end shield was detected in September 1981. Extensive test program was conducted to identify the source of leakage. Water holding and helium leak tests indicated that the leakage was occurring from lattice location F-10. This observation was confirmed by Acoustic Emission Technique. At this stage it was decided to remove the coolant channel at F-10. Channel cutting procedure, tooling and crew were qualified on full scale mock-up and using this procedure the channel was removed. The details of channel removal are discussed in Ref.1. Water leakage test conducted after removal of channel indicated that the leakage was taking place from the bearing sleeve bore area at the inaccessible calandria side of lattice tube. Suitable remote operating manipulators were developed for carrying out ultrasonic examination of this area. The ultrasonic examination carried out prior to removal of bearing sleeve indicated the presence of a defect in the groove of retaining ring. The bearing sleeve of hard tool steel having diameter larger than tube bore could not be removed by any conventional method. A technique using spark erosion was developed to remove the bearing sleeve. After removal of bearing sleeve, this portion of lattice tube was observed using boroscope. A longitudinal crack was observed in lattice tube near the calandria end along with a circumferential branch at the retaining ring groove. These observations were confirmed by ultrasonic examination. Ultrasonic examination of weld region indicated presence of a defect in the calandria side tube sheet starting from the observed defect in the lattice tube. It was thus necessary to remove the lattice tube so that further examinations could be carried out.

The space restrictions due to primary feeder pipes present in front of face of end shield made removal of lattice tube difficult. The middle portion of the lattice tube was cut in three longitudinal segments using spark erosion method so that individual segments could be easily taken out. Small stubs of lattice tube near tube sheets were removed by machining out the weld joint. Ultrasonic testing and dye penetrant examination were carried out on the tube sheet zone surrounding the lattice bore F-10. These confirmed the presence of earlier detected crack at 11 O’clock position of the lattice bore. An additional defect was also detected at 9 O’clock position of tube sheet. It was noticed that these defects were through thickness type and extended up to adjacent lattice locations at E-10 and F-9. This necessitated the removal of fuel channels from these locations. Qualified procedure established for removal of F-10 channel was again used for removing these channels. Dye check after removal of the channels indicated that the defects in tube sheet had propa-
and thus would arrest further propagation. These removal of lattice tubes from these positions and thus would arrest further propagation. These lattice tubes were also removed using spark erosion and machining. The examination carried out after removal of the lattice tubes confirmed the existence of two through thickness defects detected earlier. The locations and orientations of the defects are shown in Figure 2. It was also established that apart from these two defects, no other defect existed in the tube sheet region surrounding these lattice tubes.

The elevation of the lattice locations where the work was carried out was about 5 metres above the floor of the vault. A suitable platform was erected in front of the face of end shield. All the machining operations and inspection were carried out from a distance of about 3 metres. Access available was limited and viewing was restricted because of stream of radiation through the lattice bores. Fig.3 shows radiation streams in the Fuelling Machine Vaults.

DETECT DIAGNOSIS

Review of design and operating regime was carried out so as to identify the causes of failure. The probable causes were local stress relieving of component carried out prior to installations, radiation embrittlement of material and additional loading imposed on the component due to arrest of axial creep of coolant channel.

The manufacturing of the component was carried out in 1966. The design conformed to standards prevalent at that time. The fabrication and inspection procedures were also in line with the practices followed at that time. The raw materials of tube sheets, lattice tubes and weld metal were not subjected to ultrasonic examination as it is done now. Helium leak test carried out after final machining had revealed leakage from calandria side tube sheet at lattice location E-9. This area had been subjected to local weld repair and local stress relieving. The surrounding area of the tube sheet was subjected to pre-heating. After the local stress relieving, it was noticed that the surrounding tube sheet bores had distorted. Remachining had to be carried out on 60 bores which had become oval. This indicated the presence of residual stresses in this zone. Estimation of the stresses was difficult due to intricacy of geometry.

The material of calandria side tube sheet and lattice tubes is 3% nickel steel. Initial Ductility Temperature of this material is \(-101^\circ C\). Radiation embrittlement data available at the time of the design of the component indicated that this would rise to 32°C and thus minimum operating temperature was kept as 65°C. Subsequently, published data indicated that the rise in MDT would be larger than the estimated figure. Extensive work had been undertaken involving irradiation of impact test specimens to various levels of neutron fluence. Testing of these specimens had indicated that the end shield material was nearing saturation level of radiation embrittlement. However, the machining operations carried out at low cutting speeds had resulted in regular chip formation which meant that material was ductile at low strain rates. During the operating history of component, no impact load was ever applied to the component.

Detailed stress analysis was carried out for all the operating loads acting on the component. It was found that the stress level for normal operating loads was very low. Introduction of additional loads arising due to arrest of axial creep of coolant channels increased the value of stresses, however, the increased value of stress was well below the allowable value. But it was found that though the additional stress due to creep arrest load could not cause any failure, it could cause propagation of any defect if present.

It was thus thought that a defect which had escaped detection at the time of fabrication in 1966, under the action of residual stresses and operating stresses had propagated due to progressive radiation damage.

LEAK PLUGGING BY MECHANICAL GASKETING

Considering the embrittled state of calandria side tube sheet material of end shield, it was clear that any sealing method involving welding could not be used, and thus it was decided to resort to mechanical gasketing for plugging the leak. In order to minimise the applied loads, it was decided to use soft metals such as pure indium or pure lead as gasket materials. Development trials indicated that pure indium which is softer than lead could be used successfully. The sealing arrangement consists of plugs with indium metal deposited on face of flange and part of outside diameter near the flange. This indium is fully contained in the annular cavity formed by tube sheet bore, plug and a cap which can move axially along the plug. When the clamp bolts are tightened, the cap is pulled back so that the indium in the annular space is squeezed. This results in sealing of face and bore of the tube sheet. The ligament cracks are covered by strap assemblies. Suitable slots are machined in the plug sealing F-9 and E-10 lattice bores so that the strap assemblies are housed in these plugs during the installation of plugs. The plug sealing F-10 bore is machined with matching slots to receive the straps from F-9 and E-10. After installation of plugs, the straps are moved along the face of tube sheet so that the ligament cracks are fully covered. When the straps are tightened the indium contained in them seals the ligament cracks. The holes in sealing plugs carrying clamp bolts are sealed by providing suitable pure lead washers. The development of this design was carried out on a full scale mock-up vessel which simulated the tube sheet bores and ligament cracks. Hydromechanical tests were carried out for arriving at workable design. The sealing arrangement was later on subjected to cold water circulation and thermal cycling successfully. The bores of fuelling machine tube sheet were sealed by similar sealing plugs with pure silver gaskets. The details of sealing plugs for calandria side and fuelling machine side tube sheets are shown in Figure 4.
Remote handling tools were developed for insertion of the sealing plugs through the narrow space between primary coolant feeders in front of face of end shield. As the width of this passage was less than the diameter of the plugs the plugs were inserted in a turned position and rotated back to proper position after clearing the feeder pipes. The movement of straps and tightening of hardware were also done remotely from a distance of about three metres. The sealing plug assemblies, installation toolings and working crew were qualified on a full scale mock-up at site. After installation of all the plugs, the end shield was subjected to static pressure test, water circulation and three thermal cycles. After successful completion of these tests, the reactor has now been recommissioned and its operation is now satisfactory.

CONCLUSION:

The achievements can be summed up as follows:

(i) Without off-loading the core and disturbing the primary coolant feeders, the repair of the end shield was carried out.

(ii) NDT techniques which were used for detecting cracks were totally successful.

(iii) Design of sealing plugs is such that they do not endanger the tube sheet of the end shield. Moreover, they can be installed or removed easily without disturbing any other reactor components.

REFERENCE

Figure 2
Locations of Cracks in Calandria Side Tube Sheet

Section X-X

Diagram showing the locations of cracks in the Calandria side tube sheet, with annotations for sections and references to specific parts of the structure.
RADIATION FIELDS IN F/M VAULT

SOUTH FUELLING MACHINE VAULT

GENERAL BACKGROUND GAMMA FIELD 150-250 R/M
INSIDE THE CHANNEL AT SPECIFIED DISTANCE FROM 'E'FACE
915 FROM 'E'FACE 5SR/M
1750 FROM 'E'FACE 100R/M
1770 FROM 'E'FACE 150R/M

END FITTING

PRIMARY COOLANT FEEDER PIPES

TEMPORARY PLATFORM SOUTHERN FUELLING MACHINE VAULT

FIGURE 3
RAPS-1 END SHIELD SEALING ARRANGEMENT

FIGURE 4
REMOTE INSPECTION AND REPAIR
OF CANADIAN REACTOR VAULT COMPONENTS

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ABSTRACT

Specific maintenance problems have dictated the development of unique remotely operated equipment and procedures. Such equipment has been successfully used in inaccessible radioactive environments, namely the dry and flooded Calandria Vaults of the Pickering Nuclear Generating Station (PNGS).

INTRODUCTION

The installed capacity of PNGS represents 39% of the current total installed nuclear capacity of Ontario Hydro. Specifically, the four units of PNGS-A have a net capacity of 2060 MWe and the four units of PNGS-B have a net capacity of 2064 MWe, with the total installed nuclear capacity currently at 10584 MWe (net).

While the capacity of PNGS-A and PNGS-B are similar, certain design differences exist. One such difference is the design of the Calandria Vault. In PNGS-A, the Calandria vessel is suspended within the air filled vault above a dump tank. In PNGS-B, the Calandria is suspended in a water filled, steel lined, concrete vault, with no dump tank.

The unique maintenance problems which have occurred within these vaults have the potential to significantly impair the continued successful operation of the station, if allowed to persist. To initially identify and subsequently address the issues, remotely operated equipment, techniques and materials were developed, tested, and used.

PNGS-A

The Calandria Vault design of PNGS-A (Figure 1) is approximately 12.2 meters wide x 6.1 meters deep x 16.75 meters high. Access to the vaults is served by six 250 mm inspection ports located in the south wall of the vault, and three 100 mm inspection ports providing vertical access through the vault roof. Fields in the vault are of the order of 150 to 500 rads during shutdown conditions.

FIGURE 1: CALANDRIA VAULT

Visual Inspections

Visual examination of the vaults was initially performed by simply attaching a camera and light to a rigid arm. Recognition of the inadequacies of this equipment led to the development and use of an articulated arm, complete with lighting and zoom lens camera. This arm dubbed "BARNIE" (Bent Arm Remote Nuclear Inspection Equipment) has 5 degrees of freedom and reaches approximately 6 meters inside the vault.
The elbow and wrist joints are hydraulically operated and capable of 135° and 90° rotation respectively. These motions, combined with the 180° rotational capability of the forearm, provide the dexterity to perform a good general visual inspection of the vault internal structure.

Inspections were performed in all four units using both vertical and horizontal access ports. Initial results showed that corrosion had occurred on carbon steel components within the vault. Of immediate concern was segments of the biological shield cooling pipework, which is exposed within the vault, and moderator inlet line support brackets.

Biological Shield Cooling Pipework

The calandria vault walls (known as the biological shield), are cooled by water in pipework embedded in the concrete. On the inside of the vault, 40 individual segments of pipe emerge from and re-enter the walls, exposing between 1 and 2 meters of pipe per segment. After approximately 10 years of operation, leaks developed in this exposed pipework and have, over the last 3 years, increased in number and magnitude. It is believed that radiolysis of the vault atmosphere allows nitrogen to combine with any moisture present and form nitric acid. This, in turn, attacks carbon steel vault components. Once a leak has started, therefore, the presence of additional moisture within the vault tends to exacerbate the situation and promote further leaks. If the problem is allowed to remain unchecked, then adequate cooling to the concrete will be threatened. Loss of cooling would cause concrete degradation with a subsequent loss of biological shield integrity. This cannot be tolerated.

It was essential therefore to quickly implement corrective action to arrest the leaks and minimize the moisture in the vault. Such action would allow for a long-term solution to be carefully planned and executed.

The immediate action adopted was to inject specialized radiation proof sealants into the pipework. The sealant used has the capability of sealing defects up to 5 mm equivalent round hole diameter, enduring system pressure, temperature, and radiation conditions and poses minimal risk of blocking the pipe or associated valves etc, on the circuit under treatment.

The treatment of each defective circuit typically extended over a four day period. Initially, helium was injected into the circuit and the rate at which the pressure decayed was used to calculate an equivalent round hole diameter, since the precise shape of the actual defect is unknown. This equivalent diameter was then used to:

1. Select the appropriate sealant material and application pressure.
2. Act as a yardstick to determine the progress, if multiple applications are required.

Helium pressure was then maintained from the top of the circuit and water incrementally introduced at the bottom of the circuit. When the water level covered the defect, the helium pressure remained constant. The water head at this position then indicated the location of the defect and the quantity of sealant required for the operation.
The sealant was then mixed to specific conditions and introduced into the circuit. A pre-determined over pressure was applied for a specific time period in order to force the sealant through the defect. Following draining, the circuit was carefully flushed using procedures designed to ensure all residual sealant is removed from the circuit without disturbing the plugged or partially plugged defect. After a 24- to 48-hour air curing period, the circuit was again tested for defect size using a helium pressure decay rate. If a reduced defect was found, then the application was repeated. It was found that defects larger than 0.8 mm equivalent diameter usually required more than one treatment.

Prior to work on the reactor, a mock-up was constructed to simulate the Pickering conditions and to demonstrate the effectiveness of the sealants. To date, 14 separate leaks have been sealed.

**Moderator Inlet Line Hangers**

The moderator inlet line is a 254 mm, (10-inch) Schedule 10 stainless steel pipe which transfers the heavy water moderator between the moderator pumps and the calandria vessel. The line is supported by hangers HR29 and HR30 attached to stainless steel lugs welded to the base of the calandria.

Following visual evidence that HR29 had failed it was determined that the unsupported pipe would cause undesirable stress levels on a downstream section of pipe.

Accordingly work was initiated to develop tooling to replace the failed hanger utilizing the 25 mm ports. Two new arms were designed and constructed for this purpose and the BARNIE camera arm was used to provide an overview of the operation.

The arrangement of the remote tooling was such that three of the 25 mm access ports were utilized. Inserted through one port was a delivery track system within a cut away tube. This track carried manipulator arms to which a variety of end effectors was attached. The equipment provided the capability to inspect, cut, and remove the hanger and attaching bolts.
Inserted through a second port, was a hydraulically operated arm, dubbed ROBER1 (Remotely Operated Bent-arm Examination and Retrieval Tool). This arm has five degrees of freedom and was designed to align the end effectors and to hold the old and new hangers.

The replacement stainless steel hanger was sized to the removed hanger in order to provide the correct support tension. In a loosened condition the new hanger was installed and, once in position adjusted to the pre-determined height.

These procedures were developed on full scale mock-ups and work was successfully completed on all four of the PNGS-A units.

Biological Shield Cooling Pipe Removal

Having satisfied the immediate concerns associated with the Biological Shield Cooling and the Moderator Inlet Line Hangers, plans were initiated to address the long-term implications. As a starting point, it was decided to remove a section of biological shield pipework to confirm the corrosion mechanism and attempt to assess the remaining life. A line which could be isolated without detrimental effects to the concrete and one which was also seen to be leaking, was selected for removal.

Using the principles developed for removal and replacement of the moderator inlet line hangers, two new arms, a delivery system and a manipulator arm dubbed "ANDREA" (Articulated Nuclear Device for Remote Extraction Applications) were developed. It was also decided to utilize a separate umbilical feed system to ensure that all supply cables and hoses were clear of the working area.

The delivery system consisted of a main support tube with cut-away section and integral fixed track designed to carry the delivery trolley. The system was utilized for end effector delivery/retrieval, pipe section withdrawal from the vault, and provided a platform for overview cameras and lighting.

The ANDREA arm had six degrees of freedom and contained two cameras for close-up observation of the work site. This arm had the capability of reaching over and picking up end effectors from the delivery system.

The umbilical delivery system consisted of a support tube with an inner round glide. The system provided a route for the end effector and camera umbilicals and minimized congestion in the vicinity of the subject pipe. Two segments of pipe, each approximately 300 mm long, were removed by making three cuts with a 2 mm wide grinding wheel. The stub ends of pipe were then wire brushed and environmental plugs were installed to protect the pipe inside diameter.

As with previous activities, the entire operation was thoroughly rehearsed on a full scale mock-up with simulations of as many anticipated difficulties as possible.

PNGS-B

The PNGS-B vault design is such that the calandria is suspended in a steel lined shield tank approximately 10.1 meters wide x 5.4 meters deep x 13.4 meters high. Access is provided by a hatch approximately 1 meter square located in the roof of the vault. The tank is filled with demineralized water.
In 1985 it was found that the moderator heavy water charge in Unit 6 was becoming downgraded. Tests quickly proved that the source of the downgrading was the light water contained within the shield tank.

To identify the leak site, the moderator heavy water was drained and helium introduced into the moderator system. Ultimately, helium was detected at the top of the shield tank indicating that a leak path of helium had been established. To locate the source of helium bubbles coming from the moderator system into the shield tank, a miniature submarine, dubbed SABRINA (Submersible Apparatus Bldt for Remote Inspection in Nuclear Applications) was used. This 58 cm long unit is fully maneuverable in water and carries forward and aft cameras. The unit is tethered to a control unit via a neutrally buoyant umbilical cable.

The leak was found to occur at a weld between a moderator inlet line and the shield tank penetration. Since the leak site was accessible from outside the vault, repairs were performed by conventional methods. The initiating event, vibration of the moderator line as a result of its support geometry and flow characteristics, was addressed to prevent a recurrence in this and other units.
FUTURE WORK

Work is currently in progress in three areas:

- To develop more sophisticated inspection tooling with six degrees of freedom and a 15 meter reach.

- To develop repair procedures for the PNGS-B Unit 8 shield tank leak using an adhesive patch and the miniature submarine.

- To develop repair strategies for the replacement/repair of other vault components should the need arise.

FIGURE 12: ARRANGEMENT OF MODERATOR INLET LINE PENETRATION

Shield Tank Liner Leak

In Unit 8, water was found emerging from the concrete wall in the East Fuelling Machine vault. To determine the exact leak site, helium was injected through the concrete wall and liner, and the miniature submarine again put into service to search for bubbles.

The shield tank was field erected and it was through defects in the field welds that the submarine discovered helium bubbles.

Three distinct leak sites were identified located close to each other in an area where three liner plates are welded together.

Since access to the tank is confined to the one 1 mm square hatch located some distance away from the leak site, repairs could not be readily conducted. Development of concepts to implement repairs is currently in progress.

FIGURE 13: HELIUM BUBBLES AT SHIELD TANK LINER LEAK - VIEW FROM SABRINA CAMERAS
ABSTRACT

During routine emergency coolant injection system (ECI) testing a control panel operating error resulted in a partial reactor core bypass flow of the primary heat transport coolant, through the odd-numbered H2O injection valves. A reactor trip was averted by prompt operator action, but a turbine generator trip on high boiler level did occur. The unit was successfully recovered from this upset and returned to high power operation. The event was reviewed in some detail, as similar upsets have previously occurred. It was determined that a number of factors contributed to the event, principally in the areas of control panel ergonomics, the test procedure that was used, control room staffing, and the scheduling of safety system testing. A number of corrective measures which have been put in place are discussed, and it is anticipated that these will minimize the risk of re-occurrence of this event in the future. The findings from this investigation are generally applicable to all main control room control panel testing operations, and merit review for their applicability at other facilities.

INTRODUCTION

Pickering NGS-B is the second phase of the eight unit nuclear generating facility located on the shore of Lake Ontario about 30 kilometres from downtown Toronto. The lead unit (Unit 5) achieved first power in 1982, with the last unit (Unit 8) being declared in-service early in 1986. All the units are fitted with high pressure emergency coolant injection (HPCI) as part of the post loss of coolant accident emergency core cooling function.

A supply of light water is stored in a high level tank which provides the necessary suction conditions for the high pressure injection pumps. These pumps feed a common station header which is fed to each unit via a parallel pair of H2O injection valves, which in turn feed a total of sixteen H2O injection valves, arranged in eight parallel pairs of valves, each pair supplying one of the eight reactor headers in the primary heat transport system (two inlet and two outlet headers in each of the two main coolant loops).

Routine testing of these valves is carried out to demonstrate, via an ongoing process, that the reliability claimed in the Safety Report is substantiated. Four similar test procedures are used for this purpose, each one testing the valves in one of the four quadrants of the heat transport system. These tests are conducted at a four weekly test interval, and each contains about forty five discrete operations. A further test procedure is also used to test the two H2O injection valves, which is again conducted on a four weekly test interval, and which contains about seventy one discrete operations.
During the operating life of the station, these tests have led to simultaneous opening of two or more B20 injection valves on five different occasions. The tests have been performed about 1200 times so far in the life of the station, so that these five incidents represent a fault rate of 0.3%. A review of the five failures indicates that two were directly attributable to operator error, while the other three were determined to be the result of continuity problems with a selector handswitch, which has since been replaced on all units with a more reliable type.

**TEST DESCRIPTION**

The purpose of the B20 injection valve test is to confirm the logic associated with the valves and to test switch the valves themselves. To permit this to occur there are a number of associated valves in the system which allow testing without destroying the interface between light and heavy water systems. The ECI logic is a four channel system with three out of four channels required for initiation. Testing is further complicated by having conditioning logic which only permits initiation if one of a number of qualifying conditions is satisfied. These factors lead to a lengthy and repetitive test procedure, involving frequent selection of pushbuttons with very similar nomenclatures, for example 63335-PHB1, L,M,S and 63335-PHB2, L,M,S. This is on a panel which is relatively compact, has very few distinctive features and that is mounted at about waist height and approximately 15 degrees from the vertical. Figure 1 depicts the layout.

**EVENT DESCRIPTION**

The Second Operator (SO) assigned to the unit was performing routine test E-18, which is the test for the unit B20 injection valves. The procedure initially tests all the combinations of the three out of four tripping logic, and then, with the logic tripped, also tripe the conditioning logic to test activate the injection valve. This sequence is then repeated for the other injection valve. As previously mentioned, this test involves about seventy one discrete pushbutton or handswitch operations on the main control room panel, many of which are repeated operations on the same few push-buttons. These push-buttons are contained in one of two matrices which provide facilities for testing other component parts of the ECI system, such that an operating error within either group can affect more than the components immediately under test.

At step 53 of the procedure, the SO was required to initiate the logic for the B20 injection valve by operating three push-buttons labelled 63335-PHB1, L,M,S, being the first three buttons in the sixth row down. The operation he actually performed was to operate the three push-buttons labelled 63335-PHB3, L,M,S, which were one row before the correct push buttons, that is the seventh row down. The result of this error was that instead of using the logic to drive open one of the two B20 injection valves, he actually armed the logic to drive open all the odd-numbered B20 injection valves. Consequently, when step 57 was reached and the conditioning logic was tripped, thereby initiating valve operation, the wrong valves started to open.

A review of the simplified schematic (see figure 2) shows that the opening of the odd-numbered valves resulted in a short-circuit path between reactor inlet and outlet headers on each quadrant, thereby diverting primary heat transport coolant flow from passing through the core.

The authorised first operator (FO) who was seated at the unit control desk monitoring unit conditions immediately noted the incorrect valve operation and initiated corrective action by selecting the individual valve control hand switches to the manual CLOSE position and requesting the SO to reset the logic. The disturbance to the flow and temperature distributions in the heat transport system caused by the short circuit affected the steam rates of the individual steam generators, leading to fresh water level transients, and finally a turbine generator trip on high steam generator level, which is initiated when both very high level alarms are received on any one steam generator.

![Figure 2 SIMPLIFIED SCHEMATIC OF ONE HEAT TRANSPORT QUADRANT](image-url)
INVESTIGATION FINDINGS

During the discussions that took place as part of the detailed event review, a number of facts were established. Firstly, the SO performing the test stated that in his opinion no special factors contributed to the event taking place. The error occurred at about 05:00h but he stated that he did not feel particularly tired, nor did he have any problems with shift work generally. He was unable to offer any explanation of the error.

Secondly, the AFO in charge of the unit stated that he was satisfied with the qualification and ability of the SO, and that although he had only performed this particular test once before, under close supervision, he had routinely performed many other tests and it was normal to assign this type of work to him.

Thirdly, the details of the test facilities and their layout on the main control room panels were reviewed, and it was concluded that although it should be possible to conduct the test without making errors, the panel layout did nothing to discourage such mistakes.

Finally, the test procedure was reviewed to ascertain whether there were deficiencies in the procedure, or whether there were other changes that could be made to minimize the risk of a repeat occurrence. It was determined as a result of this review that there were confirming indication lamps elsewhere on the control panel which would assure the tester that he had pushed the correct sequence of buttons.

DISCUSSION

Procedural modifications focused on eliminating a particular error will often result in a procedure that ultimately becomes so cumbersome as to be error inducing. Technical Section review of all injection valve test procedures determined that there was a change that would prevent spurious opening of the D20 injection valves under any circumstances. This, simply, was to place the hand switches for all D20 injection valves, other than the one being tested, on STOP, so that the valves would not drive open under any conditions.

This change has reactor safety implications should genuine injection be required during the course of a test, but these concerns were reviewed considering the time at risk, and the administrative recovery procedures that were put in place, and were found to be acceptable, so this change has been implemented. In the case of the H20 injection valves, the potential for a similar elegant solution did not exist so the test procedure has been modified as suggested during the inquiry, that is, to include in the procedure a confirming check of a set of status indicating lights which are located elsewhere on the panel.

The shift crews had previously raised concerns about the uneven testing workload in the main control room, and a review had commenced with the intent of levelling the load across the shifts. This event provided a further impetus to that review. The previous pattern of testing scheduled the same test on the same shift for each of the four units, which resulted in a very inequitable testing workload as it did not take account of the varying requirements of the individual tests, and the schedule also called for more testing during the night shifts than the day shifts (see figure 3).

There has been no particular pattern of control room errors occurring during night shifts, but with the increasing evidence relating to the physiological effects of shift work, it seemed prudent to reconsider the pattern of testing. The schedule was redesigned with input from the first operator family, has been in place for some months now, and appears to be working well (see figure 4). It will be noted that there has been a shift in the distribution from nights to days as well as more evenly throughout the week.
The staffing of the main control room consists of six AFO's, but these numbers have not yet been achieved. For relief purposes therefore, the unit SO's are qualified as Supervised Control Panel Operators (SCPO) to provide a purely monitoring function under the direct supervision of the AFO from an adjacent unit, with the SCPO's also being used to complete some of the testing workload and the direct supervision of the unit AFO. A superficial review of significant event reports generated for control room operating errors in general would indicate that SO's have been involved in more errors than AFO's, but when the numbers of panel operations performed in testing is compared against the number of panel operations carried out on a unit at full power during the course of a twelve hour shift, there are no clear differences in performance between the two groups. That the testing was being performed by a SO is not therefore considered to have any bearing on the event. Recent moves to specify that more testing will be done by AFO's have resulted from a need to get the SO's into the field in their role as supervisors, and is not a result of this event. It is interesting to note that in the opinion of the operators involved, a reactor trip would have ensued from the transient if the AFO had been doing the testing himself, as he would not have been able so rapidly to identify the error and respond to the event.

There are very limited opportunities to change the characteristics of the control room, as the station was designed and built during the transition from traditional analogue control systems and information display to integrated digital process control as used in current designs. The outcome of this is that the main control room panels are crowded by having both full analogue control capability as well as the SCPO's and other features that are part of computer based systems. A further contributor to the problem is the original design intent of building a replica station of Pickering NSS A but with several added major systems, e.g. Shutdown System #2 and BORIC, which have had to be fitted within the constraints of the original overall panel layout. Two options have been proposed however, one of which is being actively pursued. A transparent plastic overlay has been developed for each of the SCPO tests that uses the test push button matrix, with cutouts that only permit access to the push buttons required by the particular test. Prototypes have been produced, but evaluation of the concept has been delayed by some minor practical difficulties with securing the overlay in place. Present indications are that the technique will prove to be successful. The other option is to apply colour coding to the push button matrix, but a decision on this approach has been deferred pending outcome of the other trials, as it is a less positive technique in reducing error rates.

**CONCLUSIONS**

Main control room panel operating errors can have significant consequences for continued unit operation, whether the errors occur during testing or other routine panel operations. Operating staff should be encouraged to report all errors whether or not they result in unit upsets, to assist support staff in defining and seeking solutions to the aspects of operation that are causing difficulty. The event discussed in this paper provides a good illustration of some of the factors that can influence errors at the man-machine interface, but it is crucial to ensure that any corrective actions will ensure a definite defence against a re-occurrence and are not just change for the sake of change, and also that procedural barriers that are erected do not themselves increase the risk of error, by compromising the simplicity and clarity of what is already in place. As a final note, reliability of station operation would be increased via reduced control room operating error rates if panel designers and operating staff can work together before finalizing control lay-outs, to ensure that many of the traps and problem areas are identified and dealt with before the design is finalized.
ABSTRACT

Mercury-Wetted Relays (MWRs) are used extensively in the implementation of control logic for safety and safety-related systems in CANDU nuclear power plants. These relays are used in a range of applications including the two-out-of-three logic for the shutdown systems (SDS1, SDS2), the emergency coolant injection system (ECIS) and the containment system. In some cases, these relays are also used to directly initiate end device functioning such as shut-off rod release or poison injection valve opening. Features such as speed of operation, high current carrying capacity, high reliability for multiple operations and compactness have led to the selection of MWRs in the above applications. This paper reviews Ontario Hydro’s operating experience, relay failure mechanisms, the issue of obsolescence for MWRs, and the actions taken to improve the reliability of the systems utilizing MWRs.

INTRODUCTION

Relay Design

The main sub-components of a MWR are a glass capsule housing the relay contact mechanism, partially submerged in a pool of mercury and a coil to initiate the switching action. The glass capsules, surrounded by the coil, is placed in a metallic container which is filled with a sealant (e.g., wax). Figures 1 and 2 show the construction of typical 5 ampere (5A) and 2 ampere (2A) rated glass capsules. The switching action provided by the two capsules is very similar, but the design of the two capsules is different.

The 5A capsule consists of an armature mounted on a spring and fixed to the seal-off tube. In the initial design, the spring was made of stainless steel. The use of tin as a dopant in this design would have prevented the proper wetting of the stainless steel by the mercury. Therefore, the capsules were not doped. The newer capsule design incorporates a spring made of nickel-iron which is compatible with tin doping.

The 2A capsule does not incorporate a spring in the design of the armature. An external permanent magnet provides the restoring force for the armature. In this type of capsule design, tin doping has been used for approximately 15 to 20 years.

Relay Operation

The following briefly describes the operation of the relay. Energization of the coil produces a
magnetic flux, a portion of which flows through the armature. The physical geometry of the capsule causes the flux to concentrate at the normally open (NO) contact enabling the armature to move and achieve circuit continuity. De-energization of the coil restores the armature to the normally closed (NC) position either by the restoring force of the spring or by use of an external permanent magnet.

Current is conducted by the mercury and not necessarily by the metallic elements of the capsule. To minimize contact resistance, wetting of the metallic elements of the armature by the mercury must be ensured. This is achieved by use of micro-grooves in the armature surface and by proper choice of materials and dopants. The dopants (copper and tin) are chosen for their ability to promote the proper mercury wetting action and for other properties which improve circuit performance.

The flow of mercury to the contact point of the armature is promoted via capillary action and by the pumping action resulting from the armature movement.

Relay Applications

Extensive use of MWRs was first made by the telephone industry. The high reliability exhibited in applications requiring multiple switching operations, low power consumption and the ability to carry large currents ensured their continued use by the telephone industry. Ontario Hydro adopted MWRs for use in the two out of three control logic of reactor safety systems and safety support systems and as interface devices between the control logic and safety equipment (shut-off rods, solenoid valves, etc.). The capability to carry relatively large currents, the compactness, speed of operation and the excellent operating experience (billions of operations with a relatively low failure rate) in the telephone industry were reasons for selecting these devices. All of the Ontario Hydro nuclear generating stations use MWRs for specific applications.

To maintain diversity in certain applications, different models of relays, including relays from different manufacturers, have been used in redundant system configurations. However, the models used in the above applications are no longer available. Where spares are required, either pin compatible relays from an alternate manufacturer are purchased, if available, or a re-design using currently available relay models must be undertaken.

BACKGROUND

Abnormal failure trends for MWRs were first identified by the telephone industry (Reference 1). In Ontario Hydro, MWR failure trends were first noticed at Bruce A. Subsequently, problems were experienced at Bruce B and more recently at the Pickering plants.

Investigations conducted at the Bell Laboratories (Reference 1), revealed that contacts were sticking closed due to the formation of metallic compounds (NiHg4). These inter-metallic compounds were found in the mercury contained between the armature and fixed contact, providing a conductive bridge which prevented opening of the contact. Operating data indicated that it required four to six years of operation before relay failures resulted in circuit malfunction. The study concluded that by adding tin as a dopant to the mercury, formation of the inter-metallic compounds would be inhibited - extending the life of the relays. Accelerated aging tests confirmed the results. The tests indicated that after a simulated in-service life of approximately four years about 25% of the untaped capsules had failed; whereas, the doped capsules had no failures.

At Ontario Hydro, problems with MWRs were first noted in 1982. A review of Significant Event Reports (SERs) revealed an increase in MWR failures (contact stuck closed) at Bruce A. A study was initiated at that time and the findings documented. It was concluded that the MWR failure rate had been increasing since 1980 and that a higher failure rate was being experienced with a specific MWR model (Model J - Table 1). It was recommended that the relay type with the abnormal failure rate be investigated to determine the cause. Such an investigation was undertaken and the findings indicated that nickel contamination of the mercury in the capsule was the most likely cause of the contact bridging problem. In addition, failures of the contact arc suppression networks had occurred and these were known to cause premature failures of the contacts. It was recommended that in future only MWRs with tin doping should be purchased and used.

During commissioning of the Bruce A High Pressure ECI, a number of MWRs were found to be faulty. The study, conducted at that time, found evidence of faulty operation specific to one relay model. Analysis of the failed relays indicated that failures had resulted from cracked capsules. Cracking of the glass capsules was traced to rough handling. Recommendations were made to improve the packaging used for shipping the relays and also that all the relays be tested prior to being installed in an operational circuit.

More recently, an increase in MWR failures has been experienced at Pickering B. The increased failure rate was first noticed with one type of MWR (Model A - Table 1). A number of in-service failures were observed, in addition to those discovered in spare stock inventory. Beginning in November of 1988, an increase in failures was also noticed with another type of MWR (Model B - Table 1). All of these recent failures have been characterized by contacts sticking closed on de-energization of the relay (generally in the safety system applications the relays are kept energized so that the "fail-safe" condition can be assured on loss of electrical control power). The failures experienced have been with the older (pre-1984) non-tin-doped MWRs. Only recently have the newer tin-doped varieties been installed as replacements.

Tests performed on the failed relays (from Pickering) both at the station and by the relay manufacturer have confirmed the existence of the stuck contacts. The test results have also indicated that a nickel-mercury amalgam was present in the capsule. This was noted by the existence of long bridging times during relay operation. One lot of failed relays (21) from Pickering was examined by the manufacturer. The results of the analysis showed a total of 26 out of 84 capsules with timing peculiarities. All of the relays were manufactured during the 1979 to 1980 time period. These results correlate with Reference 1, which showed that 25% of the population would fail after 4 to 6 years of service.

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### TABLE 1

**MWR FAILURE DATA**

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<td>---</td>
<td></td>
</tr>
<tr>
<td>E</td>
<td>PNGS A</td>
<td>12</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS A</td>
<td>48</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>F</td>
<td>PNGS A</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PNGS B</td>
<td>4272</td>
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<tr>
<td></td>
<td>BNNGS A</td>
<td>1475</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS B</td>
<td>1976</td>
<td></td>
</tr>
<tr>
<td>G</td>
<td>PNGS A</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS A</td>
<td>109</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>H</td>
<td>PNGS A</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS A</td>
<td>27</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>J</td>
<td>PNGS A</td>
<td>252</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PNGS B</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS A</td>
<td>10.4</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BNNGS B</td>
<td>---</td>
<td></td>
</tr>
</tbody>
</table>

---

7-20 CNS 10th ANNUAL CONFERENCE, 1989
A summary of the failure data collected by Ontario Hydro is given in Table 1.

The data are tabulated as a percentage of the installed quantities. This provides a more consistent basis for assessing the relative performance of the various relay models. It should be noted that the failure data recorded is relatively sparse. The data does not provide a complete history of the failure rate, due to the way it has been collected. What has been tabulated are those failures which have led to a safety system impairment. Those failures not leading to a safety system impairment, have not necessarily been reported by SEPs and thus could not be tabulated.

An examination of the data (Table 1) reveals the following:

1. The relay (Model J) exhibits an increase in total failures after four to six years of operation, as predicted by Reference 1 for non-doped relays.

2. The relay (Model A) shows a similar trend as that for Model J.

3. The relay (Model F) shows a consistently lower failure rate, with no apparent indication of a tendency for the failure rate to increase.

Due to the sparsity of the data, it cannot be proven that the increase in failure rates is due to the lack of tin-doping. Research conducted by others, the available failure data, (Table 1) and analysis of failed relays supports such a conclusion.

**STATUS**

The current increase in failures, at Ontario Hydro, is depleting the present stock of spare relays. In addition, the limited shelf life of these relays causes the depletion rate to become even greater. Many of the models, originally used by Ontario Hydro, are no longer available, due to obsolescence. Where pin-compatible substitutes are available, the sources are limited to a single manufacturer. These are generally custom made, as the quantities purchased are limited, and are relatively costly. Where suitable pin-compatible alternatives are not available, modifications must be made in order to adapt available relays to the existing applications. Table 2 provides a list of those relays used by Hydro that have become obsolete.

**TABLE 2**

<table>
<thead>
<tr>
<th>Relay Model</th>
<th>Replacement</th>
<th>Station</th>
</tr>
</thead>
<tbody>
<tr>
<td>HG-1041</td>
<td>JG1-122-11</td>
<td>Bruce A</td>
</tr>
<tr>
<td>HG4A-1007</td>
<td>JM4-114-41</td>
<td>Pickering A, B, Bruce A</td>
</tr>
<tr>
<td>HG4B-1005</td>
<td>JM4-114-42</td>
<td>Bruce A, B, Pickering B</td>
</tr>
<tr>
<td>HG5M-5121</td>
<td></td>
<td>Bruce A, B, Pickering B</td>
</tr>
<tr>
<td>HG5-5011</td>
<td>HG2M-5111</td>
<td>Bruce B</td>
</tr>
<tr>
<td>HG5-5069</td>
<td>HG2M-5111</td>
<td>Bruce B</td>
</tr>
<tr>
<td>HG3A-1008</td>
<td>JM3-112-31</td>
<td>Pickering A, Bruce B</td>
</tr>
<tr>
<td>HG2A-1015</td>
<td>JM2-112-22</td>
<td>Pickering B</td>
</tr>
<tr>
<td>HG6F-1005</td>
<td>HGR2M-5111</td>
<td>Bruce A</td>
</tr>
</tbody>
</table>

The present relay replacement program involves replacing the MWRs by tin-doped equivalents, as they fail. Due to the limited number of failures in the past, substitution by tin-doped equivalents has not advanced sufficiently at the stations to prove the validity of this solution. Therefore, there is an inadequate failure database to establish whether the tin-doped relays are sufficiently reliable, and to determine whether they exhibit the same failure mode as the non-tin-doped variety. Due to the limited availability of pin-compatible substitutes and the lack of multiple sources for those that are available, Ontario Hydro has initiated a study to resolve this situation. The study will:

1. Identify those relays which can be used as substitutes,
2. Identify the MWR characteristics which are essential for the application,
3. Evaluate the modifications required for installation,
4. Assess the reliability of these substitutes,
5. Verify that use of these substitutes does not adversely affect the diversity of equipment required in the design of redundant safety systems,
6. Ensure that there are multiple sources for the relays identified.

This study is underway and some progress has been made. The MWR characteristics essential to the various applications have been identified and documented. Contact has been made with various relay manufacturers and samples of possible substitutes are under engineering evaluation. In the next few months, it is anticipated that a list of potential substitute relays and a qualitative summary of the circuit modifications required will be established. After completion of this phase of the work, a detailed review of the applications will be conducted in order to identify any circuit modifications required and the costs of making such
modifications. When appropriate substitute relays have been identified, samples will be evaluated by Ontario Hydro Research Laboratories for long term reliability performance.

CONCLUSIONS

The most pressing problems at present, with the MWR applications are:

(1) the obsolescence of many of the relays used,
(2) the higher than expected failure rate of the installed relays,
(3) the lack of diverse suppliers for replacement relays.

Obsolescence, coupled with the higher failure rates experienced with the non-tin-doped versions of the relays, has led to more rapid depletion of the available spares and a decrease in the diversity of equipment installed in redundant safety systems. The low volume of replacement quantities ordered has resulted in small batch production runs which has escalated the cost of the devices and also caused delivery delays. Therefore, it is imperative that suitable replacement relays be found.

The higher failure rates currently experienced appear to be characteristic of MWRs which are not tin-doped. Research done by other organisations indicates that these types of relays display a more rapid aging characteristic, which is aggravated by prolonged energization. The majority of relays in CANDU safety systems are normally energized which leads to higher operating temperatures for the relays and a shorter operating life.

Industry research and experience has shown that MWRs that are tin-doped have a longer design life and exhibit a lower failure rate. The limited failure data available in Ontario Hydro tends to support this conclusion. The failure data available from Pickering B shows that the failure rate for the tin doped relay model is approximately an order of magnitude lower than that for the MWRs utilizing the non-tin-doped capsules. Experience with the relays utilizing the new tin-doped 5A capsules is limited. Large scale substitution has only been recently initiated (within the last two years) and thus insufficient data is available to assess the long term performance of these relays.

The possibility exists for finding electro-mechanical relays which can be substituted for the MWRs, but this approach will present its own unique problems. Typically, relay size and pin compatibility will be the major problems.

REFERENCES

TEN YEAR PERFORMANCE OF BRUCE NUCLEAR GENERATING STATION "A"
ON-POWER FUELLING SYSTEM

S. JAYABARATHIAN

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Tiverton, Ontario NOG 2T0

ABSTRACT

This Performance Report covers a ten-year operating period of the Bruce 'A' On-Power Fuelling Systems from 1978 to 1988, dividing into early and mature operation phases and indicating in each, the problem areas, incapability figures and improvements done to raise the fuelling capability. Also it highlights the impact of the ever-increasing demand of the CANDU pressure tube In-Service Inspection Programs on the fuelling capability in mature years.

INTRODUCTION

Bruce Nuclear Generating Station 'A' is the first multi-unit CANDU Station in Ontario Hydro to have the shared system of on-power fuelling machines, designed by G.K. Canada, based on the experience of NPD and KANUPP fuelling systems.

On-power fuelling means that the nuclear fuel is replaced in the reactor channel, while the unit is producing power. This capability has four major advantages and benefits:

- Enables the unit to have a higher capacity factor, typically 5% to 10% better in lifetime values than units with off-power fuelling, which lowers the total unit energy cost
- Permits major unit outage scheduling, independent of annual fuelling
- Yields higher average fuel burnup and, therefore, lower fuelling cost.
- Allows on-power removal of defective fuel and, thereby, reduces radioactive contamination and man-rem dose.

The Station has been provided with triplicated fuelling trolleys which enable automatic refuelling of all four reactors. Each reactor, under normal operating conditions, can be refuelled by at least two out of three fuelling systems. Each fuelling system consists of two fuelling machine heads, suspensions, heavy water and air auxiliary systems, all mounted on a transport trolley. The transport trolleys travel on two pair of rails in the containment fuelling duct, between the fuel handling service facilities and each reactor. One trolley travels on the North pair of rails, while two trolleys travel on and share the South pair of rails. The selected fuelling machine head suspension remotely couples to the carriage on each reactor bridge before fuelling operations begin on a selected channel.

The fuel channels are normally refuelled by adding new fuel at the downstream end of a channel, in multiples of two, even though fuelling with one bundle at a time is possible. Fuelling normally takes place with the reactor at power and at normal operating temperature and pressure. Fuelling off-power or at any low power and pressure is possible. The fuelling machines are controlled remotely from consoles located in the main Control Room of the Station. New fuel is supplied to the fuelling machine heads and irradiated fuel is discharged from the heads at Central Service Area through ports onto storage trays in the Primary Fuel Storage Bay.

The Fuel Handling System is designed to meet the refuelling requirements of the four reactor units, including routine inspection and maintenance of the equipment and operations, other than refuelling, on a two-shift basis, seven days a week. The anticipated fuelling rate for the four units is approximately 70 fuel bundles per day at 100% reactor power.
This Performance Report covers a ten-year period of operation from 1978 to 1988 which is divided into two phases. In the initial phase of four years, only two fuelling trolleys (North and South) were refuelling all the four reactors and in the second phase of six years, the third extension trolley also was in service.

EAKLY OPERATION (1978 - 1982)

General Performance of Fuelling Trolleys

Only North and South trolleys were refuelling the reactors in the initial phase and all four units were not in operation until 1979. Full power was restricted to only 88% in the units due to NRC licencing limitation. During this period, there were several equipment failures, forced unit outages and fuelling time of channel took over 7 hours, which caused concern in the fuelling requirements.

Bruce Operations and Central Nuclear Services (CNS) initiated two programs in 1978 to collect system performance data and to assess Fuel Handling System unavailability. This first part is mainly an extract of a CNS Report produced in May 1981. These programs were discontinued at the end of 1980.

High extremity doses were another area of concern during this period for the Fuel Handling Maintenance personnel who used to work on the F/H head. Activation products from the Heat Transport System and in some cases, fission products from spent fuel got trapped in the machine and they were the main source for the high extremity doses. The GE-designed Bruce 'A' Fuel Handling System did not have Dg0 Flow Injection System as the AECL-designed fuelling machines to prevent ingress of activated debris from the reactor channel.

F/H EXTREMITY DOSKS
(Man-Rem)

<table>
<thead>
<tr>
<th>YEAR</th>
<th>F/H</th>
<th>MAX</th>
<th>STATION TOTAL</th>
<th>TOTAL</th>
<th>%</th>
</tr>
</thead>
<tbody>
<tr>
<td>1978</td>
<td>-</td>
<td>74</td>
<td>401</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>1979</td>
<td>-</td>
<td>40</td>
<td>841</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>
| 1980 | 1335| 79  | 1590          | 84%   | 84%
| 1981 | 858 | 21  | 1364          | 63%   | 63%
| 1982 | 632 | 45  | 962           | 66%   | 66%

System Performance Monitoring Programs

System Performance data were collected through the following two programs.

Fuel Handling Evaluation and Monitoring (FHEM).

Current operating status of the System was recorded by the Fuel Handling operators on provided forms with an accuracy of 1 hour.

The subsystem and component causing unavailability or malfunction was also indicated. The information was processed by Central Nuclear Service (CNS) and distributed in monthly reports.

Performance Analysis Program (Log Tapes)

Each individual operation was recorded in F/H computer. The data was processed to give the actual time and performance factor for each operating sequence as well as a breakdown of delays. The program was discontinued at the end of 1980.
Following were some of the results of the Performance Evaluation:

**Forced Unit Outages**: Maximum of 2 events in some units per year, causing unit incapability of the order 5.4% (max) in one event and station incapability due to FHS 8.1% (max) of total.

**Equipment Unavailability Factor**: Varied from 20% to 52%.

**F/H System Unavailability**: Decreased from 52% to 40%.

**Channel Fuelling Time**: Fuelling time came done from 7 hr/channel (1978) to 4 hr/channel (1980).

**Forced Maintenance**: Decreased from 1.58 hr/channel to 1.29 hr/channel.

**Scheduled Maintenance**: Reduced from 0.87 hr/channel to 0.19 hr/channel.

**Maintenance Ratio (M.R)**

\[
M.R = \frac{\text{Total Maintenance Time}}{\text{Fuelling Time}}
\]

SUMMARY OF STATION INCAPABILITY

<table>
<thead>
<tr>
<th></th>
<th>1978(*)</th>
<th>1979(**)</th>
<th>1980(**)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Station Incapability Factor (%)</td>
<td>24.7</td>
<td>22.8</td>
<td>12.1</td>
</tr>
<tr>
<td>Portion of Incapability Due to FHS (%)</td>
<td>2.3</td>
<td>6.9</td>
<td>8.1</td>
</tr>
<tr>
<td>Contribution to Incapability Factor by FHS (%)</td>
<td>0.54</td>
<td>1.58</td>
<td>0.98</td>
</tr>
</tbody>
</table>

* 3 units in operation
** 4 units in operation

**Major Equipment That Caused Fuelling Incapability**

**Charge Tube/Ram Assembly Problems**

- Galling and binding of the ram spline was one of the major problems. It was suspected that ram bowing, due to heat stratification within F/M head, resulted in excessive stresses applied to the spline.

- Sintered spline balls which were disintegrating were replaced with the cast balls of larger diameter. The new balls performed satisfactorily.

- Charge tube rotary bushing wore out prematurely and needed to be replaced every 6 to 9 months.

**Input Drive Failures**

Several failures of input drives occurred when coupling between a gear and the shaft was lost. This was due to the failure of a fastener (tab washer, locknut, key). Some of these incidents prevented reinstallation of the closure plug in channel end fitting after fuelling. Costly unit outages were required to recover the incapacitated F/M head. The keyed coupling between the shaft and the gear were changed to more reliable splined connection.
Snout Seal Ring Life

The service life of snout seal rings was only about 50 clamping operations. Frequent exchanging consumed considerable amount of maintenance work as well as significant dose uptake to the personnel. There was also a problem with the supply, because the rings were all non-standard.

D$_2$O Hoses Failures

The average service life of the hoses had been one to one and a half years, with some hoses bursting after only a few months of service. It was determined that the material of the hose liner was not suitable for the relatively high operating temperature.

D$_2$O Auxiliary System

Circulating Pumps Failure

Early experience indicated that failures of the pumps were caused by the stator-can fatigue cracking. As a result, D$_2$O leaked into the stator cavity and eventually, when the system was depressurized, caused the stator to collapse and seize the rotor. Stator heat conductive dielectric oil was replaced with epoxy filling to eliminate possibility of oil leaking into the D$_2$O System. However, the epoxy resin made pump overhaul more difficult.

Shortening of the stator windings, due to the moisture, was another cause of the pump failures.

The later failures appeared to be due to the rear bearing wear induced by the crud in the D$_2$O. The wear caused the rotor to rub on the stator-can and consequently, can peeling which seized the rotor.

Major Contributors to FHS Unavailability

<table>
<thead>
<tr>
<th>Specific Equipment</th>
<th>1978 North South</th>
<th>1979 North South</th>
<th>1980 North South</th>
</tr>
</thead>
<tbody>
<tr>
<td>D$_2$O Aux Pumps</td>
<td>4% 6%</td>
<td>8% 7%</td>
<td>7% 5%</td>
</tr>
<tr>
<td>Charge Tube/Ram Assembly</td>
<td>13% 9%</td>
<td>6% 7%</td>
<td>- 4%</td>
</tr>
<tr>
<td>Power Track</td>
<td>- -</td>
<td>7% 8%</td>
<td>6% 25%</td>
</tr>
<tr>
<td>D$_2$O Hoses</td>
<td>5% 5%</td>
<td>- -</td>
<td>9% 8%</td>
</tr>
<tr>
<td>Input Drive Seal</td>
<td>- -</td>
<td>12% 9%</td>
<td>- -</td>
</tr>
<tr>
<td>Closure Plugs (Debris)</td>
<td>- -</td>
<td>- 10%</td>
<td>9% -</td>
</tr>
<tr>
<td>Air Aux 4% Compressors</td>
<td>- 6%</td>
<td>- -</td>
<td>-</td>
</tr>
<tr>
<td>Trolley Coarse Drive</td>
<td>- -</td>
<td>- -</td>
<td>8% -</td>
</tr>
</tbody>
</table>

Power Track Damages

In the early stage, some power track mechanical components were wearing out prematurely. The wear, combined with an inadequate guiding of the power track, caused an interference of the South power track with a support column. As a consequence, the power track derailed and was completely destroyed. Costly deratings were avoided because the third trolley power track was available at the site.

Twisting of power track power cables causes breaking of the cable armouring. Attachment of the cables to power track was modified and more pliable power cables are used now. These changes improved the situation but did not eliminate the problem entirely.

Reactor Area Bridge Components

There were several failures of bridge drive train components (coupling, universal joint, mitre gearbox). These relatively simple failures were very costly because the bridges are not accessible while on power and unit shutdown is necessary to perform maintenance. A thorough review of F/M bridges by GE was carried out to eliminate potential breakdown causes.

Major Modifications to Improve Fuelling Performance

Fuelling Machine Third Trolley

Bruce NGS 'A' Fuel Handling Task Group, in its final report in 1979, concluded that the predicted long-term fuelling capability would not meet the predicted demand for fuelling and recommended that a third F/M trolley be added to the existing Fuelling System.

The third trolley was installed in the South fuelling duct on the same pair of rails. The duct was extended in the East end to park the third trolley and to be used as an additional maintenance facility. The trolley was declared in service in the first quarter of 1983.

Sixteen Bundle Magazine

The new 16 bundle magazine was retrofitted into the existing F/M heads and replaced the original 12 bundle magazines. It improved the fuelling capability by increasing the number of channels fuelled per each visit to Central Service Area (CSA).
D$_2$O Flow Injection System

Design initiatives were taken in the year 1980 to fit this sub-system in F/H so as to minimize radiation doses to operating and maintenance personnel. Following are the benefits of this new addition to the system:

- Reduces man-rem expenditure
- Lower thermal distortion in snout, charge tube and ram
- Avoids unit forced outage due to leaky closure plug because of channel debris
- Reduces maintenance on circulating pumps, charge tube, ram and magazine and also filter change frequency.

<table>
<thead>
<tr>
<th>F/H Personnel Radiation Doses (1980) (Man-Rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>F/H Personnel</td>
</tr>
<tr>
<td>Operators (43)</td>
</tr>
<tr>
<td>Mechanical Maintainers (22)</td>
</tr>
<tr>
<td>Control Maintainers (12)</td>
</tr>
</tbody>
</table>

Station Extremity 1590 Man-Rem
F/H Extremity 1335 Man-Rem, 84% of Station Dose

CONCLUDING STATEMENTS FOR EARLY OPERATION

- Fuel Handling System performance has been steadily improving since the beginning of operation in 1977. The fuelling demand in the period 1978 to 1982 was limited and the System was able to meet it.
- All production losses incurred due to the Fuelling System in the years 1978 - 1982 resulted from fuelling incidents when unit shutdowns were required to rectify the problems. Most of these incidents were due to component failures and some were caused by debris from the Primary Heat Transport System.
- In the initial phase of operation, Station incapability factor varied from 15.0% to 22.0% and the portion of incapability due to Fuel Handling System ranged from 2.3% to 8.1%. Some of the contributors to Fuel Handling unavailability were D$_2$O circulating pumps, D$_2$O elastomer hoses, charge tube - ram assembly, bridge brakes and power track components.
- Three areas were identified in the Fuel Handling System which needed significant improvements for full power unit operation in the near future:
  1) Reliability of F/H equipment
  2) Increase in fuelling capacity
  3) Minimize radiation dose to F/H personnel.

MATURE OPERATION (1983 - 1988)

General Performance of Fuelling Trolleys

So far, all the three trolleys have refuelled over 45,600 reactor channels and handled over 229,000 fuel bundles, since the beginning, through the four units, the highest magnitude of on-power fuelling in Ontario Hydro's CANDU nuclear power plants. The third F/M trolley was commissioned in the year 1982 and was in service since March, 1983.

During this phase of operation, the severely contaminated F/M heads were all decontaminated and made available for regular fuelling operation. All the 12 bundle magazines were replaced with 16 bundle types in all the F/M heads and this increased the number of channels fuelled per fuelling trip. Each fuelling trip took 6.5 hours now instead of 4 hours. The charge tube rotary bush material was changed in heads. The D$_2$O circulating pump can rotors were modified with proper test procedures and their service life has been prolonged from premature failures to continuous operation over 9000 hours. Their failure rates were brought down from 1.5 to 0.2 failures per pumps in service per year. The old D$_2$O hoses were replaced with new types which served longer than the old ones (service life 2 to 3 years). D$_2$O filter change frequency decreased from 100 channels to 500 channels fuelled, as the ingress of the Heat Transport System debris got reduced eventually. The F/M head service between overhauls extended, on the whole, to over 4000 channels of fuelling. The flimsy power track frames were all replaced with heavy ones in all the three trolleys.

<table>
<thead>
<tr>
<th>Trolley</th>
<th>Total Channels Fuelled From the Beginning</th>
</tr>
</thead>
<tbody>
<tr>
<td>North</td>
<td>23,280</td>
</tr>
<tr>
<td>South</td>
<td>17,540</td>
</tr>
<tr>
<td>Extension</td>
<td>4,780*</td>
</tr>
</tbody>
</table>

*Extension trolley started fuelling since March, 1983
The operating factors of each trolley and channels fuelled in each year from 1983 to 1988 have been tabulated for review of performance. The performance of the South Fuelling Systems has been indicated separately as they share the same Central Service Area facilities (new fuel loading, spent fuel discharge, etc) and travel over the same pair of mechanical tracks.

### CHANNELS FUELLED AND OPERATION FACTORS

#### NORTH SYSTEM

<table>
<thead>
<tr>
<th>YEAR</th>
<th>NORTH TROLLEY CHANNELS FUELLED</th>
<th>NORTH TROLLEY OPERATING FACTOR</th>
</tr>
</thead>
<tbody>
<tr>
<td>1983</td>
<td>2486</td>
<td>89.6%</td>
</tr>
<tr>
<td>1984</td>
<td>2538</td>
<td>84.7%</td>
</tr>
<tr>
<td>1985</td>
<td>2162</td>
<td>81.8%</td>
</tr>
<tr>
<td>1986</td>
<td>2259</td>
<td>78.6%</td>
</tr>
<tr>
<td>1987</td>
<td>1721</td>
<td>68.2%</td>
</tr>
<tr>
<td>1988</td>
<td>1766</td>
<td>76.5%</td>
</tr>
</tbody>
</table>

#### SOUTH SYSTEMS

<table>
<thead>
<tr>
<th>YEAR</th>
<th>SOUTH TROLLEY CHANNELS FUELLED</th>
<th>EXTENSION TROLLEY CHANNELS FUELLED</th>
<th>SOUTH TROLLEY OPERATING FACTOR</th>
<th>EXTENSION TROLLEY OPERATING FACTOR</th>
</tr>
</thead>
<tbody>
<tr>
<td>1983</td>
<td>1545</td>
<td>490</td>
<td>68.8%</td>
<td>43.9%</td>
</tr>
<tr>
<td>1984</td>
<td>1583</td>
<td>462</td>
<td>76.8%</td>
<td>44.5%</td>
</tr>
<tr>
<td>1985</td>
<td>1752</td>
<td>692</td>
<td>67.1%</td>
<td>35.3%</td>
</tr>
<tr>
<td>1986</td>
<td>1457</td>
<td>516</td>
<td>38.6%</td>
<td>57.0%</td>
</tr>
<tr>
<td>1987</td>
<td>1268</td>
<td>613</td>
<td>78.6%</td>
<td>39.5%</td>
</tr>
<tr>
<td>1988</td>
<td>729</td>
<td>1014</td>
<td>53.3%</td>
<td>61.5%</td>
</tr>
</tbody>
</table>

The combined operating factors of the South and Extension Trolleys average to 110% on the whole for six years, indicating only a slight increment of performance because of the sharing or interfering characteristics of the systems.

### SIX YEAR AVERAGE PERFORMANCE (1983 - 1988)

<table>
<thead>
<tr>
<th>TROLLEY</th>
<th>CHANNELS FUELLED</th>
<th>OPERATING FACTOR</th>
</tr>
</thead>
<tbody>
<tr>
<td>North</td>
<td>2155</td>
<td>80%</td>
</tr>
<tr>
<td>South &amp; Extension (Combined)</td>
<td>2025</td>
<td>110%</td>
</tr>
</tbody>
</table>

### CHANNELS FUELLED BY THE TWO SYSTEMS

<table>
<thead>
<tr>
<th>YEAR</th>
<th>NORTH TROLLEY CHANNELS FUELLED</th>
<th>SOUTH AND EXTENSION TROLLEYS CHANNELS FUELLED</th>
<th>TOTAL CHANNELS FUELLED</th>
</tr>
</thead>
<tbody>
<tr>
<td>1983</td>
<td>2486</td>
<td>2035</td>
<td>4521</td>
</tr>
<tr>
<td>1984</td>
<td>2538</td>
<td>2075</td>
<td>4613</td>
</tr>
<tr>
<td>1985</td>
<td>2162</td>
<td>2444</td>
<td>4606</td>
</tr>
<tr>
<td>1986*</td>
<td>2259</td>
<td>1973</td>
<td>4232</td>
</tr>
<tr>
<td>1987**</td>
<td>1721</td>
<td>1881</td>
<td>3602</td>
</tr>
<tr>
<td>1988**</td>
<td>1766</td>
<td>1743</td>
<td>3509</td>
</tr>
</tbody>
</table>

*South Trolley was severely contaminated due to Unit 2 Channel N6 incident and isolated for 3 months.

**Most of the time, one unit had a long planned outage and South or Extension Trolley was dedicated to non-fuelling service in that unit.

### FUEL HANDLING PERSONNEL DOSE EXPENDITURE (Man-Rm)

<table>
<thead>
<tr>
<th>YEAR</th>
<th>TOTAL CHANNELS FUELLED</th>
<th>WHOLE BODY DOSE</th>
<th>EXTERIMITY DOSE</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>F/H</td>
<td>% STN DOSE</td>
<td>F/H</td>
</tr>
<tr>
<td>1983</td>
<td>4521</td>
<td>140</td>
<td>33%</td>
</tr>
<tr>
<td>1984</td>
<td>4613</td>
<td>95</td>
<td>39%</td>
</tr>
<tr>
<td>1985</td>
<td>4606</td>
<td>75</td>
<td>25%</td>
</tr>
<tr>
<td>1986</td>
<td>4232</td>
<td>14</td>
<td>56%</td>
</tr>
<tr>
<td>1987</td>
<td>3602</td>
<td>69</td>
<td>13%</td>
</tr>
<tr>
<td>1988</td>
<td>3509</td>
<td>58</td>
<td>11%</td>
</tr>
</tbody>
</table>

The North System operating factor averages to 80% over the period of six years, the South being 63% and extension 47%. In the year 1986, the South Trolley was incapacitated for over three months because of Unit 2 Channel N6 incident, when the F/M heads were severely contaminated with spent fuel debris from the ruptured channel. During the years 1987 and 1988, at least one unit out of four, had a long planned outage for "West Shift" activities on the reactor channels.
During the upcoming SLAR (Spacer Location And Repositioning) program, one of the South Systems will be dedicated for a long duration, while the other two trolleys would be used to fuel three or four units, as the case may be.

SLAR program starts from September, 1989 and stretches to the end of 1992. This indicates that the non-fuelling service demand has been increasing heavily for reactor channel inspections, as more and more data collection is mandatory for the evaluation of pressure tube integrity, before the ultimate requirement of Large Scale Fuel Channel Replacement (LSFCR).

Station performance criteria and targets have been met in the fuel handling operation in all the years. Man-rem of fuel handling personnel has been gradually tapering down over the years to a steady level. Extremity dose level has been the highest in the Fuel Handling Maintenance group, even though it has been steadily reduced, with regular decontamination, prior to maintaining the equipment.

OPERATING FACTORS FOR FUELLING ARE AFFECTED BY:

- F/H System breakdown
- F/H scheduled maintenance
- Modification needs for improvements
- Access control problems
- Interference between South Systems
- Excess reactivity in units
- Non-fuelling service
- Containment leak or PRV testing
- Lack of qualified Operators
- Lack of spare parts
- Excess radiation field in F/M duct

F/H System Breakdown

- Significant events, causing fuelling incapability, forced unit outage or severe contamination in the Fueling Trolley. Two major events that caused fuelling incapability were:
  1) Unit 1, P13 event
  2) Unit 2, N6 incident.

Repetitive Problems

- Powertrack cable kinking and wire failure
- Snout seal ring problems
- Charge tube/ram problems
- D2O system circulating pump failures
- Catenary hose failures
- Brakes and clutch problems
- Leaky closure plug due to debris
- Input drive bearing/seal failures
- Gap sensing package problems
- Inverter problems
- Suspension latch mal-operation

Scheduled Maintenance

Our proportion of preventive to breakdown maintenance has increased significantly, and rightly, over the years. Routine maintenance still represents a much larger portion of unavailability than breakdown maintenance.

Modification Needs for Improvements or Regulatory Requirements

Major Items

- Powertrack frame reinforcements
- Creep modifications on F/M head and suspension
- D2O Flow Injection System
- F/M Trolley locks
- Full fuelling capability of South Systems
- F/M conversion and re-conversion for channel in-service inspection
- Significant Event Reports follow-up changes

Access Control Problems and Interference at CSA

- Difficulty in fuelling the end units (B1 or B4) while one inner unit (B2 or B3) has been shut down and maintenance in progress

- As South and Extension Trolleys were designed to share the south pair of rails and Central Service Facilities, they interfere with each other during operation and maintenance and work on one can cause unavailability of the other, especially work involving long periods in Central Service Area such as maintenance or major ancillary port work, such as CIGAR inspection.

Excess Reactivity in Units

Sometimes, Fuelling Trolley may be available but not used for fuelling as the units may have the permissible excess reactivity for that day (1.5 mk per unit per day).

Non-Fuelling Service

Over the years, more and more Fuel Handling System time has been devoted to non-Fuel Handling uses. PIPE, CIGAR, SCRAPE and other reactor maintenance projects all consume Fuel Handling time, effort and resources.

So far, 20 to 25% of the operating time has been utilized each year for In-service Inspection of pressure tubes, dedicating either South or Extension Trolley during a planned unit outage.

CHANNEL INSPECTION PROGRAMS

The specific objective of this In-service Inspection (ISI) program is to monitor the integrity of pressure tubes believed to be at higher than average risk.
due to known causes such as manufacturing defects, fuel channel assembly variances, commissioning-induced damage and fuelling or maintenance-induced damage. In addition, the IS1 program will generate engineering data on fuel channel performance for design and research applications.

The issues which are addressed by the IS1 to meet this objective include:

- The effect of pressure tube/calandria tube contact on high (H) equivalent tubes
- Manufacturing flaws
- Rolled joint flaws
- Commissioning and operation-induced flaws in pressure tubes like:
  a) Fretting
  b) Debris damage
  c) Pressure tube sagging, ovality, etc
  d) Mislocation of spacers in between pressure tube and calandria tube, due to vibrations
  e) Pressure tube elongation due to neutron irradiation (creep).

The various tools and techniques for those regular inspection programs are:

STEM

Scanning Tool for Elongation
Measurement - to check pressure tube creep. Reactor Area Bridge and carriage will be dedicated for this data collection.

CIGAR

Channel Inspection and Gauging in Active Reactor - to check pressure tube integrity. This tool can inspect most of the channels but not all the reactor channels and needs F/M Head, Trolley Reactor Area Bridge and Carriage (burnup loss as 13 fuel bundles per channel are discharged to Bay).

PIPE

Package Inspection Probe - to inspect only the rolled joint flaws between end fitting and pressure tube. This technique needs conversion and re-conversion periods on the F/M heads in addition to the inspection duration.

Wet Scraping Technique

This tooling takes a small scrape sample of the pressure tube to measure the deuterium ingress and involves one complete F/H System for inspection purpose.

SLAR

Spacer Location and Repositioning - This inspection equipment consists of two separate, highly-sophisticated and complex modules which are bolted to the F/M magazine flathead, instead of the regular ram/charge tube assembly. This project involves major modifications on F/M Head, Trolley and Control System, during conversion and re-conversion periods (about 9 months). SLAR duration is about 3 to 7 months per unit.

SLAR technique locates and re-positions the spacer in between pressure tube and calandria tube and does all inspections in an empty channel. It re-uses the same 13 fuel bundles in the channel (so no burnup loss). This equipment can inspect all reactor channels, unlike CIGAR.

SLAR Program

The following F/M Trolleys will be dedicated solely for SLAR Program, as shown, for long duration:

- Extension Trolley to SLAR - Sep 1989 to Sep 1991, Unit 3 and Unit 4
- South Trolley to SLAR - Oct 1991 to Dec 1992, Unit 2 or Unit 1

PROPOSAL TO IMPROVE FUELING PERFORMANCE
AND CHANNEL INSPECTION PROGRAMS

The next generation Fuelling Machine (F/M) ought to be Fueling and Inspecting Machine (FIM), with capabilities for both service, to cope with the increasing demand of pressure tube inspections, as pressure tube flaws are number 1 problems in CANDU power plants. Following are some of the suggestions to improve fuelling and inspecting performance of the systems.

1) Modify, replace or improve components to eliminate or minimize repetitive problems or obsolescence.

2) Provide a "West Service Area" like the existing "East Service Area" and modify the South and Extension Trolleys to fuel all the four units on automatic mode. Convert the ESA and WSA for full scale maintenance, like the CSA (Central Service Area).

3) Modify the CIGAR Module Tooling System to enable in-service inspection of all reactor channels.

4) Procure two separate F/M heads for In-service Inspection, conversion and re-conversion purpose like PIPE, SLAR, SLAP (Spacer Location at Power) or BLIP (Blister Inspection Probe).

5) In future CANDU multi-unit power plants, provide triple pair of rail tracks in F/M duct (one for each of the three F/M Trolleys) to eliminate interference in common facilities.
CONCLUDING STATEMENTS FOR MATURE OPERATION

Contribution to Station Incapability by Fuel Handling System - Bruce N.G.S.A

<table>
<thead>
<tr>
<th>Year</th>
<th>Station Incapability (%)</th>
<th>FHS Incapability (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1977</td>
<td>40</td>
<td>11.8</td>
</tr>
<tr>
<td>1979</td>
<td>30</td>
<td>11.0</td>
</tr>
<tr>
<td>1981</td>
<td>20</td>
<td>10.5</td>
</tr>
<tr>
<td>1983</td>
<td>10</td>
<td>5.0</td>
</tr>
<tr>
<td>1985</td>
<td>10</td>
<td>5.0</td>
</tr>
<tr>
<td>1987</td>
<td>0</td>
<td>0.5</td>
</tr>
</tbody>
</table>

F/H Dose Expenditure
W.B. and Extremity Dose

REFERENCES

ABOUT THE AUTHOR
Mr. S. Jayabarathan was graduated in the year 1956, as a Mechanical Engineer with a degree in Honours, from the University of Madras, India. He joined the Department of Atomic Energy, Bombay in 1957 and had one year training in Nuclear Science and Engineering and Power Plant Design and Operation. He operated the Nuclear Research Reactor, CIRUS at Bombay as Senior Shift Engineer in 1962 and came to Canada with the Indian Group in 1966 to Douglas Point NGS to get CANDU Power Plant operating experience. He then worked for twelve years in the Indian CANDU Power Plants, RAPP and MAPP as Maintenance and Technical Services Superintendent respectively. He has been in the mainstream of operation and maintenance of the CANDU on-power Fuelling Machines of Douglas Point NGS, RAPP, MAPP and Bruce N.G.S. 'A' for over twenty years.

He joined Ontario Hydro, NGD in 1982 and has been working as Technical Supervisor Fuel Handling at Bruce N.G.S. 'A' since 1983.
UN SIMULATEUR PLEINE ÉCHELLE DE CENTRALE NUCLÉAIRE : UN OUTIL COMPLEXE : CONDUISSANT À UNE MEILLEURE FORMATION

Claude Droin, Ing. et Hervé Belhomme, Ing.

Hydro-Québec
Centrale Nucléaire, Gentilly-2
Québec, Canada

INTRODUCTION

Un simulateur pleine échelle de centrale nucléaire utilisé pour la formation spécifique du personnel de conduite est un outil très sophistiqué et représente un investissement majeur. Cependant, nous retirons d'importants bénéfices de cet investissement en ayant une formation plus efficace. Aussi, le simulateur peut être utilisé à d'autres fins, telle la vérification de procédures d'exploitation.

Le besoin de former son personnel de conduite plus efficacement pour rencontrer les sévères critères d'opération sécuritaire et fiable, n'échappe pas à la centrale nucléaire Gentilly-2. C'est ce qui a conduit Hydro-Québec en 1984 de prendre à l'achat d'un simulateur. Dans cette présentation, nous couvrirons les principales étapes du projet de simulateur de formation de Gentilly-2. Basé sur notre expérience, nous insisterons sur les aspects suivants : la description technique du simulateur, l'ingénierie et la fabrication, le suivi du projet, l'évaluation, la mise en service, le site et le programme de formation du personnel d'entretien du simulateur.

Pour conclure, nous allons discuter de la stratégie d'utilisation du simulateur et des bénéfices que nous prévoyons en retirer.

L'exploitation sécuritaire et fiable d'une centrale nucléaire exige, entre autres, d'avoir un personnel de conduite très bien formé. Cette affirmation s'est renforcée après l'accident à la centrale nucléaire américaine Three Mile Island (CMI) où on a identifié clairement des lacunes dans la formation du personnel de salle de commande. L'emploi d'un simulateur spécifique de centrale nucléaire pour la formation des opérateurs a été introduit au début des années 70. Ce n'est que depuis l'incident de TMI que cet outil de formation a véritablement pris son essor. Aussi, nous avons vu naître des normes précis, tels la norme ANSI 3.5 et le guide d'utilisation d'un simulateur pour la formation de l'Institute of Nuclear Power Operators (INPO) 86-026. De plus, la capacité de reproduire très fidèlement le comportement d'une centrale nucléaire en situations normales et surtout anormales, s'est grandement améliorée dû, entre autre, à l'évolution rapide des équipements informatiques et des techniques de modélisation. C'est en fait incontestablement un outil très efficace dans la formation pratique des opérateurs de salle de commande.

La centrale nucléaire Gentilly-2 d'Hydro-Québec, située près de Trois-Rivières sur la rive sud du fleuve Saint-Laurent, est de type CANDU (Canadian Deuterium and Uranium). Elle est en opération commerciale depuis 1963 et possède une capacité de 665 Mw. La formation de son personnel autorisé de conduite pose, depuis le début, certaines difficultés. Un des moyens identifiés pour enrayer ces difficultés fut l'achat d'un simulateur pour la formation pratique des chefs de quart et opérateurs de salle de commande.

C'est ainsi que le conseil d'administration d'Hydro-Québec donna le feu vert au projet d'acquisition d'un simulateur spécifique le 19 décembre 1984.

HISTORIQUE

L'introduction d'un simulateur, pour la formation du personnel d'une centrale nucléaire, a débuté lentement à la fin des années 60 et au début des années 70. Ce sont des manufacturiers d'équipements des systèmes à vapeur qui ont mis à la disposition de leurs clients des centres de formation dotés de simulateurs. À cette époque, les propriétaires de centrales nucléaires n'étaient pas totalement convaincus des justifications de posséder leur propre simulateur. D'autant plus que son coût était très élevé et sa performance était moyenne. Avec l'évolution des équipements informatiques, la baisse de leur prix et l'amélioration des techniques de modélisation, nous avons assisté, vers les années 75, à une nette augmentation du nombre de simulateurs vendus surtout que le marché des centrales nucléaires en construction avait augmenté grandement. Un groupe de travail composé de l'agence de contrôle américaine NRC (Nuclear Regulatory Commission), des propriétaires de centrales nucléaires et des manufacturiers ont alors émis une norme ANSI 3.5 pour l'utilisation d'un simulateur destiné à la formation. Ce n'est qu'après l'accident à la centrale TMI que l'industrie des simulateurs a réellement pris son essor. Les commissions chargées de l'étude détaillée de l'accident ont pointé des erreurs humaines comme étant des causes et ont fait ressortir des lacunes importantes dans la
formation du personnel. Ce qui a conduit la Commission de contrôle américaine NRC à élever ses exigences dans la réalisation des examens de qualification des opérateurs de salle de commande, et a du même coup, exigé d'avoir de meilleurs programmes de formation.

Dans les années 80, il y a eu plus de 60 simulateurs construits dont l'une des caractéristiques était d'être d'une grande fidélité par rapport au comportement de la centrale de référence, et dont la portée de la simulation était très étendue. C'est ce que nous entendons par simulateur pleine échelle. Au Canada, Génital-2, fait figure de proue dans le domaine. Elle a été achetée, pour chaque groupe de ses centrales, un simulateur pleine échelle fabriqué par la compagnie CAN (Canadian Aviation Electronics Ltd) de Montréal.

À Génital-2, les discussions pour l'acquisition d'un simulateur ont débuté en 1981 avant même que la centrale devienne opérationnelle. Nous étions, à l'époque, à la phase mise en service intensive de la centrale. Nous voulions devoir toutes les opportunités pour que le personnel acquiert des habiletés et des connaissances essentielles à la prise en charge de l'exploitation de la centrale. Malgré cela la certification du personnel auprès de la Commission de contrôle de l'énergie atomique du Canada (CCAE) fut très difficile. L'expérience du personnel, et les moyens de formation limités, furent identifiés comme raisons majeures de nos échecs. Au cours de l'année 1983, une étude préliminaire pour acquérir un simulateur pleine échelle à Génital-2, a été réalisée. Il était évident, selon cette étude, que le simulateur aiderait grandement à la formation pratique et au reculage du personnel de conduite en plus de servir, à l'occasion, pour la vérification des procédures d'exploitation.

Hydro-Québec était consciente que la certification du personnel de conduite serait de plus en plus difficile et que si nous voulions être prêts à assurer la sûreté du personnel existant, il fallait prendre une décision dès maintenant concernant l'achat d'un simulateur. Nous avons estimé qu'il s'écoulait un peu moins de quatre (4) ans avant de pouvoir l'utiliser pour la formation. À la fin de 1984, la direction d'Hydro-Québec approuva l'acquisition d'un simulateur pleine échelle pour la centrale Génital-2. Le projet a démarré officiellement en janvier 1985.

DESCRIPTION DU SIMULATEUR

Le simulateur de Génital-2 est un simulateur pleine échelle spécifique à la centrale conçu pour former, de façon prioritaire, les chefs de quart et les opérateurs autorisés de salle de commande.

En se référant à la figure 1, nous retrouvons, comme simulateur, une réplique fidèle de tous les panneaux de la salle de commande à l'exception des panneaux de systemes de manutention du combustible. Tous les systèmes ayant une interaction avec les équipements de salle de commande, ont été modélisés ce qui représente un peu plus de 230 modules.

L'ensemble de l'équipement comprend deux ordinateurs de simulation de la compagnie Systems Engineering Laboratories Ltd (SEL), modèle SEL-32/6700 de 8 Mo (Mega-ops/sec) de mémoire vive à double processeurs, interconnectés entre eux par l'intermédiaire d'une architecture de réseau partagée de 1 Mo. Chaque ordinateur est branché à une unité de disque de 300 Mo. Une troisième unité de disque de 300 Mo est partagée par les deux ordinateurs. Le contrôle des systèmes simulés est assumé par deux ordinateurs VAX-11/730, branchés en parallèle, qui sont une réplique exacte des ordinateurs de la centrale au point de vue équipement que logiciel. Cependant, nous avons dû modifier le système d'exploitation de ces ordinateurs pour tenir compte du contexte de simulation.

L'avantage d'avoir les programmes de contrôle dans chacun des ordinateurs de contrôle plutôt que dans les ordinateurs de simulation permet de reproduire en temps réel le fonctionnement des simulateurs similaires à celui de la centrale. Aussi nous savons du temps de développement qui serait consacré à la réécriture des programmes de contrôle. Enfin, toute modification apportée à ces derniers en centrale, peut être très facilement implantée au simulateur. Par contre, cette configuration, avec deux types différents d'ordinateurs, présente comme inconvenient une architecture informatique plus complexe et difficile à exploiter et à entretenir.

L'interface entre les ordinateurs de simulation et les ordinateurs de contrôle, se fait à l'aide d'un programme d'interface en charge de servir, au besoin, pour la vérification des procédures de production.

Description du simulateur

Le simulateur de Gentle-2 est un simulateur pleine échelle spécifique à la centrale conçu pour former, de façon prioritaire, les chefs de quart et les opérateurs autorisés de salle de commande.

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d'incidents, tels une fuite de caloporteur (PKCA) ou encore un bris de conduite de vapeur avec beaucoup de rasssemblage selon les données d'analyse que nous utilisons pour faire la comparaison.

Les modèles sont organisés selon une structure par priorités. La majorité des modèles s'exécutent à chaque second de la mise en service, la charge totale du simulateur, en territo du pourcentage de sa capacité d'exécution, se situe à 95%. Nous sommes à la limite acceptable. Avec une telle charge, il est pratiquement impossible d'exécuter un programme tant qu'il n'est pas terminé lorsque la simulation est en cours. Pour mettre au point les programmes de simulation, le personnel d'entretien du simulateur, a accès à un troisième ordinateur identique aux deux ordinateurs de simulation. Pour le moment, cet ordinateur ne fait pas partie de la simulation bien qu'il soit branché à la même portée. Il nous sert aussi de pièce de rechange pour les ordinateurs de simulation.

ÉCHAUFFEUR ET COÛT

L'objectif, en terme d'échéancier, que s'était fixé Hydro-Québec, était de compléter la mise en service du simulateur et de le rendre disponible pour la formation en octobre 1988. En se référant à la figure 3, nous voyons que l'utilisation du simulateur pour la formation a débute en janvier 1989. Le retard de quatorze (14) semaines s'explique par le fait que nous avons dû ajouter dix (10) semaines à l'écuyrier du projet, pour la correction des déficiences. Aussi, les phases de mise en service et des essais d'acceptation à l'usine ont pris un peu plus de temps. Ce retard n'a pas eu de conséquences sur la formation avec simulateur, car le programme nucléaire intégré n'a débuté qu'en janvier 1989. De plus, cela nous a permis d'augmenter la performance du simulateur et de débuter ce premier programme de formation sans problèmes importants de comportement de modèles.

À l'aide d'ingénierie préliminaire, Hydro-Québec a fait appel aux services de Ontario-Hydro pour définir les spécifications techniques du simulateur. C'est à cette phase que nous avons octroyé à la compagnie CAE le contrat d'ingénierie pour qu'elle développe les principaux modèles de simulation et qu'elle fasse la conception et l'ingénierie complète du simulateur. Les plans et devis produits ont servi à la fabrication. Cette phase a duré seize (16) semaines de plus que prévu mais sans causer un réel impact sur l'échéancier global. La phase de fabrication a débuté avant la fin de cette étape.

La deuxième phase du projet est la fabrication du simulateur et la mise au point des modèles. Au cours de cette phase, le fabricant CAE a fait face à un sérieux problème de mise au point du lien de communication. Ceci a causé un retard de quatre (4) semaines pour le début de la phase d'intégration des modèles avec les panneaux de la salle de commande du simulateur. Pour reprendre le temps perdu et permettre de débuter les essais d'acceptation à l'usine, le personnel de CAE a dû travailler selon un horaire de 24 heures par jour et sept (7) jours par semaine.

Les essais d'acceptation à l'usine ont pris cinq (5) semaines de plus que prévu. Nous avons rencontré de sérieux problèmes de comportement de modèles des systèmes de sécurité de rejet de vapeur et de confinement du bâti réacteur. Ceci nous a obligé à reprendre les simulations plusieurs fois jusqu'à ce que l'on obtienne des résultats satisfaisants.

Le retard pris dans la phase précédente, a été en partie compensé par la réduction du temps requis pour l'installation. En effet, les équipes de CAE ont travaillé selon un horaire de 12 heures par jour. Après l'installation du simulateur à la centrale Gentilly-2, nous avons débuté la phase de corrections des déficiences. C'est cette étape qui, juge essentielle, a principalement causé le retard dans le projet.

Le coût global du projet s'est chiffré à plus de 25 millions de dollars. En se référant à la figure 3, nous voyons un écart favorable de 325 000$, entre les coûts estimés et les coûts réels. Cet écart s'explique principalement par une diminution des frais généraux et d'intérêts. Nous avons estimé les coûts des intérêts en assurant un écart mensuel des paiements parfois dans les temps. En réalité, nous avons déboursé les plus grosses sommes d'argent vers la fin du projet.

Le coût d'un simulateur est impressionnant mais, Hydro-Québec a surtout justifié l'approbation du projet pour des raisons de qualité de formation plutôt que des raisons strictement économiques.

ESSAIS D'ACCEPTATION À L'USINE

Cette phase du projet est très importante pour le client. C'est à ce moment qu'il vérifie l'intégrité et la performance du simulateur en comparant les résultats des simulations avec les spécifications énumérées au devis technique. C'est, en quelque sorte, un processus itératif de vérifications et de corrections et ceci permet au client de porter un jugement au sujet de l'acceptabilité du simulateur avant sa livraison.

Cette phase du projet comporte plusieurs étapes en commençant par la vérification systématique des équipements. Puis vient la vérification des logiciels de support et finalement la validation du comportement des modèles de simulation. Le déroulement de ces étapes se fait en conformité avec des procédures approuvées par le manufacturier et le client. Ces procédures sont très élaborées et toutes les déficiences découvertes lors des essais, sont enregistrées sur des formulaires prévus à cet effet. Le manufacturier a alors la responsabilité de régler ces déficiences et de soumettre la solution au client qui procédera à de nouveaux essais de vérification.

Dans le cas du simulateur de Gentilly-2, les essais en usine ont débuté par des essais...
préliminaires, effectués en collaboration avec le personnel du fabricant et les représentants d'Hydro-Québec. Nous avons exécuté durant les essais préliminaires une partie seulement des essais, le but étant de remplacer les plus représentatives de l'ensemble du simulateur. Nous avons tenté, alors, de déceler le maximum de déficiences afin qu'elles soient solutionnées avant le début des essais formels. À cet étape, nous avons déjà un bon aperçu de la performance de notre simulateur et nous avons aussi une bonne idée que nous pourrons déceler les essais formels d'exception sans risque de perdre de temps importante ou encore d'être arrêté pour solliciter des problèmes fondamentaux.

Comme première étape, nous avons procédé à une vérification visuelle complète de tous les équipements et du câblage. La comparaison des pannaux de la salle de commande simulée avec les photos récentes des pannaux de la centrale, a fait ressortir plusieurs erreurs d'Hydro-Québec. Plusieurs desssins d'arrangement général fournis au constructeur n'étaient pas à jour. Par ailleurs, nous avons effectué les essais fonctionnels de tous les équipements selon les procédures approuvées. À cet endroit, les ingénieurs des projets ont été vérifiés à l'aide de diagnostics qui sont fournis par les fabricants de ces appareils. Il faut être vigilant sur ce point car les systèmes d'exploitation des données changent souvent et les diagnostics fournis ne sont pas toujours compatibles. Posséder des diagnostics incomplets que ce soient les traçants que nous aurions utilisés serviront lors de la réalisation de notre programme d'entraînement. Nous avons utilisé quatre (4) semaines pour compléter cette étape tel que prévu. Nous avons rencontré aucun problème majeur à l'exception d'avoir à déceler les différentes versions des diagnostics des fabricants. Nous avons identifié 150 déficiences dont seulement 10 étaient jugées majeures.

En deuxième étape, nous avons vérifié les logiciels avant des fonctions se rapportant au système d'exploitation des données. En effet, nous avons constaté que certaines fonctions spécifiques de notre simulateur, comme la capacité pour les simulateurs de prendre des décisions en point de vue de la récupération de l'eau de condensation, et le calcul des performances du simulateur, ne sont pas indispensables. Nous avons alors éliminé la majorité de ces déficiences corrigées. Ceci réduisait, de plus, le risque d'invalider les déficiences corrigées. Nous avons également identifié plusieurs déficiences d'importance majeure, qui ont dû être corrigées. Nous avons ensuite mis en place des diagnostics supplémentaires pour les vérifier.

Finalement, la validation du comportement des essais de simulation est une étape des plus importantes car elle permet de vérifier plusieurs déficiences de manière réaliste. Nous avons ainsi identifié 150 déficiences dont 50 étaient d'importance majeure, 30 étaient de deuxième importance, et 70 étaient de troisième importance. Nous avons alors éliminé la majorité de ces déficiences corrigées. Ceci réduisait, de plus, le risque d'invalider les déficiences corrigées. Nous avons ensuite mis en place des diagnostics supplémentaires pour les vérifier.

En conclusion, nous avons identifié à cette étape plus de 1000 déficiences dont 50 étaient d'importance majeure. Les systèmes qui nous ont causé le plus de difficultés sont les systèmes de confinement du bâtiment réacteur et le système des générateurs de vapeur. Nous avons dû réécrire ce modèle au complet lors de la mise en service. Les systèmes électroniques et le système de gazolaire pour leur
part ont affiché une bonne performance.

Le lien de communication, pour interfacer les ordinateurs de simulation aux ordinateurs de contrôle, nous a amené plusieurs problèmes et causé du retard.

Nous avons terminé cette étape par trois (3) semaines consécutives de vérifications de déficiences de ce que nous avions permis d'autoriser la livraison du simulateur à Gentilly. Le nombre de déficiences restant à solutionner était de 225 au lieu de 100 tel que prévu initialement. Nous avons décidé d'accélérer la livraison du simulateur de façon à permettre à CAE de corriger ces déficiences et de conserver la dynamique du projet.

**INSTALLATION ET ESSAIS DE MISE EN SERVICE**

L'installation complète des équipements du simulateur était la responsabilité du fabricant CAE. Dû au retard accumulé dans les activités précédentes, ce dernier a redoublé d'efforts pour procéder au démontage des équipements à l'usine, réinstaller l'installation à la centrale Gentilly-2 et faire les essais fonctionnels de tous les équipements à l'intérieur de l'usine pendant une période de huit (8) semaines au lieu des douze (12) semaines initialement prévues.

Cette étape s'est déroulée sans problème majeur, mais à part celui des câbles d'interconnexion des ordinateurs de contrôle, le passage de ces câbles à l'intérieur des panneaux des ordinateurs, au lieu d'utiliser les étagères à câbles, a causé des problèmes d'emplacement de câbles bloquant l'accès à certains équipements. Nous avons décidé de tolérer la situation jusqu'à l'été 1989. À ce moment, nous corrigerons cette anomalie.

Les activités de mise en service ou les essais d'acceptation au site se sont déroulées d'une manière différente de celle prévue. En effet, nous avons préalablement utilisé une période de dix (10) semaines pour régler et vérifier les déficiences déjà identifiées lors des essais à l'usine. Il y avait encore à la fin des essais en usine un nombre de déficiences de 225 qui devait être toléré la situation jusqu'à l'été 1989. A ce moment, nous corrigerons cette anomalie.

Les activités de mise en service ou les essais d'acceptation au site se sont déroulées d'une manière différente de celle planifiée. En effet, nous avons préalablement utilisé une période de dix (10) semaines pour régler et vérifier les déficiences déjà identifiées lors des essais à l'usine. Il y avait encore à la fin des essais en usine un nombre de déficiences de 225 qui devait être toléré la situation jusqu'à l'été 1989. A ce moment, nous corrigerons cette anomalie.

**ORGANISATION DU GROUPE D'ENTRETIEN**

L'entretien, les modifications et la mise à jour d'un outil très spécialisé, requièrent du personnel très bien formé dans le domaine. Nous avons recruté à l'intérieur d'Hydro-Québec un groupe d'individus, dont la plupart travaillaient déjà à la centrale Gentilly-2, afin de se former auprès du fabricant CAE. En effet, il avait été prévu au contrat que du personnel d'Hydro-Québec soit intégré aux groupes de travail du projet Gentilly-2 et considéré comme tout autre employé du fabricant. Au début de la phase de fabrication, en février 1986, deux ingénieurs joignaient l'équipe de projet de CAE pour apprendre de spécialiste d'intégration et de modélisation. Huit mois plus tard, un ingénieur et deux techniciens, responsables des équipements, intégraient le groupe. À la même période, deux instructeurs débutaient leur stage de formation en ayant comme responsabilité la rédaction des procédures d'essais les plus difficiles. La formule de participation active au travail semblait offrir beaucoup d'avantages pour acquérir rapidement de connaissances techniques et de l'expérience. Mais à cause des contraintes de l'échec en serré et de l'intensité du travail demandé aux spécialistes de CAE, il a été difficile d'obtenir, de leur part, l'attention espérée pour aider notre personnel dans son apprentissage. Mise à part cette difficulté, nous pouvons dire que l'expérience a été très positive et que l'équipe a pris suffisamment d'expérience pour prendre adéquatement la relève de l'entretien du simulateur à la phase mise en service.

Le groupe d'entretien s'est enrichi de personnes additionnelles durant la phase mise en service pour finaliser l'organisation suivante: un ingénieur spécialiste d'intégration, trois ingénieurs en modélisation, un spécialiste
La formation du personnel de conduite en vue de sa certification auprès de la Commission de contrôle de l'énergie atomique (CCEA) est un processus d'évaluation exigeant qui pose des difficultés à Gentilly-2 depuis le début de l'exploitation de la centrale. Il faut compter actuellement cinq (5) ans pour qualifier un premier opérateur (excluant la formation pour sa qualification première) ou un chef de quart. Au minimum des examens spécifiques (nucléaire et conventionnel), nous avons en outre, dans le passé, des réussites conditionnelles demandant un complément de formation. L'enseignement magistral en salle de cours, l'étude personnelle dirigée et l'apprentissage structuré sur le tas, étaient jusqu'à maintenant les seules méthodes de formation disponibles. De plus, le maintien des connaissances et des habiletés pour le personnel de conduite autorisé, est difficile à réaliser en période normale d'exploitation. Les exploitants ont donc dû remédier à cette situation. Les simulations sont un complément valable de nos installations.

Avec la venue du simulateur, nous visons à faciliter l'apprentissage dans chaque programme spécifique (nucléaire et conventionnel) et augmenter ainsi le taux de réussite des candidats qui se présentent aux examens de la CCEA. Le simulateur ne supprimera pas les méthodes traditionnelles mais sera un complément de première importance et modifiera la structure des programmes spécifiques. Pour le personnel de conduite déjà autorisé, nous développons un programme de formation continue impliquant, en grande partie, l'utilisation du simulateur.

Nous avons intégré en février 89, suite à la mise en service, l'utilisation du simulateur au programme de formation nucléaire spécifique. Le simulateur a contribué, dans ce programme d'une durée de 24 semaines, pour l'équivalent de 9 semaines. Nous avons réduit (10) candidats que nous avons séparé en trois groupes. Ainsi, chaque groupe a reçu l'équivalent de trois semaines de formation sur le simulateur. Les plans de cours que nous avons utilisés sont principalement des procédures d'essais d'acceptation et de formation adaptées pour la formation, tels l'arrêt et le redémarrage de la centrale, le déclenchement du réacteur et retour à pleine puissance et plusieurs incidents nucléaires comme les différents types de PMCA, le bris d'un tube de force ou blocage de débit d'un canal. Nous avons pour le moment, une bonne dose pour qualifier l'impact de l'introduction du simulateur sur les résultats des examens de la CCEA, mais nous pouvons affirmer que les candidats ont beaucoup apprécié leur formation avec simulateur et ils ont manifesté une grande motivation pour ce genre de programme.

Selon notre plan quinquennal du programme de formation pour les chefs de quart et premiers opérateurs en vue de leur accréditation auprès de la CCEA, nous prévoyons une utilisation intensive du simulateur. À partir de janvier 1989 jusqu'à la fin de 1993, il est prévu d'entreprendre sept (7) programmes spécifiques (nucléaire ou conventionnel) pour des classes de 10 à 15 candidats. Chaque programme comprend deux parties dont la première, d'une durée moyenne de vingt (20) semaines, implique l'étude des systèmes ou le simulateur sera peu utilisé. Une deuxième partie intégrée de 24 semaines utilisera le simulateur environ un (1) jour par semaine par groupe. Au niveau des groupes de quatre (4) candidats, le simulateur sera utilisé à 75% du temps pour ces programmes de formation selon un horaire de huit (8) heures par jour.

Le programme de formation continue, pour sa part, exigera l'utilisation du simulateur huit (8) semaines. Nous menons un programme de 60 heures par équipe de quart (chef de quart et premier opérateur) selon une recommandation d'INTO.

L'équipe de formation proposée au simulateur est composée de trois instructeurs qui ont reçu une formation approfondie sur le simulateur, d'écrit les plans de leçons et de faire passer les tests. Ils auront, de plus, à préparer et diffuser les leçons pour le programme de formation continue du personnel déjà autorisé.

CONCLUSION

La conception, la fabrication, la mise au point et la validation d'un simulateur de formation font appel à du personnel très spécialisé ayant une bonne expertise dans le domaine. La participation du personnel d'Ontario-hydro fut incontestablement bénéfique pour le projet et l'expérience de la conception CNE dans le secteur d'activité des simulateurs a largement contribué au succès du projet. Il est important de bien définir, dès le début, les spécifications techniques et de bien déterminer la portée de la simulation pour chaque système. Ces spécifications seront nos guides pour recevoir les données de référence nécessaires lors de la conception ultérieure des modèles. De plus, elle seront de première importance pour la validation du simulateur. Pour éviter de faire de la mauvaise formation, dû à des modèles inadaptés, il est essentiel de valider le comportement du simulateur en se basant sur des données réelles et précises.

Dans tout projet, il faut se fixer un
échéancier précis. C'est très pertinent quand il s'agit de réaliser un simulateur car c'est un produit très personnalisé selon les spécifications du client et comportant plusieurs éléments techniques sophistiqués. Il faut d'abord établir les priorités et les éléments sur lesquels il sera important de se concentrer si on désire éviter de s'attarder à des problèmes n'ayant que peu d'impact sur l'avancement des travaux. Nous avons vu ce problème lors des essais d'acceptation et nous avons dû établir des listes catégorisant l'importance des déficiences à régler. Le retard, dans le projet, vient du fait que nous souhaitions absolument régler le maximum de déficiences avant d'entreprendre le premier programme de formation.

L'envoi d'une partie du personnel d'entretien du simulateur chez le manufacturier ONE pour acquérir de l'expérience s'est avéré une méthode rentable, le personnel a pris rapidement charge de l'exploitation et de l'entretien du simulateur à son retour à la centrale.

Après le premier programme de formation sur le simulateur, on peut dire que c'est un outil très complet pour la formation pratique du personnel de conduite. Nous sommes très satisfaits de sa performance et nous entrevoyons de nombreuses possibilités tant qu'à son utilisation. En effet, non seulement des programmes de formation spécifiques pour la certification du personnel de conduite et des programmes de formation continue, nous comptons utiliser le simulateur pour la vérification et la correction des procédures d'exploitation. A cet effet, le personnel d'exploitation a récemment validé le déroulement d'une procédure spéciale d'exploitation en vue de l'arrêt annuel planifié de la centrale, en juin 89. Le crédit que donne le personnel de conduite au simulateur est dû à la fidélité du comportement de ce dernier.

En somme, le projet d'acquisition d'un simulateur pour la centrale Centrally-2 fut un succès, grâce principalement à l'étroite collaboration entre les partenaires, à savoir: Ontario Hydro, ONE et le personnel d'Hydro-Québec.
FIGURE 1
SCHEMA BLOC DU SIMULATEUR GENTILLY-2
<table>
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<tbody>
<tr>
<td>INGENIERIE PRELIMINAIRE</td>
<td></td>
<td></td>
<td>16 SEM*</td>
<td></td>
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</tr>
<tr>
<td>FABRICATION</td>
<td></td>
<td></td>
<td></td>
<td>4 SEM</td>
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<tr>
<td>INTEGRATION H/W, S/W</td>
<td></td>
<td></td>
<td>2 SEM</td>
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<td></td>
</tr>
<tr>
<td>ESSAI A L'USINE</td>
<td></td>
<td></td>
<td></td>
<td>5 SEM</td>
<td></td>
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<td>INSTALLATION</td>
<td></td>
<td></td>
<td>4 SEM</td>
<td></td>
<td></td>
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<td>CORRECTION DES DEFIENCIES</td>
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<td></td>
<td></td>
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<tr>
<td>MISE EN SERVICE</td>
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<tr>
<td>DEBUT DE LA FORMATION</td>
<td></td>
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</table>

FIGURE 2
ECHEANCIER DU PROJET DU SIMULATEUR GENTILLY-2

ECHEANCIER PREVU : 196 SEM
TEMPS REEL : 210 SEM
ECLAT 14 SEM

*SEMAINES D'ECART SELON L'ECHEANCIER PREVU
**FIGURE 3**

**BILAN DES COUTS DU SIMULATEUR GENTILLY-2**

<table>
<thead>
<tr>
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<th>COUTS EN MILLIONS $</th>
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<tr>
<td></td>
<td>0 1 2 3 4 5 6 7 8 9 10 11 12</td>
</tr>
<tr>
<td><strong>INGENIERIE</strong></td>
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</tr>
<tr>
<td>PRELIMINAIRE</td>
<td>4,700</td>
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<tr>
<td>ET CONSULTANT</td>
<td>4,900</td>
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<tr>
<td>Δ = +,200</td>
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<td><strong>FABRICATION</strong></td>
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</tr>
<tr>
<td></td>
<td>11,600</td>
</tr>
<tr>
<td></td>
<td>12,000</td>
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<td>Δ = +,400</td>
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<td><strong>MANDAT DE LA REGION</strong></td>
<td>1,325</td>
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<tr>
<td></td>
<td>1,200</td>
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<tr>
<td>Δ = +,875</td>
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<tr>
<td><strong>FRAIS GENERAUX ET INTERETS</strong></td>
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<td>7,200</td>
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<td>Δ = -3,800</td>
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<td><strong>TOTAL</strong></td>
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<tr>
<td>REEL</td>
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<tr>
<td>ECART</td>
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<td>COUTS ESTIMES</td>
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<td>COUTS REELS</td>
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ABSTRACT

Investigations following the rupture of Pickering Unit 1 pressure tube G16 in 1983, led to the shutdown of Units 1 and 2 for pressure tube replacement and numerous other upgrades. They were commissioned in 1987 and 1988 respectively. This paper surveys the procedures used during the reactor physics recommissioning of these two reactors and presents the results of these measurements. Special note is made of the differences between this recommissioning work, and the initial commissioning of new CANDU reactors.

From a physics point of view, the restarted units differed substantially from the original design. The main difference in the core configuration involved the conversion of 10 of the original adjuster rods into shutoff rods. The reactivities of the remaining adjusters were increased. These substantial changes to the core, together with the full core of fresh fuel, necessitated a complete set of reactor physics recommissioning experiments.

Some of our procedures differed from those used to commission a new reactor. This was due mainly to the high levels of tritium in the moderator D2O and to radiological hazards on the reactivity deck. Also, the high residual activities in the rebuilt cores lead to increased difficulties in neutron monitoring and higher subcritical neutron count rates (hence a higher than usual reactor power at first criticality).

In general the results of our recommissioning measurements closely matched the results of presimulations using the OHRPSP and SMOKIN computer codes. Results for Unit 2 were generally better than those for Unit 1. This was due to improved procedures which resulted from our experiences with Unit 1.

INTRODUCTION

Investigations following the rupture of Pickering Unit 1 pressure tube G16 in 1983 resulted in the discovery of misplaced garter springs (separators) and the resulting contact between the pressure and calandria tubes. This contact is thought to have been a key factor in the uptake by the pressure tube of large amounts of deuterium. Formation of regions of massive zirconium deuteride in the zircaloy-2 material of the pressure tube had led to its embrittlement and subsequent rupture. The discovery of similar conditions in other intact pressure tubes led to the decision to replace all pressure tubes in Units 1 and 2. It was also decided to perform numerous other upgrades in order to take full advantage of this major outage.

Major Changes

The following changes were most important from the reactor physics point of view.

Adjuster and Shutoff Rod Replacement. The effectiveness of the reactor protective system was enhanced by the addition of ten new shutoff (SA) rods. These new shutoff rods took the place of ten of the original 18 adjuster (AA) rods. The effectiveness of adjuster functions (i.e. poison override and reactivity shim) and their cobalt production capability were maintained by increasing the reactivity worth of the remaining eight adjusters. Figure 1 schematically displays the locations of reactivity devices in the reconfigured core.

New Fuel Load. All fuel bundles in both reactors were discharged to the irradiated fuel bay and after pressure tube replacement, the cores were loaded with unirradiated fuel bundles. This new loading contained 112 depleted fuel bundles with an enrichment of 0.5 weight percent U-235 in uranium (compared to the 0.711 weight percent in the other 4568 natural bundles). Figure 2 displays the new fuel loading scheme in Pickering-A.

Increased Heat Transport D2O Isotopic. The heat transport systems of both reactors were reloaded with high isotopic virgin D2O. The moderator systems were reloaded with their original D2O inventories.
There were many other changes and upgrades in related reactor systems. The overall impact of these changes on reactor physics recommissioning of these reactors was the need for a complete program of tests, similar to those performed for a new reactor. These tests were called the 'Phase B' part of the recommissioning procedure. They carried the recommissioning program from the pre-critical testing of systems (Phase A testing) with shutdown guarantees in place, to the testing program that could only take place with the reactor critical (Phase C) at a variety of power levels. It is the intent of this paper to discuss the procedures and results of these tests.

The recommissioning procedures used for Units 1 and 2 were very similar. Most of the following discussion will concern Unit 1. Special mention will be made of significant differences between procedures and results for Units 1 and 2.

Phase B was divided into the following steps:
- moderator pump-up
- approach to critical
- regulating system checks
- reactivity calibration of the liquid zone control system
- reactivity calibration of adjuster rods and shutoff rods
- reactivity calibration of moderator level reduction
- protective system power rundown test
- moderator dump test
COMPUTER SIMULATIONS

Static Calculations

The OHRFSP (Ontario Hydro Reactor Fuelling Simulation Program) family of codes was used to simulate the static Phase-B tests, prior to recommissioning.

OHRFSP is a computer code intended to perform core design, core physics, and fuel management calculations for CANDU reactors. It calculates neutron flux and power distributions using a core design, core physics, and fuel management calculations for CANDU reactors. It calculates moderator boron concentration, to derive a reactivity worth of control devices, determination of reactivity worths of control devices, determination of harmonic modes, xenon transients and flux mapping.

OHRFSP was used to calculate the critical moderator boron concentration, to derive a reactivity calibration curve for the light water liquid zone control system, and to provide static reactivity worths for individual adjusters and shutdown rods. Adjusters in withdrawal sequence, and the reactivity worth of moderator level reduction (a regulating system function in Pickering-A).

Dynamic Simulations

The test of power rundown following shutoff rod insertion was simulated using the SMOKIN code.

SMOKIN is a three dimensional neutron kinetics code which uses the modal expansion approximation involving one neutron energy group. This method is based on the assumption that the power distribution at any time can be synthesized from the weighted sum of a set of harmonic modes (which are the solutions to the diffusion equation), multiplied by local effects functions which are dependent only on device positions. The natural harmonic modes were generated separately by the MONIC module of OHRFSP.

The power rundown was simulated twice, once assuming the slow shutoff rod insertion rates used in safety analyses and then using faster, more realistic, best estimate insertion characteristics.

PROCEDURES

The Phase B recommissioning of Units 1 and 2 were similar to the commissioning of new reactor units. Some differences were due to the radiological hazards (e.g., tritium and carbon-14 hazards in the reactor buildings and high boiler fields on the reactivity deck) involved in working with contaminated reactor systems. Other differences were the result of our intent to, whenever possible, follow procedures similar to those that would be used for the startup of an in-service reactor after a long shutdown. For example, it was decided that boron (in the form of powdered D₂O added to the poison addition tank) would be used as a neutron poison in the moderator. During the calibration of reactivity devices, despite the fact that boron has a specific reactivity about a factor of 3.5 less than that of gadolinium (the other available neutron poison) and is also much less soluble. Gadolinium has never been used in Pickering-A due to corrosion concerns of some components of the moderator system. The small amounts that would have been required were not allowed on this basis, despite the conclusion that the chemical effects of such small amounts of gadolinium for a short time would have been insignificant. This decision probably contributed to the relatively large error in our measurements (to be discussed later).

Also of concern to us, was the novel nature of the situation. This would be our first experience in commissioning a rebuilt reactor. Of particular concern was the large amounts of residual radioactivity in the core (mainly in the calandria tubes which had not been replaced). We anticipated increased subcritical neutron count rates, a higher than usual reactor power at first critical and difficulties in neutron detection (using our startup He-3 ion-chambers) due to high background gamma noise. A result of these concerns was the decision to keep the moderator boron concentration at a very conservative 15 mg/kg prior to and during moderator pump-up. The expected critical boron concentration was approximately 10.6 mg/kg (depending on actual conditions at first critical).

Initial Conditions

The state of the reactor prior to Phase B, i.e., immediately prior to removal of the last administrative "shutdown guarantee", was as described below.

The reactor was in a guaranteed shutdown state with the moderator dumped and in the dump tank, and containing at least 15 mg boron/kg D₂O. This moderator poison concentration was enough to hold the reactor deeply subcritical (about -35 mk) even at full moderator level. Moderator circulating temperature was held at 60 degrees Celsius and was kept stable throughout these procedures. Various guarantees were in place to ensure that the moderator level could not be raised and that additional D₂O could not be transferred into the moderator system. Moderator pumps were operating to ensure the availability of the liquid poison addition system, moderator mixing, representative sampling and seal flows to the shutoff rod mechanisms.

The reactor protective system was poised (i.e., shutoff rods out of core) and its three logic channels were connected to special channelized startup instrumentation which was capable of measuring even very low neutron count rates from a deeply subcritical core. Figure 3 schematically describes this startup instrumentation. Neutron monitoring was by three He-3 neutron detectors which were mounted in the out-of-core ion-chamber locations. This arrangement contrasts to that during the first commissioning of Unit 1 in 1971. At that time startup instrumentation consisted of three fission counters and one He-3 detector mounted in an aluminum tube in an empty (i.e., no fuel or D₂O) fuel channel (U-11). Additional He-3 detectors had also been mounted in the out-of-core startup instrumentation positions for the purpose of measuring an in-core to out-of-core attenuation factor.
The startup instrumentation trips in the protective system were in addition to the normal protective system trips.

The logic of the reactor regulating system was kept operational by use of a dummy signal to one of its three channels. This made up for the lack of rational neutron signals from the regulating system ion chambers. All liquid zone controller levels were fixed at 50 percent and the reactor regulating system was in the "special shutdown mode". All adjuster rods were in-core, although AA14 and AA5 were withdrawn from core prior to moderator pump-up, for later use in count rate stabilization and doubling time control during the approach-to-critical.

**Moderator Pump-up**

After removal of the last shutdown guarantee, moderator pump-up took place in small steps, and in a slow and controlled manner. The neutron count rate was monitored continuously and analyzed to determine our margin to criticality. The moderator boron concentration was measured periodically to ensure that it did not decrease. The moderator level was brought to full tank (8.0 meters).

**Approach to Critical**

The approach to critical was effected through the removal of moderator boron by a single ion exchange column. The flow through the column was preset so that a fresh column would remove about 1.0 mk of reactivity in about 10 minutes of operation.

Our procedure was to valve in the ion-exchange column for a predetermined time and then to valve out purification and allow count rates to stabilize (Note: AA14 and AA5, which had been withdrawn earlier, were available for partial reinsertion and count rate stabilization if required). During these intervals, calculations were done to measure our margin to criticality and to determine the length of the next period of purification. Moderator samples were periodically analyzed to ensure that the boron concentration was as expected.

When criticality was imminent, the rate of reactivity removal was substantially reduced (i.e., some steps were less than 0.1 mk) as required to obtain a low and controlled super-criticality (i.e., a count rate doubling time of 5 minutes or more corresponding to an excess reactivity of about $\leq 0.2$ mk was desired).

Once the reactor was supercritical, the count rate increase was stopped by partially inserting an adjuster and the moderator boron concentration was measured. After this the neutron count rate was allowed to increase (by withdrawing the adjuster partially) with a doubling time of about 5 minutes. Regulating ion-chamber signals were closely monitored and reactor power was allowed to increase bringing the ion chamber signals into their rational range at about $10^{-7}$ full power (full power). As the regulating ion-chamber signals became rational reactor power control was smoothly transferred to the bulk control function of the reactor regulating system. Reactor power was increased to about $10^{-6}$ full power and once we were sure that the regulating system was properly maintaining power, AA5 and AA14 were fully inserted in-core.

**Regulating System Checks**

A series of detailed checks were performed at low power to ensure that the regulating system was functioning properly (under bulk control). These tests included power doubling tests, tests of the independence of the three regulating system channels and tests of the various rates of power increase and decrease allowed by the regulating system (over two orders of magnitude in reactor power). The tests were undertaken alternately with either of the two digital control computers in control.

**Calibration of the Liquid Zone Control System**

The reactivity worths of the liquid zone control system under bulk control (i.e., average zone controller level changes) were calibrated by measuring zone controller level changes resulting from the addition of boron to the moderator system.

Using an assumed reactivity coefficient of $7.8$ mk/(mg boron/kg $D_2O$) and a moderator $D_2O$ inventory of 294 Mg, accurately weighed batches of $P_2O_5$ were prepared having reactivity worths of 1.0 mk and 0.5 mk. The average zone controller level was brought to about 75 percent, by the removal of moderator boron. Then the boron addition tank was thoroughly flushed and isolated from the moderator system. The starting average zone controller level was recorded. The zone control system was then calibrated using the following procedure:

- a batch of $P_2O_5$ was added to the boron addition tank
- $P_2O_5$ was recirculated through the tank in order to dissolve the $B_2O_3$
- the contents of the tank were flushed into the moderator system
The tank was filled with moderator D$_2$O and recirculated and flushed at least once more (until flushing no longer changed the average zone controller level), and

the new stable average zone controller level was recorded. (Note: the pressure in the liquid zone system delay tank undergoes a cycle that affects zone controller level readings. All zone controller level readings were recorded when the delay tank pressure was in the stable portion of its cycle.)

This procedure was repeated until the average zone controller level reached about 10 percent. Although our intent was to record zone controller levels only when they had stabilized, they never completely stabilized. The observed fluctuation in average zone controller level was about $\pm$ 4 percent over the entire range, i.e., $\pm$ 0.2 mk. The data resulting from this procedure were least square fitted to a quadratic polynomial and the resulting calibration curve was used to calculate the reactivity worths of adjuster and shutoff rods and of moderator level reductions.

**Measurement of Reactivity Worths of Adjuster and Shutoff Rods**

The reactivity worths of adjuster rods were measured individually and in withdrawal sequence, and the reactivity worths of shutoff rods were measured individually.

The individual adjuster worths were measured by first stabilizing the average zone controller level to about 30 percent, and then withdrawing and reinserting each rod. The stable average zone controller level after each withdrawal and reinsertion was recorded. As mentioned above, average zone controller levels were recorded when the liquid zone control system delay tank pressure was in the stable portion of its cycle.

Adjuster worths in sequence were measured by first stabilizing the average zone controller level at a low value, then withdrawing the adjusters in the normal withdrawal sequence. The stable average zone controller level before and after each withdrawal was measured. When an adjuster withdrawn increased the average zone controller level above 60 percent, a calibrated batch of B$_2$O$_3$ was added to the moderator system to reduce the average zone controller level to the low end of its range.

Individual shutoff rods worths were measured by first stabilizing the average zone controller level at 60 percent, and then inserting and withdrawing each rod. The stable average zone controller level after each insertion and withdrawal was recorded.

**Reactivity of Moderator Level Reductions**

The average zone controller level was stabilized at 70 percent with the moderator at full tank. The moderator level was decreased to about 89 percent full tank and the resulting average zone controller level was recorded. Boron was then extracted from the moderator system to bring the average zone controller level to about 70 percent. The moderator level was reduced to 80 percent full tank and the resulting average zone controller level was recorded.

**Protective System Power Rundown Test**

Our intent in this test was to demonstrate the effectiveness of the protective system by monitoring the neutron power rundown at an in-core and an out-of-core position during a reactor trip. To meet licensing requirements, the two most effective shutoff rods were to be inhibited from falling following the trip. The moderator was to be prevented from dumping.

We monitored signals from a spare in-core flux detector and an out-of-core regulating ion-chamber. The in-core detector was a platinum-clad inconel detector in a vertical assembly. The two most effective shutoff rods were chosen with respect to their effect on the ion-chamber signal and were SA8 and SA21.

The protective system was jumpered to meet the requirements mentioned above. The jumpers simulated a trip on Channel F, on both digital computers. They allowed all shutoff rods to fall after a single channel trip (on Channels D or E), except SA8 and SA21 which would have fallen if trips occurred on both channel D and E. The jumpers allowed a full drop test without dumping the moderator, although the moderator dump was still available on any two or three channel trip.

Prior to the test the reactor power was $10^{-3}$ full power and stable, and the reactor was in steady state operation. Moderator and heat transport temperatures were stable at 60°C and 45°C respectively.

After the test the trip was reset and the power rundown was stopped prior to loss of rational signal on the regulating ion-chambers.

**Moderator Dump Test**

The purpose of the moderator dump test was to demonstrate the effectiveness of the moderator dump by monitoring the neutron power and the characteristics of moderator level versus time during the dump.

Our test procedure allowed us to dump the moderator by tripping a single channel of the protective system. The test was designed so that the shutoff rods did not fall into core, although the protective system was kept operational. Shutoff rods would have fallen in the event of a two or three channel protective system trip.

Neutron signals were measured using the same detectors used for the power rundown test. A signal from the calandria wide range level indicator was used to monitor the moderator level.

**RESULTS AND DISCUSSION**

**Moderator Pump-up and Approach-to-Critical**

Startup instrumentation count rates decreased exponentially during the moderator pump-up (Figure 4).
FIGURE 4: UNIT 1 START-UP INSTRUMENTATION RESPONSE TO MODERATOR LEVEL CHANGE

This observation differed from our experience during the first commissioning of Unit 1 in 1971. At that time the Unit was brought to critical on moderator pump-up, and first criticality was at a moderator level of 4.32 m and a boron concentration of 7.25 mg/kg. It was also different to observations made during normal start-ups on start-up instrumentation (i.e., after long outages). In these circumstances just enough boron is added to the moderator system to ensure subcriticality at full tank. During moderator pump-up count rates are observed to first decrease due to the shielding effect of the moderator, and then increase as the subcriticality of the core decreases.

In our case the high moderator boron concentration made the reactor extremely subcritical (about -35 mk at full tank). At low moderator levels the detectors were getting most of their signal from epithermal neutrons from spontaneous fissions throughout the core and also from gamma noise (i.e., although gamma signals are at lower energies than those from neutrons, the high gamma background would have resulted in significant noise in the detector). The effect of pumping up a highly borated moderator was to reduce neutron count rates by thermalization of epithermal neutrons and their absorption by the boron. Photo-neutrons were similarly absorbed. Gamma noise was reduced due to shielding effects of the increasing moderator.

Following moderator pump-up, the approach to critical was by boron removal by intermittently valving in a single ion-exchange column at a constant flow rate. The first column became saturated and lost its effectiveness part way through this procedure and a fresh column was required for the final approach to critical.

Unit 1 was declared critical at a poison concentration of 10.8 ± 0.1 mg boron/kg D$_2$O, with a moderator temperature of 40°C, moderator isotopic of 99.87 weight percent, average zone controller level of 40 percent and AA5 and AA14 out-of-core.

The simulated critical boron concentrations at the conditions of first criticality were 10.8 mg boron/kg D$_2$O for Unit 1 and 10.6 mg boron/kg D$_2$O for Unit 2.

Reactor power at first criticality was estimated to be 10$^{-10}$ full power. After first criticality, reactor power was slowly allowed to increase and control was smoothly transferred to the regulating system at a power of 5 x 10$^{-7}$ full power.

Regulating System Checks

The regulating system checks confirmed that the system maintained reactor power as required, that it successfully raised and lowered reactor power to new setpoints at the rates specified and that the regulating ion-chambers operated independently.

We also confirmed that the protective ion-chambers responded correctly to changes in reactor power.

Calibration of the Liquid Zone Control System

The results of the liquid zone controller calibration procedures were least square fitted to a quadratic curve of reactivity ($\rho$) versus average zone controller level ($h'/A.$) of form

$$\rho(h'/A.) = a \times h'/A. + b \times (h'/A.)^2$$

where $h'/A.$ is expressed as a fraction. Table 1 presents the fitted constants for Unit 1, Unit 2 and the presimulation data. The calibration curves for Units 1 and 2 and the presimulation curve are presented as Figure 5.

FIGURE 5: LIQUID ZONE CONTROLLER LEVEL CALIBRATION CURVES FOR UNITS 1 AND 2

Reactivity (mk)

<table>
<thead>
<tr>
<th>Average Zone Level (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0  10  20  30  40  50  60  70  80  90  100</td>
</tr>
<tr>
<td>0  1   2   3   4   5</td>
</tr>
</tbody>
</table>

- Simulated
- Unit 2
- Unit 1
For Unit 1, over the working range of 20 percent to 70 percent fill, the reactivity worth was measured to be 2.80 mk and the slope of reactivity versus zone controller level was 0.056 mk/percent AZL. This contrasted to a presimulated worth of 2.58 mk and a presimulated slope of 0.052 mk/percent AZL.

Table 1

<table>
<thead>
<tr>
<th>Constant</th>
<th>Unit 1</th>
<th>Unit 2</th>
<th>Pre Simulated</th>
</tr>
</thead>
<tbody>
<tr>
<td>a (mk/fractional fill)</td>
<td>6.2619</td>
<td>6.600</td>
<td>7.2088</td>
</tr>
<tr>
<td>b (mk/fractional fill^2)</td>
<td>-0.7334</td>
<td>-2.592</td>
<td>-2.2666</td>
</tr>
</tbody>
</table>

\[ \rho (AZL) = a * AZL + b * AZL^2 \]

where \( \rho \) = reactivity (mk), \( AZL \) = fractional average zone level

In an attempt to reduce potential errors in the liquid zone calibration (and also to reduce errors in our measurements of shutoff and adjuster rod worths), we slightly modified our procedure during the recommissioning of Unit 2. Instead of dumping the powdered H_2O_2 into the poison addition tank, the poison batch was premixed with D_2O in a canister and added to the tank via a hose at a new location in the system. The canister was then rinsed with D_2O and emptied into the tank.

The liquid zone calibration results for Unit 2 were moderately better than those for Unit 1. Over the active range of 20 percent to 70 percent level, we measured a reactivity slope of 0.047 mk/percent AZL and a reactivity worth of 2.1 mk. The shape of the curve for Unit 2 matched the presimulation curve better than did that of Unit 1, indicating a lower random error in the calibration.

Reactivity Worths of Adjuster Rods and Shutoff Rods

The measured reactivity worths of individual adjuster rods, adjuster rods in withdrawal sequence and shutoff rods are presented in Tables 2, 3 and 4 respectively. All measured values were calculated using the liquid zone calibration curves discussed above.

In the case of individual adjuster rods (Table 2), symmetric rods were grouped together, so that devices of equal design value could be compared. For Unit 1 we found that the presimulated reactivities were systematically below the measured worths. The mean error was -16.2 percent with a standard deviation of 3.0 percent. Results were significantly better for Unit 2. In this case the mean error was only -3.4 percent.

The measurement of adjuster reactivities in withdrawal sequence gave similar results for Units 1 and 2 (Table 3). For Unit 1 the maximum error in bank worth was -13.0 percent (-0.45 mk) and the mean error was -9.2 percent (-0.32 mk). For Unit 2 the maximum error in bank worth was -9.4 percent (-0.25 mk), and the mean error was -6.4 percent (-0.21 mk).

For individual shutoff rod measurements (Table 4), symmetric rods were again grouped together. For Unit 1 predicted worths were consistently less than the measurements. The mean error was -19.6 percent (-0.45 mk). The mean error for Unit 2 was significantly less at -5.5 percent (-0.12 mk).

We attribute our better results in Unit 2 measurements partially to the improvements in the procedures used to calibrate the liquid zone control system. We were unable to identify the source of the other errors, although it was noted that during the individual adjuster worth measurements, the starting zone controller level (all adjusters in-core), varied by almost 4 percent for Unit 1 even though all reactor conditions were to our knowledge identical. Only a 1.6 percent variation was observed for Unit 2. We observed similar variations in the starting liquid zone controller level during the individual shutoff rod measurements. A 4.9 percent (-0.3 mk) variation was observed in the Unit 1 data. This fluctuation was 8.7 percent for Unit 2.

Despite the large errors, the shutoff rod reactivity worths exceeded specifications and our measurements were acceptable from a licensing and safety perspective.

Moderator Level Reduction

The reactivity worths of moderator level reduction were successfully measured and the results met test specifications.

For Unit 1, a decrease in moderator level from 100 percent to 88 percent resulted in a reactivity decrease of about 1.3 mk. The simulated worth was 1.1 mk. When the moderator level was dropped to 79 percent the reactivity decrease amounted to 3.6 mk (compared to a simulated worth of 3.2 mk).

Protective System Rundown Test

The protective system rundown tests were successfully completed for Units 1 and 2 and as intended, the moderator was not dumped. The trips were reset, the protective system repositioned and reactor power raised prior to loss of rational signal on the regulating ion-chambers.

The measured power rundown at in-core (Figure 6) and out-of-core (Figure 7) monitoring locations, were faster than the SMOKIN predictions done using the conservative licensing assumptions (ie. slower shutoff rod drop curves). They matched closely our simulations done using the best estimate insertion curves.

The tests were deemed to have been successfully completed and the results acceptable for licensing purposes.
Moderator Dump

The moderator dump test was successfully completed. As intended in the test design, the shutdown rods did not drop into core. The trip was reset, the moderator was pumped up and the reactor brought critical within about 15 minutes and prior to loss of rational signal on the regulating ion-chambers.

The rate of moderator dump compared favorably to design curves. It was demonstrated that the moderator dump could effectively shut down the reactor and the criteria for success in this test were met. Figure 8 presents the moderator level versus time curve for Unit 1. Figure 9 displays the in-core and out-of-core detector signals.

CONCLUSIONS

The reactor physics 'Phase B' recommissioning tests for Units 1 and 2 at the Pickering Nuclear Generating Station were successfully completed.

This test program was followed by the rest of the recommissioning program in which detailed tests of other reactor systems were performed at higher powers. The reactors were returned to service on October 10, 1987 and November 11, 1988 respectively.
### TABLE 2

**Reactivity Worth of Individual Adjusters**

<table>
<thead>
<tr>
<th>Adjuster</th>
<th>Simulation (S) (mk)</th>
<th>Measurement (M) (mk)</th>
<th>Error (S/M-1) (percent)</th>
<th>Measurement (M) (mk)</th>
<th>Error (S/M-1) (percent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>AA-5</td>
<td>1.45</td>
<td>1.68</td>
<td>-13.7</td>
<td>1.40</td>
<td>3.8</td>
</tr>
<tr>
<td>AA-14</td>
<td>1.45</td>
<td>1.63</td>
<td>-11.0</td>
<td>1.51</td>
<td>-4.1</td>
</tr>
<tr>
<td>AA-7</td>
<td>1.24</td>
<td>1.51</td>
<td>-17.9</td>
<td>1.30</td>
<td>-4.4</td>
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<tr>
<td>AA-12</td>
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<td>1.42</td>
<td>-12.7</td>
<td>1.17</td>
<td>5.7</td>
</tr>
<tr>
<td>AA-9</td>
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<td>1.54</td>
<td>-19.5</td>
<td>1.25</td>
<td>-0.7</td>
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<tr>
<td>AA-10</td>
<td>1.24</td>
<td>1.53</td>
<td>-18.9</td>
<td>1.33</td>
<td>-6.8</td>
</tr>
<tr>
<td>AA-8</td>
<td>1.22</td>
<td>1.47</td>
<td>-17.0</td>
<td>1.39</td>
<td>-12.3</td>
</tr>
<tr>
<td>AA-11</td>
<td>1.22</td>
<td>1.50</td>
<td>-18.7</td>
<td>1.33</td>
<td>-8.3</td>
</tr>
</tbody>
</table>

**Mean Error** -16.2  **Standard Deviation** 3.0

### TABLE 3

**Reactivity Worth of Adjusters in Withdrawal Sequence**

<table>
<thead>
<tr>
<th>Adjuster</th>
<th>Measurement (M) (mk)</th>
<th>Error (S/M-1) (percent)</th>
<th>Measurement (M) (mk)</th>
<th>Error (S/M-1) (percent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>AA-5</td>
<td>1.42±0.1</td>
<td>-4.2</td>
<td>1.37±0.1</td>
<td>+5.8</td>
</tr>
<tr>
<td>AA-14</td>
<td>1.61±0.1</td>
<td>-14.9</td>
<td>1.50±0.1</td>
<td>-23.3</td>
</tr>
<tr>
<td>Rank A</td>
<td>3.63±0.2</td>
<td>-9.6</td>
<td>2.87±0.2</td>
<td>-9.3</td>
</tr>
<tr>
<td>AA-7</td>
<td>1.42±0.1</td>
<td>-2.1</td>
<td>1.49±0.1</td>
<td>-8.1</td>
</tr>
<tr>
<td>AA-12</td>
<td>1.45±0.1</td>
<td>-11.7</td>
<td>1.21±0.1</td>
<td>-10.7</td>
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<tr>
<td>Rank B</td>
<td>2.87±0.2</td>
<td>-7.0</td>
<td>2.70±0.2</td>
<td>-9.4</td>
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<tr>
<td>AA-9</td>
<td>1.69±0.1</td>
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<td>1.46±0.1</td>
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<tr>
<td>AA-10</td>
<td>1.77±0.1</td>
<td>-17.5</td>
<td>1.39±0.1</td>
<td>-9.4</td>
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<tr>
<td>Rank C</td>
<td>3.46±0.2</td>
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<td>2.85±0.2</td>
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<tr>
<td>AA-8</td>
<td>2.15±0.1</td>
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<td>2.03±0.1</td>
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<tr>
<td>AA-11</td>
<td>2.70±0.1</td>
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</tr>
<tr>
<td>Rank D</td>
<td>4.85±0.2</td>
<td>-7.0</td>
<td>4.47±0.2</td>
<td>-6.2</td>
</tr>
</tbody>
</table>

**Mean error in bank worth** -9.2  **Standard Deviation** 2.5

**Mean error in bank worth** -6.4  **Standard Deviation** 3.5
<table>
<thead>
<tr>
<th>Shutoff Rod</th>
<th>Simulation (S) (mk)</th>
<th>Measurement (M) (mk)</th>
<th>Error (S/M-1) (percent)</th>
<th>Measurement (M) (mk)</th>
<th>Error (S/M-1) (percent)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SA-1</td>
<td>2.01</td>
<td>2.41</td>
<td>-16.6</td>
<td>2.12</td>
<td>-5.3</td>
</tr>
<tr>
<td>SA-10</td>
<td>2.01</td>
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<td>-19.3</td>
<td>2.25</td>
<td>-10.8</td>
</tr>
<tr>
<td></td>
<td>2.01</td>
<td>2.45</td>
<td></td>
<td>2.19</td>
<td></td>
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<tr>
<td>SA-11</td>
<td>1.69</td>
<td>1.98</td>
<td>-14.6</td>
<td>1.57</td>
<td>7.9</td>
</tr>
<tr>
<td>SA-9</td>
<td>1.69</td>
<td>2.12</td>
<td>-20.3</td>
<td>2.06</td>
<td>-17.9</td>
</tr>
<tr>
<td></td>
<td>1.69</td>
<td>2.05</td>
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<td>1.81</td>
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<tr>
<td>SA-3</td>
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<td>2.62</td>
<td>-18.3</td>
<td>2.29</td>
<td>-6.6</td>
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<tr>
<td>SA-4</td>
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<td>2.33</td>
<td>-18.9</td>
<td>2.18</td>
<td>-13.5</td>
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<tr>
<td>SA-5</td>
<td>2.13</td>
<td>2.82</td>
<td>-24.5</td>
<td>2.67</td>
<td>-20.2</td>
</tr>
<tr>
<td>SA-6</td>
<td>2.15</td>
<td>2.80</td>
<td>-23.1</td>
<td>2.56</td>
<td>-16.1</td>
</tr>
<tr>
<td>SA-7</td>
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<td>-23.5</td>
<td>2.72</td>
<td>-19.7</td>
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Mean Error: -19.6, Mean Error: -5.5
Standard Deviation: 3.1, Standard Deviation: 11.1
Session 8:

Reactor Physics Simulation

Chairman:

A.L. Wight, Ontario Hydro
COUPLED NEUTRONICS - THERMOHYDRAULICS SIMULATIONS
OF FAST TRANSIENTS IN CANDU

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ABSTRACT

The neutron kinetics code CERBERUS and the thermohydraulics code FIREBIRD-III MOD1-77 have been integrated into a computer-program package capable of performing accurate coupled simulations of fast power transients. The package executes on a desk-top computer (APOLLLO) for convenience and economy. The integrated package consists of several interlinked programs, including the lattice code POWDERPUFS-V, the reactor-modelling code MATMAP, the neutron 3-d kinetics code CERBERUS, the thermohydraulics code FIREBIRD-III MOD1-77, and the detector- and electronics-modelling code POINTSIM. The program package was used to study hypothetical large-loss-of-coolant accidents in an equilibrium CANDU 6 core. The transients were assumed to be terminated by shutdown-system-1 action. In these simulations, an unprecedented level of detail was built into the thermohydraulics model. The pass downstream of the break (in which the coolant density changes are rapid) was modelled by 5 representative channel groups, to take into account zones of different power and of different channel elevation. The other three passes were modelled by one representative channel group each. Different coolant-density and fuel-temperature transients applied in different parts of the core. Results show that even under very conservative assumptions at all stages of the calculations, the power-pulse energy deposition in the fuel easily meets safety-related criteria.

1.0 INTRODUCTION

The analysis of fast power transients, such as those resulting from hypothetical large-loss-of-coolant accidents (LOCA), requires several computational tools, of which the two major ones are:

a) a thermohydraulics code to calculate (mainly) the time dependence of the coolant density in the core, and

b) a neutron kinetics code to calculate the change in neutron flux and power with time.

The standard thermohydraulics code in use at Atomic Energy of Canada Limited, CANDU Operations, is FIREBIRD, a program for one-dimensional hydraulics networks. The current version is FIREBIRD III MOD1. As for the neutronics, the most accurate kinetics code for CANDU is CERBERUS, which solves the time-dependent neutron diffusion equation in its finite-difference form.

Most previous analyses of LOCA in CANDU have employed thermohydraulics and neutronics codes in a decoupled mode. In such a mode, to calculate the power transient arising from a particular break size and location, FIREBIRD was executed first to compute the coolant density variation over the entire time of interest, for instance using a power pulse from a previous calculation as input. The predicted coolant density transient was then fed to CERBERUS, which computed the power pulse corresponding to the LOCA in question. While the analysis could always iterate back to time zero and proceed again through FIREBIRD and CERBERUS, this was seldom considered cost effective or necessary for a particular LOCA, as the major determinant of the coolant voiding is the blowdown.

In the present work, FIREBIRD and CERBERUS have been linked together and into a larger package of programs which constitutes a convenient framework for performing neutronics-thermohydraulics simulations of fast transients in a coupled mode.

2.0 THE PROGRAM PACKAGE

2.1 The Code Set

The program package includes the following units:

a) the cell code POWDERPUFS-V which calculates the lattice nuclear cross sections at various positions in core and at different instants in the transient.

b) the reactor-modelling code MATMAP which is used to set up the finite-reactor model, introduce model changes arising from the coolant voiding and reactivity-device movements, and superimpose device incremental cross sections on the lattice properties.

c) the neutron kinetics code CERBERUS, based on the Improved Quasi-Static method (1). CERBERUS computes the time-dependent neutron flux in three spatial dimensions and two energy groups, taking into account delayed-neutron effects which are very significant in fast transients.

d) CERBSPOW, the power module of CERBERUS, which calculates the 3-dimensional power distribution. The energy from the decay of fission products created during the power pulse is also taken into account.

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* the thermohydraulics-network code FIREBIRD-III MOD1-77, which solves a one-dimensional homogeneous two-phase-flow conservation equation and heat-conduction equations in the fuel and pipes. Slip and drift correlations are given to account for velocity differences between the steam and liquid phases.

* INTREP, a program to calculate fluxes at detector positions by interpolating in the 3-dimensional flux distribution provided by CERBERUS.

* POINTSIM, which models the electronics of the protection systems, determines in-core-detector and ion-chamber responses, and calculates the time(s) of actuation of the shutdown system(s).

The program package has been installed on an APOLLO (series DN3000) microcomputer workstation. The high execution costs associated with mainframes are then avoided, removing one of the roadblocks to more frequent 3-d calculations of transients. While program execution is slower than on mainframes (typically by a factor of 6-10 compared to FORTRAN-77), this is offset by the convenience of desktop computing. The programs are all written in FORTRAN-77. This makes the package very easily transportable.

2.2 Sequence of Execution

The transient of interest is simulated by the repeated execution of the programs in the sequence shown in Fig. 1.

![Diagram of sequence of calculations in simulation of transient](https://via.placeholder.com/150)

The cycle is repeated at every "flux-shape" time interval (typically 0.05 - 0.10 s), that is, the interval at which CERBERUS explicitly recalculates the full 3-dimensional flux distribution. Thus the neutronics-thermohydraulics coupling is continuous at all stages of the transient.

The power distribution from CERBERUS/CERBSPOW at each "flux-shape" time is fed to FIREBIRD, which evaluates the coolant densities and fuel and coolant temperatures for the next time, when these are fed back to CERBERUS via the lattice properties (in POKERPUFS-V). Thus void and temperature effects on reactivity are taken into account at all points in the core. In the early stages of the transients, INTREP and POINTSIM are run to evaluate ion-chamber and in-core-detector responses and thereby determine the times at which trip setpoints are reached.

The improved quasi-static method on which CERBERUS is based concentrates most of the time dependence of the flux \( \psi(r,t) \) in a space-independent amplitude factor \( A(t) \). Thus the shape function \( \psi(r,t) \), the other factor in the total flux, \( \psi(r,t) = A(t) \psi(r,t) \) has a relatively weak time dependence. This allows relatively large values for the "flux-shape" time intervals. For instance, when modelling the action of shutdown-system-1 (the cadmium shutoff rods), the time it takes the rods to drop by one or two successive lattice pitches serves as an appropriate time step.

Note that within each "flux-shape" time interval FIREBIRD uses its own small time step (e.g. 0.001 s) appropriate to fast LOCA transients. Similarly, even in CERBERUS, a small time step is used to solve the point-kinetics-like equation for the amplitude \( A(t) \).

Routines for automatic file-name generation (using an alphanumeric label for the transient being calculated, plus the CERBERUS case number) have been put in place, allowing a high level of automation in the linking of the codes.

The procedure just described "walks through" the transient only once. There is no backtracking. As a result, the coupled calculation is performed at little or no additional computational effort relative to the previous uncoupled mode. There is consequently no incentive remaining to perform transient calculations without the coupling.

3.0 COUPLED SIMULATION OF A LOCA TRANSIENT

The program package described above has been used to study hypothetical large loss-of-coolant accidents in an equilibrium CANDU 6 core. These transients have been assumed to be terminated by the action of shutdown-system number 1 (SDS-1), and have been simulated to a time of 5 s after the break.

The results will be illustrated here mainly by means of one typical transient, a 30% reactor-inlet-header break. Some results of other transients will be used to make certain points.

The initial reactor condition prior to the LOCA was assumed to be full-power operation, but following a long shutdown, with the absence of saturated fission products in the fuel. The enhanced core reactivity corresponding to this condition was held down by 5.25 ppm of boron in the moderator. This increases the lattice void reactivity by more than 3 milli-k.
Other conservative assumptions were used in the calculation, such as neglecting fuel-pin dryout when computing the heat transfer from fuel to coolant, assuming conservative detector trip setpoints (including uncertainties), crediting the backup (rather than the first) trip on the third logic channel, and hypothesizing that two of the 28 shutoff rods remain out of core (i.e. actuating the least effective set of 26 rods).

The return pass of the broken loop (95 channels) was represented by channel group 6. The passes in the intact loop were represented by channel groups 7 and 8.

The FIREBIRD nodalization for the broken loop, corresponding to the core division given in Figure 2, is shown in Figure 3. Note that channel groups 1 to 5 were each subdivided into 12 nodes (one per axial bundle position), whereas channel group 6 was subdivided into only 4 nodes (as were channel groups 7 and 8).

3.2 Mechanics of Neutronics-Thermohydraulics Coupling

At each CERBERUS calculation, CERBSPOW produced a full 3-d power map consisting of 4560 bundle powers.

From this information, CERBSPOW created a file which was fed to FIREBIRD. This file contained, for each of the 8 channel groups described in the previous section, the values of power corresponding to the FIREBIRD nodalization. For each channel group, 12 axial values of power were provided, summed over all channels belonging to that channel group (as defined by Fig. 2).

FIREBIRD used these values of power to calculate the change in densities, fuel temperatures, and coolant temperatures over the next interval of the transient to the next CERBERUS calculation time. The end-of-interval values were input back into POMDERPUS-V, which updated the lattice neutronic properties. These were subsequently combined by MATMAP with the device (e.g. shutoff-rod) incremental cross sections to produce the full 3-dimensional distribution of core properties used by CERBERUS in the next flux-shape calculation.
3.3 Flux-Shape Time Interval

A variable flux-shape time interval $\Delta t$ was used in this transient. Up to the time of actuation of SDS-1, a constant $\Delta t$ value of 0.1 s was adopted. Following SDS-1 actuation, variable values of $\Delta t$ were chosen to make the flux-shape calculations coincide with the instants at which the leading edge of the shutoff rods crossed each horizontal model mesh line down to the reactor horizontal midplane, and every second mesh line afterward (justified since the transient was then winding down) — see Figure 4. Based on this criterion, values of $\Delta t$ in the range of 0.05 - 0.10 seconds were used until full shutoff-rod insertion. These small values ensure good convergence in the time behaviour of the CERBERUS solution. Following shutoff-rod complete insertion, which occurred about 1.7 s after the time of actuation of SDS-1, the flux-shape changes are small and larger values of $\Delta t$ are appropriate. At this stage the calculation used a maximum value of 1 second for $\Delta t$.

Within each time interval $\Delta t$, the FIREBIRD calculation used its own time step of the order of 1 millisecond and assumed a constant core power distribution. This means the bottom "staircase" (see Figure 5) has been used to calculate the density transient. It has been verified that for the time steps used, the upper staircase in Figure 5 gives essentially identical results in density (differences of the order of 1%).

4.0 RESULTS

Unless otherwise indicated, the results shown are for the 30% reactor-inlet-header break from full power, described in section 3.0.

4.1 Thermohydraulics Results

The coolant density transients obtained for FIREBIRD channel groups 1-8 are shown in Fig. 6. The large variation obtained from channel group to channel group is evident.

The most important time interval as far as determining the size of the power pulse is concerned is from 0 to 1 second, since the shutdown system is not yet fully effective. It can be seen that it is mostly the coolant densities in channel groups 1 to 5 (i.e. the critical core pass) which are important, since the densities in groups 6 to 8 are practically unchanged from the steady state value. Due to greater boiling, the coolant densities in the higher power channels (channel groups 1 and 2) are smaller than those in the lower power channels (channel groups 3 to 5).
greater coolant flashing, the densities in the higher elevation channels are lower than those in the lower elevation channels (e.g. channel group 1 vs. 2, or channel group 3 vs. 5). At 0.6 s after the break, for instance, the reduction in coolant density from the steady-state value is about 50\%.

The various void fractions showed even greater variation one from another for a 20\% reactor-inlet-header transient. These results are shown in Figure 7. This type of spatial effect is of course not provided by previous LOCA calculations.

Figure 8 shows the flows at the axial centre of channel groups 1 to 5. There is a large variation in flow from one channel group to another. In general, the closer these flows are to zero the smaller the density, leading to a larger power pulse. It can be seen that the 30\% break leads to almost perfect flow stagnation (i.e. flow = 0 kg/s) after about 0.35 seconds.

The reactivity is also depicted graphically in Figure 9. It is seen to peak at about 1 s after the break (and about half a second after the actuation of SDS-1). After this time the shutoff-rod negative reactivity quickly overcomes the positive void reactivity, and the reactor is rendered subcritical by about 1.35 s after the break.

Figure 10 shows the power transient for the left and right halves of the reactor, putting in evidence the side-to-side tilt due to the voiding of one pass in the left loop.

Figure 11 shows the power transients for channel groups 1, 2 and 4. Channel group 2 experiences a higher power peak and a significantly higher energy generation in the power pulse (area under the curve) than channel group 1. The main difference is the time at which the shutdown system reaches the two channel groups.

### 4.2 Neutronics Results

Table 1 shows the variation of system reactivity and of total and left and right-half reactor power through the transient.

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4.3 Effect of Power Distribution on Void Transients

The interplay of the power distribution on the coolant density transients can be seen by comparing power pulses with different initial conditions.

The coolant densities for a different transient starting from a (bottom-high) top-to-bottom flux tilt are shown in Figure 13. By comparing to Figure 6, one can see a dramatic change in the relative timing of the voiding of channel groups 1 and 2, because of the lower power level of group 1 associated with the tilt.

A similar effect, but due to a radial redistribution of power, can be seen by comparing Figures 14 and 15. Both of these transients are 100% pump suction breaks from an initial reactor power level of 65%, but with different reactivity-device configurations. For the transient in Figure 14 all adjusters were in their
nominal (in-core) positions, while for the transients of Figure 15 all adjusters were withdrawn (shim configuration). It is evident that the radial peaking due to the withdrawal of the adjusters created a differential effect in the voiding transients of channel groups 3-5 relative to groups 1 and 2.

These examples illustrate some of the new effects emerging from the coupled calculations.

4.4 Assessment of Fuel Integrity

The criterion which was used for the assessment of fuel integrity is the energy content of the fuel. This includes the initial stored energy, the decay heat, and the energy added by the power pulse. For conservatism this last term was evaluated in the adiabatic approximation, i.e., the power removed by the coolant during the pulse was ignored. The energy added was integrated to about 5s after the break, although the power generation dropped to a smaller value than the power to coolant before that time (typically at about 3s after the break for a large LOCA).

The energy content of the fuel in the hottest element for the most severe transient studied was about 650 J/g. This is only about 77% of the conservative but often used lower-limit value for fuel breakup, 840 J/g.

It has been shown therefore that the criterion for fuel integrity has been easily met.

5.0 WHEN DOES THE POWER TURN OVERT?

An interesting effect relating to reactivity is apparent from the results in Table 1. This effect is not particular to this specific transient, and is discussed here because it seems to run counter to expectations. By examining Table 1 it can be seen that the reactivity is increasing up to about 0.97 s, at which time it begins to decrease due to shutoff-rod "bite". The reactivity does not become negative until approximately 1.35 s. The total prompt reactor power, however, turns over at around 1.10 s, at which time the reactivity is still positive.

At first glance, it may appear that the power should increase as long as the reactivity is positive, and should start to decrease only when the reactivity changes sign. However, this conclusion is not necessarily correct in the presence of delayed neutrons and when the reactivity is smaller than the total delayed neutron fraction. The reason is that when the reactivity is less than the delayed fraction, the reactor is actually subcritical with respect to prompt neutrons, which has the following consequences.

In the steady state, the delayed source just compensates for the prompt subcriticality. However, in a transient condition, if the power has risen higher than the steady-state level due to a temporary positive reactivity, and the delayed source has not yet changed significantly, it is not possible to sustain the higher power level unless
the reactivity is maintained above a threshold positive value, which depends on the relative power. This may be seen numerically from a simplified point-kinetics equation for a single delayed-neutron-precursor group (more than one group will not change the physics):

\[
\frac{dA}{dt} = p - \beta A + \lambda C
\]

where \( A \) is the total power, \( p \) the reactivity, \( \beta \) the total delayed fraction, and \( C \) the delayed source.

From this equation, it is easy to deduce that \( \frac{dA}{dt} \) can be negative (i.e. \( A \) can decrease) if \( p \) is sufficiently small, i.e. if \( (p-\beta) \) is sufficiently negative to make the first term larger than the second term. Thus the power will decrease for certain combinations of \( p, \beta, A \), even if \( p \) is positive (but only in the case of prompt subcriticality).

6.0 SUMMARY

The package for coupled neutronics-thermohydraulics simulations of fast transients in CANDU has been demonstrated by its application to LOCA transients.

The low cost and convenience of executing the package on a desk-top computer should make it possible to apply this method routinely in the analysis of fast transients, where spatial effects are important. Also, the computational effort expended by the coupled package is essentially the same as that for uncoupled simulations with the same models.

Novel effects of spatial variation were found with very detailed thermohydraulics models.

Even with many levels of conservative assumptions, the safety criterion of fuel integrity has been shown to be satisfied for the hottest fuel pin in the most severe transient studied.

7.0 REFERENCES


8.0 ACKNOWLEDGEMENTS

The authors are grateful to C.R. Calabrese and D.B. Buss who assisted in the calculations of the fast LOCA transients.

Thanks are due to M.H. Younis for providing Figure 13 which illustrates a transient from an initial tilt.
TRAWA-FEE SIMULATION OF REACTIVITY TRANSIENTS IN PWR CORES

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ABSTRACT

TRAWA-FEE is a version of the Finnish code TRAWA, which solves the two-group neutron diffusion equations simultaneously with the heat conduction equations and the nonequilibrium separated two-phase hydraulic equations for one or more channels in one dimension. The modifications of original TRAWA code include: new models of control rod movement, additional options of DFDT and OOTD trips based on Westinghouse methodology, time dependent triggering conditions, changes in fuel rod heat conduction model, incorporation of improved DNBR correlations, FEE developed system for group constants generation based on LEOPARD, EXTERMINATOR-2 and SURFIT codes. TRAWA-FEE model was tested on a two loop Westinghouse PWR core of NPP Krsko by simulation of reactivity transients.

INTRODUCTION

The Faculty of Electrical Engineering (FEE), University of Zagreb is coordinating the activity of Reactor Safety Group in Zagreb, with the aim to upgrade the domestic capability in nuclear safety analysis as well as to predict the behaviour of safety related systems in NPP Krsko following specific transients and accidents. The computer code TRAWA (1), developed by Technical Research Centre of Finland, Helsinki, was acquired from NEA Data Bank, Paris for reactor core dynamic analysis.

COMPUTER CODE TRAWA-FEE MODEL

The reactor core in TRAWA code is divided in three radial flow regions and up to ten axial fuel regions with different nucleonic characteristics. Two-group time dependent diffusion equations are solved by FE discretization in axial direction and theta-method or Wi spectral method time discretization. The spatial dependence in radial direction could be taken into account by using spatial and time dependent radial shape functions. Thermohydraulics is coupled to neutronics by means of fuel temperature and coolant temperature, coolant density and soluble poison density dependent group constants. The hydraulic model includes normal and hot channels with one-dimensional conservation equations for mass, energy and momentum. Separated nonequilibrium flow equations are solved by implicit theta-method discretization in time and space. There is one representative fuel rod in each flow region. The heat conduction equation in pellet and cladding is solved by FEM in radial direction and theta-method in time. All material thermal properties, gap and coolant heat transfer coefficients are temperature dependent. The TRAWA code was successfully implemented and tested on UNIVAC 1108 computer. TRAWA code simulation of reactivity transients showed the necessity for some model improvements and modifications. The FEE version of the TRAWA code includes: the changes in input processing; the new model of control rod movement; the additional options of DFDT and OOTDT trips based on Westinghouse methodology; the time dependent triggering conditions; the changes in the fuel rod heat conduction model in order to improve fuel modelling near UO2 melting point; the introduction of the MATPRO (2) thermal properties; the incorporation of improved DNBR correlations (Smolin, W-3, BM2, CE-1); FEE developed system for group constants generation based on LEOPARD (3), EXTERMINATOR-2 (4) and SURFIT (5) codes; the introduction of new options in fuel rod heat conduction calculations, one based on FEM with additional temperature iteration during transient and the other one based on FD method with additional temperature iteration during transient; the conversion from UNIVAC 1108 FORTRAN v to IBM 4341 and VAX 11-750 FORTRAN 77.

The TRAWA-FEE model was tested on a two loop Westinghouse PWR core of NPP Krsko, by simulation of rod ejection accident. The nucleonic data are calculated for cycle 5 BOC and thermohydraulic data are used for 5% steam generator U-tube plugging conditions. Steady-state results are validated by comparison with FSAR (6) data: DNBR in hot channel is 2.09 (FSAR 2.01); maximum fuel center temperature 2470 K (FSAR 2475 K); average void fraction in hot channel 7.48 % (FSAR 7.68 %).

The rod ejection accident simulation was performed using four different fuel heat conduction models:
1. original finite element model (FE), 15 nodes in fuel, 2 nodes in clad, no temperature iterations, thermal properties are linear function of temperature (except for UO2 thermal conductivity),
2. new FE model, 15 nodes in fuel, 2 nodes in clad, with temperature iterations, thermal properties are linear function of temperature (except UO2 thermal conductivity),
3. new FE model, 15 nodes in fuel, 2 nodes in clad, with temperature iterations and MATPRO thermal properties,
4. standard finite difference model (FD), 15 nodes in fuel, 2 nodes in clad, with temperature iterations and MATPRO thermal properties.

Westinghouse UO2 thermal conductivity correlation was used in all four models. Distinguished characteristics of new FE and FD models are: introduction of arbitrary functional temperature dependence of specific heat; simulation of fuel behaviour near melting point and usage of temperature iteration loop, which includes heat transfer coefficient calculations. Maximum number of iterations is 20 and maximum allowed absolute temperature difference between two iterations is 0.5 K. Tridiagonal system of equations is solved using Gauss elimination method. Standard FD model is similar to RELAP-5/MOD1 (7) heat slab calculation.

The conditions assumed during simulations are the
following: initial reactor power 1.02 Pn; reactivity increase 230 pcm in 100 ms; nuclear power trip setpoint 1.18 Pn; scram delay time is 0.5 s; scram reactivity 5000 pcm; time dependent hot channel shape factor is linear function of time changing from 2.32 in t=0. s to 6.90 in t=100. ms; hot channel flow reduction 7%; gap thermal resistance is linear function of the fuel surface temperature changing from 1. cm²-K/W for 830. K to 0. cm²-K/W for 910. K; DNBK is calculated using W-3 correlation with R-grid spacer factor; 41 mesh point in axial direction; 3 time regions with total number of 101 fixed time steps.

Figures 1. and 2. show maximum fuel center temperature vs. time and maximum clad surface temperature vs. time, respectively. Model 1 results are denoted by KEMP and model 2 results are denoted by KEMPFM+L. The results of the models 3 and 4 are practically identical and are denoted by KEMPFM+H and KEMPFD+H. Maximum fuel center temperatures, maximum clad surface temperatures and peak fission powers are given in Table 1 for all models.

The corresponding CPU times on VAX 11-750 computer for models 1 to 4 are 1169 s, 1546 s, 1546 s and 1392 s, respectively. The cumulative CPU time vs. transient time is given in Figure 3. The simulation of the same accident using automatic time selection option would take only 411 CPU seconds for 54 time steps. The maximum fuel temperatures during accidents are lower than FSAR values because of more conservative FSAR assumptions. The analysis of different heat conduction models shows that model 1 with automatic time step selection option is appropriate for preliminary calculations. The models 2, 3 and 4 have higher accuracy but are more time consuming. The MATPRO properties should be used if melting of UO2 is expected. FD model is faster than FE model for the same nodalization and accuracy.

### Table 1

<table>
<thead>
<tr>
<th>Model no.</th>
<th>Max. UO2 center temperature value (K)</th>
<th>Max. clad surf. temperature value (K)</th>
<th>Peak fission power value (W)</th>
<th>Time (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2749.9</td>
<td>1178.4</td>
<td>3204.3</td>
<td>8.105</td>
</tr>
<tr>
<td>2</td>
<td>2778.2</td>
<td>1185.5</td>
<td>3187.9</td>
<td>8.105</td>
</tr>
<tr>
<td>3</td>
<td>2827.8</td>
<td>1220.1</td>
<td>3178.4</td>
<td>8.105</td>
</tr>
<tr>
<td>4</td>
<td>2827.5</td>
<td>1219.4</td>
<td>3176.7</td>
<td>8.105</td>
</tr>
</tbody>
</table>

---

**Figure 1:** Maximum fuel center temperature vs. time

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8-10 CNS 10th Annual Conference, 1989
FIGURE 2: MAXIMUM CLADDING SURFACE TEMPERATURE VS. TIME

FIGURE 3: CUMULATIVE CPU TIME VS. TRANSIENT TIME
ANALYSES OF REACTIVITY INDUCED ACCIDENTS

The following reactivity induced accidents have been analyzed:
a/ rod cluster control assembly (RCCA) ejection,
b/ uncontrolled RCCA bank withdrawal,
c/ uncontrolled boron dilution.

Rod_cluster control assembly ejection

RCCA ejection accident has been analyzed for HFP, 60% of nominal power, and HFP for BOC and EOC conditions. Effects of positive MTC of 5 pcm/k, for cycle 4, BOC conditions (possibility of switching to 18 months cycle) and influence of steam generator U-tube plugging level for cycle 6 have been considered. Standard assumptions (Westinghouse, RASP) have been used in our analyses. The assumptions for full power case are given in previous chapter. Thermal-hydraulic feedback is given via group constants dependence on fuel and coolant temperatures, the coolant density and the soluble poison density. The group constants for given range of thermal-hydraulic variables are generated using computer code LEOPARD and fitted using computer code SURFIT. This process is automated and enables fast calculation of reactivity coefficients using computer code FOS in 1D model and EXTERMINATOR-2 in 2D model.

Initial thermal-hydraulic conditions are determined for 10% plugging level using RELAP5/Mod1. Time sequence of events for RCCA ejection accident for NPP Risnj, cycle 6 BOC, HFP for 0% and 10% plugging levels are given in Table 2. Fission power space-time dependence for 0% plugging level, HFP is depicted in Figure 4. Clad surface temperature vs. axial distance and time is given in Figure 5.

<table>
<thead>
<tr>
<th>EVENT</th>
<th>PLUGGING LEVEL</th>
<th>0%</th>
<th>10%</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>t (sec)</td>
<td>value</td>
<td>t (sec)</td>
</tr>
<tr>
<td>Start of accident</td>
<td>0.000</td>
<td>-</td>
<td>0.000</td>
</tr>
<tr>
<td>Power range high neutron flux setpoint reached</td>
<td>0.026</td>
<td>1.18 Pn</td>
<td>0.027</td>
</tr>
<tr>
<td>Peak nuclear power</td>
<td>0.105</td>
<td>3221 MW</td>
<td>0.105</td>
</tr>
<tr>
<td>Min DNBR: 1.3 (hot ch.)</td>
<td>0.175</td>
<td>-</td>
<td>0.175</td>
</tr>
<tr>
<td>Rods begin to fall into core</td>
<td>0.527</td>
<td>-</td>
<td>0.527</td>
</tr>
<tr>
<td>Peak thermal power</td>
<td>0.700</td>
<td>2196 MW</td>
<td>0.700</td>
</tr>
<tr>
<td>Peak heat flux (W/cm²)</td>
<td>0.750</td>
<td>66</td>
<td>0.750</td>
</tr>
<tr>
<td>Max outlet steam quality (hot ch.)</td>
<td>1.750</td>
<td>26.0 %</td>
<td>1.750</td>
</tr>
<tr>
<td>Peak clad temperature (hot ch.)</td>
<td>2.050</td>
<td>1195 K</td>
<td>2.100</td>
</tr>
<tr>
<td>Peak fuel center temperature (hot ch.)</td>
<td>2.200</td>
<td>2891 K</td>
<td>2.200</td>
</tr>
</tbody>
</table>

Uncontrolled RCCA bank withdrawal

Uncontrolled RCCA bank withdrawal transient has been analyzed for HFP, 60% of nominal power, and HFP for BOC and EOC conditions, with speeds of 80 pcm/s and 5 pcm/s. Effects of positive MTC of 5 pcm/k for cycle 4 BOC conditions have been considered.

An introduction of OPDT and OTDT trips was necessary for modelling of slow transients. Constant inlet temperature, constant pressure and constant coolant flow were assumed. A loop modelling with point kinetics code FWR SIM (B) was performed in order to calculate effects of the rest of primary loop. The results show the adequacy of TRAWA-FEE assumptions that core inlet temperature is constant. The slow increase of system pressure obtained by PWRSIM indicates the conservative TRAWA DNBR values. Time sequence of events for uncontrolled RCCA bank withdrawal for negative MTC, HFP, and speed of 80 pcm/s is given in Table 3. Different DNBR correlations time dependence for 80 pcm/s HFP transient is shown in Figure 6. DNBR space-time dependence for 80 pcm/s HFP transient is shown in Figure 7. Figure 6. depicts cumulative CPU time and CPU time per step vs. transient time for 5 pcm/s withdrawal.
FIGURE 4: FISSION POWER SPACE-TIME DEPENDENCE

FIGURE 5: CLADDING SURFACE TEMPERATURE SPACE-TIME DEPENDENCE

CNS 10th ANNUAL CONFERENCE, 1989, 8-13
### Table 3

<table>
<thead>
<tr>
<th>EVENT</th>
<th>t (sec)</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial power</td>
<td>0.00</td>
<td>1876 MW</td>
</tr>
<tr>
<td>Power range high neutron flux setpoint reached</td>
<td>1.18</td>
<td>1.18 FN</td>
</tr>
<tr>
<td>Peak nuclear power</td>
<td>1.68</td>
<td>2327 MW</td>
</tr>
<tr>
<td>Rods begin to fall into core</td>
<td>1.68</td>
<td>-</td>
</tr>
<tr>
<td>Peak clad temperature (hot ch.)</td>
<td>1.68</td>
<td>620 K</td>
</tr>
<tr>
<td>Peak thermal power</td>
<td>1.76</td>
<td>1992 MW</td>
</tr>
<tr>
<td>Min subcooling outlet temperature</td>
<td>1.76</td>
<td>2.0 K</td>
</tr>
<tr>
<td>Min DNBR (hot ch.)</td>
<td>1.68</td>
<td>1.67</td>
</tr>
<tr>
<td>Peak fuel center temperature (hot ch.)</td>
<td>2.00</td>
<td>2401 K</td>
</tr>
</tbody>
</table>

---

**Figure 6: DNBR VS. TIME**

![Graph showing DNBR vs. Time](image-url)
FIGURE 7: DNBR SPACE–TIME DEPENDENCE

FIGURE 8: CUMULATIVE AND STEP CPU TIME VS. TRANSIENT TIME
Uncontrolled boron dilution

Uncontrolled boron dilution transient have been analyzed for H2P and HFP, BDC and EDC conditions. The HFP analysis of this transient has been performed for 0% and 10% plugging levels. We assumed that plant was operated under manual control. Deboration rate is calculated conservatively using specific NPP Krsko CVCS characteristics. Time sequence of events for uncontrolled boron dilution, for 0% and 10% plugging levels, HFP conditions, is given in Table 4.

<table>
<thead>
<tr>
<th>EVENT</th>
<th>0% t (sec) value</th>
<th>10% t (sec) value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Start of accident</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>OTT setpoint reached</td>
<td>37.60</td>
<td>43.00</td>
</tr>
<tr>
<td>Rods begin to fall into core</td>
<td>38.30</td>
<td>43.50</td>
</tr>
<tr>
<td>Peak nuclear power</td>
<td>38.30 2100 MW</td>
<td>38.30 2000 MW</td>
</tr>
<tr>
<td>Peak thermal power</td>
<td>38.30 2000 MW</td>
<td>38.30 2000 MW</td>
</tr>
<tr>
<td>Max. outlet steam quality (hot ch.)</td>
<td>38.30 7.2 %</td>
<td>38.30 8.1 %</td>
</tr>
<tr>
<td>Min DNBR (hot ch.)</td>
<td>38.30 1.72</td>
<td>38.30 1.65</td>
</tr>
<tr>
<td>Peak fuel center temperature (hot ch.)</td>
<td>38.40 2154 K</td>
<td>43.70 2173 K</td>
</tr>
<tr>
<td>Peak clad temperature (hot ch.)</td>
<td>38.40 621 K</td>
<td>43.70 622 K</td>
</tr>
</tbody>
</table>

The TRAWA-FEE results are in a good agreement with the results obtained by vendor and the data from RASP package.

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EVALUATION OF NEUTRON RESONANCE DATA AND ANALYSIS TECHNIQUES FOR CANDU TYPE FUEL BUNDLES

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ABSTRACT

We describe the evaluation of the magnitude of the differences when an approximate method is used for the treatment of resonance data -like WIMS method- and when different basic resonance cross-section data are used, for CANDU-Embalse type fuel rod bundles and core neutron calculations. The main conclusions are: the approximate resonance treatment in the WIMS-D/4 code over estimates the resonance absorption more for the interior fuel rods than for the exterior fuel rods in the bundle and under estimates the core reactivity by about 3%. Differences between the basic U data for the uncorrected and corrected values in the WIMS library compensated for the 3% difference.

INTRODUCTION

The group constants used as input in reactor calculations are affected by errors associated to different approximations and data adopted in cell calculations. The aim of this work is to analyze and to appraise the magnitude of the errors associated to different methods and data adopted for obtaining the effective group cross-sections of resonance nuclides inside the resonance region of energy for CANDU-Embalse type fuel rod bundles and core neutron calculations.

The resonance treatment - at the level of cell calculations - mainly affect the fast group cross sections i.e. the cross sections of the group 1 used in two-group diffusion calculations of the entire reactor core.

The most popular methods used in fast resonance treatment are those methods using equivalence theorems to find effective cross sections from resonance integrals of homogeneous mixtures with its different variations, and the multi-band method or sub-group theory. Besides, treatments with details in space and energy have been developed to perform exact calculations for testing fast methods and other purposes, like obtaining resonance integrals vs. background cross sections for homogeneous mixtures, used in fast treatments.

In this report, the impact of different treatments and data for resonance calculations on cell and core results for CANDU-Embalse type reactors is analyzed. A variety of validated computational codes, including some developed at the Bariloche Atomic Center were used in the evaluation.

First, cell calculations have been made to obtain two-group cell cross sections for diffusion core calculations using two methods and cross-section data, for resonance treatment. Then, two-dimension diffusion core calculations have been made for obtaining fast and thermal flux distribution and core reactivity.

MAIN CHARACTERISTICS OF THE FUEL ELEMENT

The main characteristics of the fuel element studied here are included in Table 1. All the data needed for calculations were extracted from Reference 1.

| Number of fuel rods per element | 36 |
| Fuel rod outside diameter, cm | 1.2 |
| Cladd thickness, cm | 0.418 |
| Fuel element spacing, cm | 28.575 |
| Cladding material | Zry-2 |
| Fuel material | UO₂ |
| Minimum fuel to cladd gap (BDL), cm | 0.00445 |
| | Pressure tube |
| | outside diameter, cm | 11.2064 |
| | thickness, cm | 0.4343 |
| | material | Zr,5%Nb |
| | Calandria tube |
| | outside diameter, cm | 12.8956 |
| | thickness, cm | 0.1397 |
| | material | Zry-2 |

COMPUTER CODES AND CROSS-SECTION DATA

Cell Calculations

The computer code adopted for obtaining the 2-group cross sections and diffusion coefficients for reactor calculations was WIMS-D/4 code (2) with its 69 group cross-section library.

Generation of Resonance Effective Cross Sections for Resonance Nuclides

Two methods were used: 1) the WIMS method included in cell calculations; 2) the method included in the following codes, developed at the Bariloche Atomic Center (3): RMET21 code, for the generation of basic microscopic resonance cross section from resonance parameters and for obtaining resonance integrals of homogeneous mixtures used in equivalent relations between homogeneous and heterogeneous systems; RMET22 code, for cross-section calculations in the resonance
region in cluster two-dimension geometry.

The method included in RMET codes is based on the integral transport theory, with details in space and energy.

Other programs used: SIGMA1 (4) Doppler-broadens basic evaluated cross-sections in ENDF/B format. Besides, other auxiliary programs were used e.g. for generation of cross-section data in the format corresponding to the RMET codes.

Core Calculations

DIPOBAR code (4): A FORTRAN IV programmed code for the solution of the multigroup bi-dimensional diffusion equation. The geometry is divided into homogeneous rectangular regions. The boundary conditions are implemented in the form of an albedo matrix. Axial buckling is given as a constant.

BORDE code (5): Albedo matrix calculations with the one-dimension multi-group diffusion theory. This code is used to obtain the albedos for simulating the core reflector. The method implemented in BORDE code is the Response Matrix Method. ANISN code (6): This popular one-dimension Sn code is used here for obtaining the two-group cross sections for reflector materials needed in BORDE code.

Basic Cross Sections for RMET Codes

The basic microscopic cross sections for $^{235}$U were generated with RMET21 code, using Breit-Wigner Formulae, with the resonance parameters taken from Reference 7. For $^{238}$U, the basic data were taken from ENDF/B4 (8). SIGMA1 code was used to Doppler-broaden the cross sections. These -punctual energy- cross section were used for two purposes: 1) generation of resonance integrals vs. background cross sections included in WIMS library; 2) calculations with details in energy and space with RMET22 code.

In this report, the basic cross sections used in RMET codes and the resonance integrals generated with those basic data with RMET21 code, are identified as (.5) data. For WIMS cell calculations, two different tabulations of resonance integrals were used (named (.4) and (.5) data). The values named (.5) are very similar to the values named (.2) in resonance data of WIMS library, i.e. the non-corrected resonance tabulations. The values named (.4) are the corrected resonance tabulations of WIMS library.

CALCULATIONS

The flow diagram of the calculation method used in this work is shown in Figure 1. For Beginning of Cycle 1 (BOC) condition, two resonance methods and data were used: - WIMS method - resonance integrals vs. $\sigma$, (.) data; calculations D4 and D2; - RMET22 method - resonance cross sections (.5); calculations MR. When comparisons between the two methods are made, WIMS method is named MW (that is the same calculation named D2).

The impact of different resonance treatments and cross-section data is evaluated on: 1) effective group resonance cross sections of $^{235}$U; 2) fast cell constants for core calculations; 3) thermal flux of each fuel element in the core; 4) normalized power of each fuel element; 5) core reactivity.
The core has 380 fuel channels. In this work there is only one bundle type. Control rods and burnable absorbers were suppressed for the calculations.

Other data: geometrical buckling, $7.618 \times 10^5$ cm$^2$; average fuel temperature, 290°C; average moderator temperature, 71°C.

RESULTS

Effective Group Resonance Absorption Cross Sections of $^{238}$U

For the same geometry as for WIMS cluster calculations, the effective absorption cross sections for the more important resonance WIMS groups were obtained with RMET22 code (groups 20 to 27 of 69 groups library). In Table 2 the results for groups 24, 25, 27, energy between 48.052 eV and 27.7 eV, 27.7 eV and 15.968 eV, 9.877 eV and 4.0 eV respectively (the main contribution) are shown.

TABLE 2: EFFECTIVE GROUP RESONANCE ABSORPTION CROSS SECTIONS FOR $^{238}$U

<table>
<thead>
<tr>
<th>WIMS group</th>
<th>Position</th>
<th>(MR) (barns)</th>
<th>(1-MW/MR) x100</th>
<th>(1-D4/D2) x100</th>
</tr>
</thead>
<tbody>
<tr>
<td>24</td>
<td>IR</td>
<td>2.693</td>
<td>.9</td>
<td>-1.6</td>
</tr>
<tr>
<td></td>
<td>ER</td>
<td>3.613</td>
<td>-.7</td>
<td>-2.6</td>
</tr>
<tr>
<td>25</td>
<td>IR</td>
<td>3.377</td>
<td>-.3</td>
<td>1.9</td>
</tr>
<tr>
<td></td>
<td>ER</td>
<td>4.732</td>
<td>-.2</td>
<td>.7</td>
</tr>
<tr>
<td>27</td>
<td>IR</td>
<td>4.185</td>
<td>-11.8</td>
<td>.5</td>
</tr>
<tr>
<td></td>
<td>ER</td>
<td>6.161</td>
<td>-5.2</td>
<td>.3</td>
</tr>
</tbody>
</table>

The maximum differences are found for WIMS group 27, the most important contribution, and between MR and MW (resonance treatment) and it is around 11% for the interior rods and 8% for the exterior rods in the cluster. The maximum differences between D2 and D4 (basic data) are less than 3%. The values obtained with MR calculations are in general smaller than the values obtained with MW calculations. This means that the approximate method of resonance treatment in WIMS code introduces an over estimation of the resonance absorption of $^{238}$U.

Cell Constants for the Fast-Energy Group (table 3)

TABLE 3: CELL CONSTANTS FOR THE FAST ENERGY GROUP ($[\Sigma]:$ cm$^{-1}$; $[D]:$ cm; $(-x:10^{-6})$)

<table>
<thead>
<tr>
<th>Calculation</th>
<th>$\Sigma_a$</th>
<th>$\nu\Sigma_f$</th>
<th>$\Sigma_{rem}$</th>
<th>D</th>
</tr>
</thead>
<tbody>
<tr>
<td>MR</td>
<td>1.628-3</td>
<td>8.501-4</td>
<td>7.450-3</td>
<td>1.502</td>
</tr>
<tr>
<td>(1-MW/MR)x100</td>
<td>-.4</td>
<td>.9</td>
<td>1</td>
<td>0.0</td>
</tr>
<tr>
<td>(1-D4/D2)x100</td>
<td>1.2</td>
<td>.7</td>
<td>-.1</td>
<td>-.2</td>
</tr>
</tbody>
</table>

CONCLUSIONS

The conclusions of this research are:

1) The approximate resonance data treatment in the WIMS-D/4 code underestimates the core reactivity by about 3% for a CANDU BOL core. Differences between the basic $^{238}$U data for the uncorrected and corrected values in the WIMS library compensated for the 3% difference, but does not account for the possible over estimation of the resonance absorptions in the data. When other data were used in these calculations greater differences were found (over-estimation of absorption);

2) The approximate method in the WIMS-D/4 code overestimates the resonance absorption more for the interior fuel rods than for the exterior fuel rods in the bundle. Although this difference may not be very
important for average integral results, it can significantly affect the prediction of actinide isotope production in CANDU fuel bundles;

3) The RHEMT codes can be used when more exact calculations are required by taking into account the specific details in the fuel geometry and the neutron energy distribution for the fuel resonance calculation.

This evaluation of the differences in the various sets of resonance data and calculational techniques has significantly improved our capability to predict accurately the fuel performance parameters of CANDU-Embalse type fuel bundles.

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THE SUDBURY NEUTRINO OBSERVATORY (SNO)

H.C. EVANS AND THE SNO COLLABORATION *
Queen's University, Kingston, Ontario

ABSTRACT

A group of Canadian scientists, in collaboration with colleagues from the United States and the United Kingdom, plans to establish a world class laboratory in INCO's Creighton mine near Sudbury, Ontario. The laboratory would be dedicated to the study of neutrinos from the sun and other astrophysical objects to advance our understanding of the physical processes which govern the properties of stars, as well as our understanding of the fundamental properties of matter. The laboratory would take advantage of two Canadian resources, namely access to one of the deepest mines in the western hemisphere and Canada's temporary surplus of heavy water.

INTRODUCTION

The SNO collaboration has proposed the development of a world class observatory for neutrino astrophysics. The observatory will make use of heavy water as the sensitive medium. Deuterium has properties which make it an ideal detector material and we plan to exploit the availability, on loan, of 1000 tonnes of heavy water from Canada's reserves. This observatory will contribute to astrophysics by clarifying the basic energy generating processes in the sun, and to nuclear and particle physics by determining fundamental properties of the neutrinos themselves. The appearance of supernova 1987A two years ago high-lighted the importance of neutrino detection in the understanding of stellar collapse. The sensitivity and versatility of the proposed detector would enable it to play an important role in studying a supernova occurring in our Galaxy.

A fundamental hypothesis in models of the evolution of the sun and stars is that energy generation takes place by nuclear fusion in their interiors. The only direct test of this hypothesis is to detect the neutrinos (v) produced in these reactions since neutrinos interact so weakly that they escape largely unimpeded from the solar interior. The sun is thought to be four or five million years old and has been formed by the mutual gravitational attraction of material from the initial formation of the universe and from supernova explosions of previous stars. As gravity caused the volume occupied by the constituent particles to contract, the temperature of the mixture rose and the energy available in the collisions between particles increased, until it became sufficient to sustain fusion reactions of hydrogen nuclei (protons) in the central region. The energy generated by these reactions, produces radiation pressure to counteract the tendency towards further contraction under gravity. A state of equilibrium was reached wherein the size and temperature of the sun, and the energy it radiates, are maintained at a steady level.

In astrophysical terms, the sun is a quite ordinary star, but because of its proximity it is uniquely accessible to measurement and observation, and plays a special role in defining our understanding of the stars. The standard solar model (SSM) for the sun has been developed, it depends on sophisticated mathematical theories of stellar evolution and makes definite predictions of the properties of the sun which follow from the laws of physics. The sun is in the hydrogen burning phase in which mass is converted into energy in a chain of reactions (pp chain) which may be summarized by

\[ p + p + p + p + ^3He + 2e + 2\nu + 26\text{ MeV}. \]

In the steady state, the rate of energy generation in the solar interior is equal to the rate at which energy is radiated from the surface of the sun (the luminosity). Charged particles and electromagnetic radiation interact strongly and the energy released in the centre of the sun takes approximately a million years to diffuse out to the surface. In contrast, neutrinos, travelling at the speed of light, reach the earth eight minutes after their production in the solar interior. Based on the observed solar luminosity, the total flux of neutrinos at the earth is expected to be \( 6.6 \times 10^{10}/(\text{cm}^2\cdot\text{s}) \), if the sun is indeed in a steady state. One branch in the pp chain which occurs with a relative strength of 0.01% is

\[ ^8B \rightarrow ^7Be + e^- + \nu + 14.6\text{ MeV}. \]

Although this branch is weak, it plays an important role in the evaluation of the SSM. It is the only source of solar neutrinos with energy above 2 MeV and the principal source of solar neutrinos detected until now. The flux of \(^8B\) neutrinos, which is strongly dependent on the temperature at the centre of the sun, is predicted to be \( 6 \times 10^9/(\text{cm}^2\cdot\text{s}) \) at the earth.

Neutrino Properties

Neutrinos are unique particles. They have long been believed to have no mass and so to travel at the speed of light. Recent attempts to measure the mass of the neutrino have set a limit of \(<10^{-9}\text{eV}^2\) where...
m_e is the electron rest mass. They carry no electric charge, but they spin around the axis defined by the direction of their motion. As a result of these properties the neutrino hardly interacts with matter at all, and is unique in that it interacts only through the so-called Weak Interaction. Three different types of neutrino have been identified and are known as the electron, muon, and tau neutrinos. As suggested by the names, they are associated with electrons (ordinary beta decay), muons and taus respectively. Beyond this the differences are not understood. It is possible that the three neutrinos have finite mass and that their masses are different. If neutrinos do possess a small mass then it is possible that they can change from one type to another as they travel through space. This process is referred to as neutrino oscillations.

Solar Neutrino Detection

A radiochemical experiment designed to detect solar neutrinos has been in operation in the Homestake mine in South Dakota since the late 1960's. The detector consists of a large tank containing 3.8x10^5 litres of liquid C_2Cl_2 in which the neutrinos interact with ^37Cl atoms to produce ^37Ar. The tank is purged periodically to collect and count the radioactive ^37Ar atoms. The measured ^37Ar production rate is 0.38±0.05 atoms per day above the known background. The SSM predicts a production rate of 1.1±0.1 ^37Ar atoms per day. Recently the Kamiokande II light water Cerenkov detector, built initially to look for proton decay, has been used to search for ^8B neutrinos. Again the predicted flux is not observed and an upper limit of about half the SSM prediction has been set. This discrepancy has come to be known as the solar neutrino problem.

The solar neutrino problem has puzzled physicists for almost two decades as more and more data has been accumulated. The physics of the fusion reactions in the pp chain and the efficiencies involved in the neutrino detection have been carefully checked, and are not believed to be the origin of the discrepancy. Two classes of solution to this problem have been suggested; either the model of the sun is deficient and neutrinos are not being produced in the expected quantities, or the neutrinos change in some way (e.g. oscillate) in transit to the earth and so are not detected in the chlorine experiment which is only sensitive to electron neutrinos.

New tools are required to solve the solar neutrino problem. The main deficiencies of the chlorine experiment lie in its inability to determine the energy of the neutrinos, their time of arrival at the detector and their direction. The Kamiokande II detector, which was not designed for the detection of solar neutrinos, is limited in their study by the background and sensitivity.

The observation of neutrino oscillations would have fundamental implications in particle physics. The existence of oscillations depends on (m^2), the squared difference in the mass between the neutrino types, and on a phase angle, $\phi$, which appears in the quantum mechanical theory. Thus it would be the first proof that at least some types of neutrino have finite mass.

The successful observation of neutrinos from SN1987A has demonstrated that new information on stellar collapse can be expected from a sensitive neutrino detector. (2) Although, the last visual observation of a supernova in our Galaxy prior to SN1987A occurred in 1604, most nearby supernovae are thought to be obscured from optical observation by interstellar dust. Neutrinos are not affected by this material and reasonable estimates of the number of stellar collapses within our Galaxy, based on the number of medium and massive stars and their expected evolution lifetimes, indicate that one may occur on the average every 10 years. A detector, sensitive to all types of neutrino, can be expected to provide important new information on stellar collapse. For example, the emission of neutrinos is expected to take place in a few tens of seconds and the detailed time structure of their detection would provide information for the development of models of stellar collapse.

THE SHO DETECTOR

The outline of the proposed detector is shown in Figure 1.

![Figure 1: Design of the proposed detector. The diameter of the cylindrical rock cavity is 20 metres](image-url)

It is located in a cavity 20 m in diameter by 32 m high, and consists of 1000 tonnes of D_2O, of enrichment greater than 99.85%, contained in an acrylic vessel 10 m in diameter and 14 m high. This vessel constructed from 6 cm thick panels, is surrounded by 4 m of high purity H_2O and 0.9 m of low-activity concrete. Mounted uniformly around the acrylic vessel, at a distance of 2.5 m, are 1960
photomultiplier tubes (PMTs) each 50 cm in diameter. The light sensitive photocathode area of the PMTs covers 40% of the area surrounding the acrylic vessel. The PMT array is sensitive to Cerenkov light produced by relativistic electrons and muons in the central regions of the detector. The electrons are produced by neutrino interactions and by background sources. Cerenkov light is emitted when a particle travels in a medium with a velocity greater than the velocity of light in the medium. The number of photons emitted is proportional to the energy lost by the particle and usually one photon strikes the photocathode of an individual PMT. Thus the number of PMT hits in the array is proportional to the energy, typically 10 MeV corresponds to 55 PMT hits. The direction of emission of the Cerenkov photons is correlated with the velocity of the electrons. The hit pattern and relative timing of the PMT pulses allows the direction of motion and position of an event to be reconstructed. The background due to cosmic-ray muons will be reduced to a negligible level by placing the detector in a cavity located at a depth of 2070 m in INCO's Creighton mine near Sudbury Ontario (Figure 2). The opening of a cavity

\[ d + \nu_e \rightarrow p + p + e^- \]  \hspace{1cm} (Reaction I)

corresponding to inverse beta decay of the deuteron. In this reaction, most of the kinetic energy of neutrino is transferred to the electron, the relation between the electron and the neutrino energy is \( E_e = E_\nu - 1.4 \text{ MeV} \).

In the second reaction neutrinos are scattered elastically at electrons:

\[ e^- + \nu_x \rightarrow e^- + \nu_x \]  \hspace{1cm} (Reaction II)

where \( \nu_x \) is any type of neutrino. The cross section for scattering the electron neutrino is six times that for the muon or tau neutrino. Reaction II is strongly directional and can be used to establish that the neutrinos come from the sun.

The total neutrino flux, independent of neutrino type can be measured by a third reaction:

\[ d + \nu_x \rightarrow p + n + \nu_x \]  \hspace{1cm} (Reaction III)

where gamma rays from subsequent neutron capture in the D\(_2\)O, or in a dissolved additive, are observed. Reaction III is equally sensitive to all types of neutrino.

The proposed SNO detector will have a sensitivity more than an order of magnitude greater than the chlorine experiment and will make use of the above three reactions to characterize the high energy neutrino flux. The expected reaction rates in 1000 tonnes of D\(_2\)O for an incident flux of \(6.0 \times 10^6\) (the SSM prediction) are given in Table 1. As examples,

<table>
<thead>
<tr>
<th>Reaction</th>
<th>Rate Above 6 MeV</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>19 Events/day</td>
</tr>
<tr>
<td>II</td>
<td>1.8 Events/day</td>
</tr>
<tr>
<td>III</td>
<td>6 Events/day</td>
</tr>
</tbody>
</table>

TABLE 1: NEUTRINO REACTION RATES IN 1000 TONNES OF D\(_2\)O

observations that could lead to possible resolutions of the solar neutrino problem are outlined below. If the high energy neutrino flux from the sun is less than that predicted by the SSM, the rate for all three reactions will be reduced from the values in Table 1. On the other hand, if the SSM is correct and oscillations occur, then the rate for reactions I and II would be reduced while that for reaction III would remain at the value given in Table 1.

Backgrounds

The success of SNO will depend on the level of background obtained and the ability to determine the background. Although the total reaction rate in the SNO detector is expected to be an order of magnitude greater than in the chlorine experiment, extreme care will be required in shielding the detector from all sources of radiation. The cosmic-ray background due
to throughgoing muons is expected to be about 25 events per day at a depth of 2070 m. Although comparable to the expected neutrino event rate, muons can easily be distinguished from neutrino induced events and rejected. The naturally occurring radioactivity (232Th, 232U and their daughters) in the rock and in the materials used to construct the detector is an important source of background because beta and gamma rays from their decays can produce Cerenkov light. The design calls for the D2O to be surrounded by increasingly pure layers of material as one progresses inwards from the rock. The concentrations of 232Th and 235U, in equilibrium with their decay products, that are assumed for the design are listed in Table 2. A large part of the development work that has been done on the detector has concentrated on identifying materials of the required purity. The low background concrete will be sulfurcrete in which dolomite from Haley Ontario is the aggregate and sulfur is the binder. A program to identify low activity glass and to establish that it can be used successfully in the manufacture of PMTs has been undertaken. The H2O and D2O will be circulated through special filters designed to reduce thorium and uranium to the required concentrations. New techniques of improved sensitivity for the measurement of residual concentrations of 232Th and 235U are being developed. The progress to date indicates that the required concentrations can be achieved and measured.

The response of the detector to neutrinos and to the background radioactivity, which is outlined above, was determined by extensive Monte Carlo calculations. Comparisons between the performance calculations and the experimental data for existing H2O Cerenkov detectors have been made. The calculations indicate that the energy resolution for 7 MeV electrons is 18%, that the position of the reaction can be reconstructed to about 70 cm and that incident direction of the neutrino can be determined to within 25 degrees. It is concluded that the proposed SNO detector has the sensitivity, signal to background ratio and energy resolution to elucidate the solar neutrino problem.

TABLE 2: 232Th AND 235U CONCENTRATIONS ASSUMED FOR MATERIALS

<table>
<thead>
<tr>
<th>Material</th>
<th>235U Concentration (g/g)</th>
<th>232Th Concentration (g/g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>D2O</td>
<td>1.1x10^-1</td>
<td>1.1x10^-1</td>
</tr>
<tr>
<td>H2O</td>
<td>1.5x10^-1</td>
<td>2.2x10^-1</td>
</tr>
<tr>
<td>Acrylic</td>
<td>3.6x10^-12</td>
<td>1.9x10^-12</td>
</tr>
<tr>
<td>Glass (PMTs)</td>
<td>3.0x10^-7</td>
<td>3.0x10^-7</td>
</tr>
<tr>
<td>Sulfurcrete</td>
<td>0.9x10^-8</td>
<td>1.1x10^-8</td>
</tr>
<tr>
<td>Norite</td>
<td>1.2x10^-6</td>
<td>3.3x10^-6</td>
</tr>
</tbody>
</table>

PRESENT STATUS

A detailed proposal for the Observatory, including the scientific case, design and budget was submitted for peer review in October 1987. (3) The request for capital funding was for $35.39 million Canadian. It was sent to the Natural Sciences and Engineering Research Council, the National Research Council of Canada, the U.S. Department of Energy, and the United Kingdom Science Research Council. The proposal was reviewed by an International Scientific and Technical Review Committee which reported in June 1988. (4) The committee made the following recommendations:

1. It is the unanimous decision of the Scientific and Technical Review Committee to recommend strongly that the Sudbury Neutrino Observatory be approved and that it be funded as proposed.

2. In the opinion of this committee, the time constraints imposed on the guaranteed loan of the heavy water require that the funding agencies make every effort to ensure that SNO is approved and funded at the earliest possible date.

3. The committee considers that the interim funding of $128 K, which is required immediately for the confirmation of the cavity location, is essential to the project and should be made available now.

The SNO proposal and the report of the Scientific and Technical Review Committee has been reviewed by the funding agencies and funding of the project is under consideration.

The SNO collaboration has continued to refine the design of the detector to optimize the detector performance and increase the understanding of background effects. New developments since the original proposal include the development of a spherical design for the acrylic vessel which should allow the thickness of the acrylic to be reduced by a factor of two. The design, which is shown in Figure 3, would make use of two H2O levels to partially support the filled acrylic vessel by floatation. The possibility of using a larger number of smaller PMTs, which have better timing and would allow improved...
spatial reconstruction and background rejection, is under study. Work is continuing on the identification of detector component materials with sufficiently low radioactivity. We are confident that the SNO detector can be built and that it will make a major contribution to the solution of the solar neutrino problem and to our understanding of the properties of neutrinos.

REFERENCES


ON FACTORS CONTRIBUTING TO QUALITY OF
NUCLEAR CONTROL COMPUTER SOFTWARE

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Ontario Hydro

ABSTRACT

Safety related computer software has increasingly come into focus in the software engineering field over the past decade. This paper describes how Ontario Hydro has addressed the software industry concerns in the methodology used for designing and implementing the unit control computer software for the new Darlington Generating station.

The cornerstone of the methodology is a software quality assurance (SQA) program, which was initially set up to cover only the software development portion of the software life cycle, but which is now being extended to cover the entire software life cycle, including commissioning, operation and maintenance of the software.

1. INTRODUCTION

Good quality and reliability of each critical part of a nuclear generating station are essential for the safe and efficient operation of the plant. The control computer software is no exception. Consequently, Ontario Hydro put in place a program for the new Darlington Generating Station, which attempts to achieve the best possible quality for the control computer software. This has been done by implementing a Software Quality Assurance (SQA) Program as an integral part of the design process. This paper discusses the SQA program, how it was conceived and implemented, how it works in practice, how it is believed to enhance the quality of the resulting software product, and what other perceived benefits are.

2. SOFTWARE QUALITY ASSURANCE PROGRAM DEFINITION

A Software Quality Assurance Program can be defined as a planned and systematic process, described through specific procedures, for how to effectively implement the functional requirements in the final software product, thereby achieving acceptance of the system by the user community and third party stakeholder, such as regulatory bodies etc.

It is essential that such a program be defined prior to the start of any project development work and that commitment to the application of the program is re-affirmed by management periodically throughout the product life cycle.

3. SQA PROGRAM INCEPTION

Large software projects, especially those for safety related software, are difficult to manage and control. Problems, such as large cost overruns and failure to meet the design requirements, are not uncommon. Some of the specific difficulties associated with large software projects are as follows:

Firstly, the translation from original requirements to software specifications and subsequently to actual program code is difficult. Typically, the requirements are specified by designers from non-computer disciplines. Often, they have not taken into account the capabilities and features, as well as the limitations, of the computer system. Conversely, the computer designers usually do not have in-depth familiarity with the other disciplines, such as process control and human factors engineering, from which the application requirements originate. They may therefore sometimes partly misinterpret the requirements and introduce functional errors in the resulting software design.

Secondly, in a large real time computer system the interactions between various computer programs are generally complicated. Consequently, it is difficult to build and to revise the software. Implementing localized revisions without affecting other parts of the system is especially difficult and revisions must therefore be very carefully controlled.

Thirdly, due to system complexity it is difficult and time consuming to remove all residual errors in the software and, by results of testing, demonstrate that the final product is highly reliable and meets the functional and computer system specific requirements.

Additionally, at the beginning of the project it was recognized by the Darlington Control Computer Software design group that the software would be larger in size and more complex than that of previous Ontario Hydro nuclear stations. Some of the reasons for this were:

- additional functionality, especially in the area of operator interface, but also more control and monitoring programs were to be included.
- computer hardware with larger memory and memory management.
- computer software with a modified commercial operating system and not just a special purpose executive.
- implementation language still assembler and not a high level language, in order to ensure fast enough program execution for all programs to run reliably in real-time.

The SQA program for the Darlington project has attempted to address these difficulties by specifying how the software system should be designed, built and revised, how the requirements should be verified and how the problems (uncovered...
development methodology and the associated organizational structure, a stepwise software development methodology and the associated procedures for the designers and project management personnel to follow. The SQA program, although not in place before the start of the project, was put into effect during the Requirements Definition/Preliminary Design phase. It augments the general Quality Engineering Program applicable to all engineering disciplines on the Darlington Project. However, the software engineering design and development process includes activities additional to those of the other engineering disciplines in that the design is also implemented (in computer code) in house and not just implemented by acquiring manufactured components which are described by documentation, drawings and bills of material. Because of the greater scope of design, additional quality engineering procedures specific to software were clearly required.

4. SQA PROGRAM IMPLEMENTATION

The Darlington SQA program has been implemented in accordance with the Software Quality Assurance requirements defined during the Requirements Definition Phase. Many implementation details were based on the IEEE Software Quality Assurance standard at that time available only in draft form.

4.1. Project Organization

![Software Project Organization Diagram]

The SQA Program clearly defines the tasks and responsibilities of each position in the organization. Procedure compliance reviews have been carried out during the course of the project to ensure that all duties were being adequately fulfilled. In order not to compromise the quality of the end product, special emphasis has been placed on maintaining independence between the various parts of the organization throughout the "Software Development Life Cycle." As illustrated in Figure 1, the organization consists of three units - two design units and one configuration management and testing unit. Each unit is further divided into teams. Each team typically consists of 2-5 members and is headed by a team leader, who has well defined design, verification or other responsibilities. The team size was deliberately chosen to be small in order to enhance the team leader's ability to fully understand the work tasks of his team members and in order not to cause communication breakdown. A few times during the course of the project teams have had to grow to beyond the recommended size. This has made the team leader unable to fulfill all his duties and partly lose control of the team's activities.

4.1.1. System Software Design Unit. This unit is responsible for the System Software which provides the necessary services and support for the Application Software. This unit consists of the following teams:

- Operating System: This team is responsible for the Executive Software (such as program scheduling, program synchronization), drivers (the software which interfaces with the hardware, such as analog inputs and outputs), checking of system integrity (such as providing programs to perform checking of the computer hardware).
- Man-Machine Interface: This team is responsible for providing the necessary software for implementing operator interface functions. These consist of graphical display functions (such as bar charts, trends and process flow diagrams) and the processing of alarm annunciation messages.
- Common Data: This team is responsible for providing common subroutines and data (such as conversion data - e.g. from analog input values in raw counts into engineering units, such as temperature and pressure).

4.1.2. Application Software Design Unit. The function of this unit is to develop application programs. It consists of the following teams:

- Control Software: This team is responsible for control program design and snag clearance. Primary Heat Transport Control and Reactor Control are typical examples.
- Monitoring Software: This team is responsible for monitoring software design and snag clearance. Gamma Radiation Monitoring and Generator Temperature Monitoring are typical examples.

It is also worth noting that all design teams carry out planned and documented testing of their software. The results of this testing are later verified by an independent part of the organization, the Integration Testing Team.
4.1.3. Configuration Management and Testing Unit. This unit is responsible for managing the software system and for conducting verification and validation of the work produced by the other two units. This unit consists of the following teams:

Configuration Management: This team is responsible for managing the software configuration, such as periodically rebuilding the software system by including software from the two design units. This team is also responsible for the software library. The function of the software librarian is to keep track of various software revisions, documentation and software problems, referred to as "snags". For further details regarding configuration management refer to subsection 4.3 below.

Integration testing: The prime responsibility of this team is to perform independent testing of the software produced by the design teams. The testing focuses on the interfaces between the individual programs and the software subsystems. Examples of software subsystems are Operating System, Display Programs, Alarm Annunciation Programs and Control Programs.

System Testing: This team is responsible for verifying the overall performance of the software system. In the earlier stages of software development, various studies (such as bottle-neck analysis) of the critical resources (such as CPU, memory and disk usage) were carried out. In the later project stages prior to system turn over, this team was also responsible for performing overall functional testing of the system. These tests employed a real-time simulation model of the plant processes and served as validation of the system against the formal requirements to ensure that the correct functions were being performed. As many as possible of the important system functions were tested prior to delivery of the software to the station. Not all functions could be tested by the System Testing Team at this stage, since the test bed did not comprise a complete target hardware configuration and was not connected to the real plant processes.

Field support: This team is responsible for providing the necessary liaison with the user. The teams primary functions are to log and help investigate problems reported by the user and to assist the user during commissioning.

Software shipment: This team is responsible for coordinating all the work necessary for shipping the software to the user. This includes packaging, delivery and completely documenting the configuration of the shipped system, i.e. with respect to which of the reported software snags have been cleared by this shipment.

4.2. Software Life Cycle

The key to controlling the software creation process successfully lies in understanding and controlling the Software Life Cycle, which is the total period of time which starts when the software product is conceived and ends when the product is no longer used.

The subset of the Software Life Cycle, which ends when the software is turned over to the user, is often referred to as the Software Development Life Cycle.

4.2.1. Software Development Life Cycle. The Software Development Life Cycle is the phase during which the software is designed and produced. The various stages of the Cycle are illustrated in Figure 2. The stages within the development process are basically sequential. However, in practice, there has been iteration between one stage and another as the development process, in this case, has spanned over a period of more than eight years. Iterations also occurred because the original requirements for software were often too loosely defined and consequently constantly had to be refined, requiring extensive changes. Documents are produced at the end of each stage and are revised periodically as more detailed information on the overall plant design becomes available or changes. Each of the stages is outlined briefly below.

**Verification** - the process of ensuring that the product of each phase of the software development process fulfills all the requirements imposed by the previous phase.

**Validation** - the testing and evaluation of the integrated computer system to ensure compliance with the overall functional, performance and interface requirements.

![FIGURE 2: SOFTWARE DEVELOPMENT LIFE CYCLE (SHOWING DOCUMENTS PRODUCED AT EACH STAGE)]
Requirements Definition Stage: During this stage, functional requirements are prepared by the interfacing project disciplines, mainly Process Control and Human Factors Engineering. As a result of these functional requirements, the computer designers define hardware and overall software system requirements. During this stage, software quality assurance requirements are also prepared, which are subsequently used to prepare the specific SQA program procedures.

Preliminary Design Stage: During this stage, existing requirements are translated into a precise "Subsystem Implementation Specification" of the software (or Software Requirements Specification). The subsystem is further structured into modules by specifying the function of each module, the hierarchical structure of the module and the interfaces between the modules. A module, in this case, is equivalent to a complete real-time program, task or function. It is typically of a size equivalent to 3000 source code instructions and is described in Preliminary Pseudo-code (an English-like structured language). The purpose of this type of pseudo-code is to only give a very high level description of the program logic for a particular module. It does not consider error conditions and must be independent of specific hardware and operating system software.

Detailed Design Stage: The top level software modules defined in the Preliminary Design Stage are further decomposed into smaller low-level modules described in Detailed Pseudo code to facilitate the management of their implementation. The size of each module is equivalent to approximately 80-300 source code instructions. The specific requirements for each low-level module are derived from the requirements specified for the parent top-level modules. Error diagnosis and error handling capability to insure the operational integrity of the system are built into these low-level modules. At this stage, detailed data structures are also defined. The design of the low-level modules is verified and approved by the Team Leader. Verification in this case means checking compliance to all applicable guidelines, standards and procedures, e.g. coding guidelines.

Coding and Debugging Stage: The coding and debugging activity begins after the draft "Detailed Design Description" for a particular system has been completed and verified. The actual program codes are produced and debugged at this stage. The debugging, also known as unit testing, is "white box" in nature, since it is mainly the internals of the software that are tested. Unit Testing is performed by the designer and no resulting formal documentation is required.

Development Testing Stage: At this stage, the low-level modules are integrated together by the designer and tested to ensure that groupings of modules perform the functions of the system as specified. This testing may be termed "grey box", since both the software internals and the external functionality are tested. It is performed by the designer and witnessed by the team leader. Formal documentation is produced which, together with test programs and execution procedures, is transmitted to the Configuration Management Team. The software itself is subsequently integrated into the system. From this point on the software is under "Configuration Management" and said to be "frozen", i.e. it cannot be changed without following detailed, rigorous procedures.

The remaining stages of the Software Development Life Cycle, namely, Integration, Integration Testing and System Testing, are part of Software Configuration Management. They are discussed in more detail below, under "Software Configuration Management" and "Design Verification and Validation".

4.3. Software Configuration Management

Software Configuration Management defines how the software system should be built and revised.

A project's software configuration is the state of its components and the way in which these software components are arranged and interfaced with each other. In a large project these software components are parts of an evolving software product, which must be carefully and methodically controlled in order to maintain its integrity. This methodical control is known as Software Configuration Management (SCM). Figure 2 illustrates the Software Configuration Management process in relation to the rest of the Software Development Life Cycle.

The key concept of Darlington's Software Configuration Management is the Software Design Freeze (SDF), which takes place after the software designers have completed all of the activities of the development and production phase of the Software Development Life Cycle for a particular program or software subsystem. A SDF is a scheduled, formal turnover point of the software from the design units to the configuration management and testing unit, for which the designers must prepare their software as "freeze packages" with strictly controlled contents and format. Each freeze package contains the following:

- A cover sheet (software submission form) outlining the software submitted;
- An automatic procedure (called an indirect command file) for assembling/compiling the software;
- All the programs, test programs, documentation and other necessary information which the designers have prepared, so that they can be checked and retested by the Integration Testing Team.

4.3.1. Software Integration - Building and Rebuilding the System. When the freeze packages are submitted to the Configuration Management (CM) Team, the CM Team verifies the contents and formats of all freeze packages. They then build a new version of the software based on the previously frozen version and the latest freeze packages and release the system for testing (Integration Testing and System Testing) and subsequent use, either by the software developers or by the users in the field. Once the software packages are turned over to the CM Team and have been incorporated into the system, they are considered as "frozen". Changes to "frozen
Software Deficiencies: The existing software may be found to be inadequate or incorrect by design, implementation, assumption or for other reasons.

Hardware Changes: Problems with hardware components and interfaces may yield a solution only through software modifications.

Requirement Changes: The ground rules for the software system's operation as designed may be modified for whatever reason.

To ensure the traceability of the changes and that changes are made in an orderly manner, changes are requested via a Software Change Request (SCR) form. The SCR is initiated by the software team leader and approved by the supervisor. It contains specific information about the changes (e.g. which software modules need to be changed), as well as the estimated number of man hours and the scheduled completion date. The SCRs are submitted to the librarian, who subsequently assigns to each one a SCR number and keeps all relevant information in a database. These changes to the existing software are also submitted for software freeze. In this case, the software freeze package contains the following:

- A cover sheet (software submission form) listing the software modules changed and the corresponding SCR numbers;
- An automatic procedure (called the Correction File) for automatically revising the existing module using utility software packages supplied by the computer manufacturer.

All software changes are required to be fully documented. To provide complete traceability, the following steps are taken:

- Program source: Only the changes to the program source are submitted (i.e. no resubmission of the source) using the correction file facilities provided by the computer manufacturer. The correction files are used to modify the software according to defined rules, which include the following:
  - Correction identification: which defines the Audit Trail ID and specifies the purpose of the change, the author of the change and the date the change was made.
  - Audit trail: which causes the Audit Trail ID to be printed in the listing where the change was made. This facilitates tracking of the changes.

4.4. Design Verification and Validation

To ensure that the final software product meets the design requirements and intent, the SQA Program requires that the following be performed:

- Design walkthroughs;
- Design reviews;
- Software testing

4.4.1. Design Walkthroughs. To ensure that the final product meets the functional design requirements prepared by the non-computer designers, design walkthroughs are carried out. From these functional requirements, software specifications are prepared and the associated high level program logic is developed by the software designers using an English like high level language, pseudo-code. The pseudo-code has been chosen as the walkthrough medium because it allows the software designer to present his software logic in a manner that can be understood by non-software designers. An official walkthrough meeting is conducted and the results are documented.

A design walkthrough is required to be carried out for each major revision of the program requirements.

In addition to the formal design walkthroughs between the application requirements specifiers and computer software designers, there is another ongoing informal walkthrough process within the computer discipline, namely peer review of the program code (code reading). The purpose of these peer reviews is to ensure that the requirements have been correctly translated into software code and to confirm that the code satisfies the technical level that is required by the project. Although the reviews are not formal, the team leaders are responsible for ensuring that all valid comments are addressed and all differences of opinion are adequately reconciled with resulting corrections to the program code if necessary.
4.4.2. Design Reviews. Design reviews are conducted by a technically competent, but organizationally independent, part of the company. In order to achieve as broad a perspective and thorough a review as possible, the review team is made up of members of both internal departments and external organizations such as Operations, In-service Nuclear Departments, Atomic Energy of Canada Ltd and GE of Canada.

Historically, a preliminary design review was held during the preliminary design phase of the project. The purpose of this design review was to verify the appropriateness and soundness of the conceptual design. Two detailed design reviews, one for system software and one for application software were conducted before the first shipment of software to the station. Their purpose was, similarly, to verify the appropriateness and soundness of the detailed design as implemented. In addition to the above, a post-operational design review is also scheduled.

4.4.3. Software Testing. Software testing is an iterative process and has two main purposes: 1) to detect errors in the software and 2) to ensure that the final product meets the original design requirements. To improve effectiveness (i.e. to detect problems as early as possible), software testing starts in the software development stage prior to software freezes. There are four levels of testing, two of which are not part of Software Configuration Management (SCM). They are Unit Testing and Development Testing and are described in subsection 4.2.1. The two levels of testing that do belong under SCM are described below:

Integration Testing: This testing is carried out to verify correct operation of the interfaces between the major programs and software subsystems. It is therefore mainly "black box" in nature. The tests are performed by the Integration Testing Team after a new software version has been integrated and built by the Configuration Management Team and declared to be frozen. The references for this level of testing are the Software Implementation Specifications. Formal test plans, test results and summaries are also produced. The test plans are periodically reviewed for consistency and test coverage by an independent reviewer.

System Testing: This testing is carried out on the entire software system and must be classified as "black box" testing. It is performed by the System Testing team. The references for this level of testing are the design requirements. Just as for Integration testing, formal test plans, test results and summaries are produced.

4.5. Problem Reporting and Disposition

4.5.1 Reporting. Problems found in "frozen software" are documented by SNAG reports. A snag report can be prepared by anyone using the system and is not restricted to the testing teams. Once a snag report is filled out, it is passed on to the Integration Testing Team for verification. The verified snag is forwarded to the appropriate design team for concurrence. A snag report contains the following information:

- Problem or deficiency description;
- Verification and concurrence information;
- Cancellation information.

4.5.2 Disposition. If the design team concurs with the snag as documented, the team leader will fill out a Software Change Request (SCR) form and the normal change control procedure will be followed. If the design team rejects the snag, reasons must be given for its rejection as described on the snag form. It is then returned to the originators for their review and approval.

Occasionally, when correction to a problem is so urgently needed (such as during commissioning) that the user cannot wait for the completion of the formal change control cycle, a patch is provided. Patches are used to temporarily resolve by-pass problems and are sections of program code (usually small) which are superimposed on the already installed code. The implementation of the patches is coordinated by the field support team. Although patches can be prepared by anyone, they have to be reviewed and approved by the original design team. Testing is then performed by the Integration Testing Team prior to the formal installation of the patches. The patch information and status (e.g. installed, removed) are strictly maintained by the software librarian, so that the complete configuration of the installed and operating software is always known and is traceable over time. The patches are incorporated as corrections in the subsequent software freeze.

4.6. Extensions to the SOA Program

At the time of writing the Darlington Control Computer Software is being commissioned at the station. This phase of the project is considered to be an extension to System Testing (part of the Software Development Life Cycle), since the software is now finally being tried out on a complete target hardware configuration and under actual operating conditions. As a result of this, procedures have been added to the SOA program to handle such aspects as problem reporting by the station, incorporation of corrections to these problems and general support to the station by the Field Support Team.

At this time extensions to the SOA Program are also being prepared for the Post Software Development Life Cycle activities, i.e. procedures for how to maintain the software during its operational life at the station. In addition to the SOA procedures themselves, plans are being developed for the actual software maintenance facilities, both hardware and software, to be used during the lifetime of the station.

5. SOA PROGRAM VERIFICATION, CORRECTIVE ACTION AND MAINTENANCE

Approximately two years after the SOA Program was first launched, a separate procedure for "Corrective Action and Maintenance of the Darlington SOA Plan" was issued. This was in direct response to a recommendation from the 1983 Audit of the Darlington SOA Program conducted by the Quality Engineering Department. Further details of this audit are described in subsection 5.1.3 below.
This separate procedure describes, in detail, activities to verify the appropriateness and effectiveness of the SQA Program, as well as the required corrective actions and maintenance activities. Detailed responsibilities for these activities are also included in the procedure.

5.1. Verification Activities

The verification activities for the SQA Program itself that are being carried out on an ongoing basis are described below:

5.1.1. Periodic SQA Program Review. The Darlington SQA Manual, which describes the SQA Program by way of detailed procedures, has a designated document maintainer. It is the responsibility of this individual to review, at least once annually, the SQA Manual to determine if the procedures meet the current SQA Program Requirements and if they reflect the current Software Development or Maintenance Activities. If any discrepancies are detected, corrective action must be taken according to the Corrective Action and Maintenance procedures described in subsection 5.2 below.

5.1.2. Adequacy and Effectiveness Monitoring. The intent of this monitoring is to ensure that the SQA procedures, program coding standards, etc., as well as the software tools and methodologies being observed and used consistently on an ongoing basis, and that continued effectiveness of the SQA Program is being achieved.

Part of this monitoring is carried out by the software supervisors periodically, based on need, and takes the form of short interviews of randomly chosen members of the project organization. The process consists of questions testing the conversancy of the members with the procedures and standards of the SQA Program. If any member is found to have insufficient working knowledge and understanding of the program he or she is given additional training over and above that which is normally provided to all members of the software project organization.

The other part of the monitoring takes the form of an annual SQA Program status and adequacy review by the Senior Design Engineer and the Software Supervisors. A report summarizing the review findings are then issued to project management.

5.1.3. Quality Audits. Although not part of the Corrective Action and Maintenance Procedures, SQA audits may be conducted at any time by either internal or external stakeholder organizations.

So far two procedural quality audits have been conducted on the Darlington SQA Program by Ontario Hydro's Quality Engineering Department, both during the detailed implementation phase of the project, but separated by two years. Both of these audits resulted in a few findings and a number of recommendations, all of which were detailed in nature and described either non-compliance between procedures and practices or additional or modified recommended practices.

All findings and recommendations have been resolved and the main lesson learned by the project personnel from these exercises has been that in order to maintain an effective SQA Program, in-depth familiarity with the program by the project team, as well as direct correspondence between the program procedures and project practices, must be confirmed on an ongoing basis.

5.2. Corrective Action and Maintenance Activities

Provisions have been made in the separate Corrective Action and Maintenance Procedures of the Darlington SQA Plan for the reporting of deficiencies in the program for subsequent corrective action. All users of the program procedures have a responsibility to report perceived problems on a "SQA Plan Deficiency Reporting Form". All deficiencies including the following types must be reported:

- Manual deficiencies - errors in the manual
- incomplete documentation of a procedure
- lack of procedure in a certain area.

Compliance deficiencies - procedure documented in the SQA Manual, but not followed in practice.

Subsequent to the submission of a deficiency report it is the responsibility of the SQA Manual maintainer in cooperation with the Senior Design Engineer to resolve the documented problem according to a mutually agreed schedule.

6. QUALITY RECORDS

A number of documents produced throughout the Software Development Life Cycle are permanently kept as quality records to be retrieved at any time during the life cycle of the software product. These records include:

- Design Requirements
- Design Documentation (general and detailed)
- Program Listings
- Software Submission Forms
- Software Snag Reports
- Software Change Request Forms
- Integration Test Record Book
- System Test Record Book

The purpose of these quality records is to provide evidence that the produced software complies with the SQA procedures as well as applicable codes, standards and user requirements.

7. CONCLUSIONS AND SIGNIFICANT EXPERIENCE

Over the past ten years, Ontario Hydro has had direct experience with the application of a SQA program for the Darlington Control Computer Software project.

The project team strongly believes that a number of benefits have been derived from the application of such a quality program, although no statistics have been produced to prove this claim.

First of all, we believe that such a program builds quality into the final product through a disciplined and structured development process. By
requiring independent review and verification throughout the various stages of the process, it enables successful validation of the end product against the requirements. Systematic validation is becoming increasingly important for safety critical and safety related software, as third party reviewability, especially by licensing bodies, is now the norm.

It is difficult to show quantitative savings from a development process such as the one described here, since no direct comparison to an alternate development method for the same project is feasible. However, the project group believes that the application of the described quality program has not only impacted positively on the quality of the end product, but has also enabled them to:

- Maintain schedules in spite of high staff turnover. To illustrate this point, it took only approximately 8 years for the equivalent in numbers of complete staff turnover to take place (approximately 30 people). In 1986 alone the staff turnover was 5.
- Give an early demonstration of actual system performance.
- Respond to design changes and detail errors and to re-implement with minimal schedule impact.

With respect to design changes, one significant experience and conclusion drawn from the software development process on this project over the last 10 years is that control of requirements is essential to both good quality of the software end product and to minimizing development costs. Changing and migrating requirements often cause ripple effects through large parts of the software system and may introduce hard to detect errors. As a minimum this means significant, costly rework of the software, including extensive re-testing.

The major source of requirements instability on this project has been that the part of the organization which originally specified the application requirements (process control and human factors engineering designers) was a different group from the end user organization, the operations staff of the station. This in itself would have been an acceptable situation as long as operations staff had been assigned to the project and available in force to review and comment in detail on the application requirements generated by the design arm of the organization. However, on this project the end-user group was not available at an early date and a certain amount of rework is now, in the commissioning phase, being carried out as a result.

Nevertheless, as compared with previous practice, which extends over 20 plus years in the application of control and monitoring computers to generating stations, the described project, utilizing a quality program from early in the development process, has been easier to manage and is expected to provide a more maintainable and reliable product for long-term support.

In final conclusion, the Darlington SQA Program has been an integral part of the software design process throughout the Software Development Life Cycle and, as such, has undoubtedly been our most valuable tool for designing and producing quality software in a timely manner. This has been proven by the design group's achievements over the past several years and, we believe, will be equally important during the operation and maintenance phases of the Software Life Cycle.

REFERENCES:


(2) "ANSI/IEEE Standard 730-1984 Software Quality Assurance Plans"

GESTION DU COMBUSTIBLE INTEGREE

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RESUME

A la centrale nucléaire Gentilly 2, grâce à un effort d'intégration soutenu de la part du propriétaire/exploitant Hydro-Québec, le terme GESTION DU COMBUSTIBLE a pris une définition large qui englobe toutes les activités reliées au cycle complet du combustible nucléaire. Le personnel assigné à ces activités est organisé sous une seule unité de travail afin d'obtenir une efficacité optimale. Le réseau informatique intégré qu'utilise ce groupe relie programmes et banques de données situés au site, à l'intérieur, ainsi qu'à l'extérieur de l'entreprise. Ceci a pour effet d'éviter les erreurs et les efforts non-productifs reliés à la duplication ou aux entrées manuelles de données dans les divers codes. La décision d'adopter l'intégration comme voie vers une meilleure gestion du combustible et l'engagement de la poursuivre à mesure que de nouveaux moyens se présentent ne fut jamais regrettée et a contribué grandement aux excellents résultats obtenus jusqu'à ce jour dans ce domaine à Gentilly 2.

INTRODUCTION

L'objectif principal poursuivi par Hydro-Québec dans son effort d'intégration de la gestion du combustible à la centrale nucléaire de Gentilly 2 est:

En ce qui a trait au cycle complet du combustible nucléaire,

1) regrouper dans une seule unité de travail tout le personnel pertinent à cette activité,

2) regrouper dans un seul réseau informatisé tous les outils nécessaires à cette activité,

le tout en définissant le terme "cycle complet du combustible nucléaire" le plus largement possible en y englobant toutes les activités suivantes:

- Activités pertinentes à l'approvisionnement de concentré d'uranium, l'affinage de ce dernier ainsi qu'à la fabrication de grappes de combustible.

- Activités pertinentes à la gestion des stocks d'uranium hors-site, ainsi qu'à la gestion des stocks de combustible neuf et usé au site même.

- Activités pertinentes à la gestion du combustible sous irradiation en cœur.

Les bénéfices récoltés suite à une telle intégration sont abondants. Au point de vue du personnel, les communications et les échanges d'informations sont faciles et directs. Les duplications de travail sont évitées du fait que chacun connaît bien le rôle des autres à l'intérieur de l'unité. De ce fait la motivation du groupe est améliorée puisque chacun a une conception claire et précise de son propre rôle à l'intérieur d'un groupe dont les objectifs sont très bien définis. D'un point de vue outils de travail, ici encore les bénéfices sont évidents. Ces outils se composant essentiellement d'outils informatisés, peuvent être plus facilement standardisés rendant ainsi l'échange de données informatiques à l'intérieur du groupe, très facile et efficace, tout en évitant, duplications, réencombrement manuel, etc., souvent des sources d'erreurs et de temps non-productif.

Le but de ce papier est donc de vous présenter le modèle intégré utilisé dans gestion du combustible à Gentilly 2, ainsi que les directions futures que nous entendons prendre sur la voie de l'intégration qui nous mènera vers une efficacité encore accrue.
Si on examine les grandes lignes des tâches de la Section Gestion du combustible on s'aperçoit que le premier objectif de notre intégration est assez bien atteint. En effet, les tâches décrites s'étalent sur toute notre définition large du "cycle complet du combustible". De son tout début avec l'approvisionnement de concentré d'uranium, en passant par la gestion en et hors cœur, jusqu'à sa presque toute fin, soit l'éventuel stockage à sec sur le site.

Dans le deuxième volet de la présentation (organisation des outils de travail) nous verrons plus en détails les interactions entre les différents préposés à ces tâches.

Afin de réaliser l'intégration de ces groupes, Hydro-Québec, lors de différentes réorganisations passées, a dû rapatrier à la Section Gestion du combustible certaines responsabilités qui relevaient de sections ou de divisions parfois organisationally très proches, parfois très éloignées, comme dans le cas de l'approvisionnement qui était un service au niveau de l'entreprise. L'effort fut justifié, puisqu'aujourd'hui, la justesse des résultats ainsi que l'efficacité avec laquelle l'unité les atteint ne sont plus à démontrer.
S'il est vrai que nous pouvons définir l'artisan à l'examen de son coffre d'outils, examinons ensemble les outils de la section Gestion du combustible afin de mieux comprendre son rôle.

En se référant au tableau 2, commençons au réacteur G-2. Les données d'exploitation sont recueillies, via une multitude d'instruments greffés au réacteur, par les ordinateurs de contrôle. Ici nous rencontrons un premier groupe hors de la section avec lequel il y a des communications fréquentes. Ce groupe est dans le même service que la section.

Le prochain module est TRADDEX (TRAduction d'Exploitation). Ce dernier, toujours dans la foulée de l'intégration, fut récemment transféré des ordinateurs de CRNL au système IBM Hydro-Québec situé à Montréal au siège social. Exploité par la Section Gestion du combustible, essentiellement, ce module recueille les données d'exploitation, les trie et les met en fichiers. Résultats, notre premier produit soit les fichiers maître de TRADDEX, qui correspondent au premier mandat du deuxième "groupe" de la section (voir tableau 1), soit la collecte des données d'opération du réacteur.

À l'aide d'un deuxième module, soit SIMEX (SIMulation d'Exploitation) également récemment déménagé de CRNL à IBM HQ, les données pertinentes des F/M TRADDEX sont traitées, et les résultats des calculs, diffusion, cartographie de flux, etc sont entre autres:

- les données nécessaires pour rapporter des performances du cœur (distribution de puissance de canal et grappes, etc.)
- une série de fichiers maître SIMEX
- les taux de combustion massique des grappes de combustible

Ces trois items correspondent également aux trois tâches majeures de notre deuxième groupe (voir tableau 1).

Avec un troisième module, SELECT, installé sur PC à la section, on accède au F/M SIMEX pour produire la liste hebdomadaire des canaux à être rechargés. Cette liste est transmise à la salle de commande et un deuxième groupe d'intervenants hors section, mais de la même division, soit le groupe manutention du combustible entre en action, et exécute les rechargements. On ferme ainsi la première boucle de notre cycle.

Le groupe manutention du combustible, initie alors une deuxième boucle. En effet, depuis quelques temps, le fabricant des grappes de combustible identifie ces dernières à l'aide d'un BARCOU. Ces BARCOUS, seront sous peu lus à l'aide d'un petit ordinateur portatif. Les données de rechargement passent de ces PC portatifs via un PC/TÉRMINAL au module INPHYCOM (INventaire PHYsique du COMbustible) installé sur IBM HQ. Ce module comme son acronyme l'indique sert à faire la gestion des stocks de combustible sur tout le site de G-2. En plus de recueillir, tester et traiter les informations sur les rechargements du cœur, ce dernier traite également de la même façon toutes les autres données pertinentes à ses fonctions ci-haut décrites. Ces informations proviennent de sources différentes:

- Toutes les données de base pertinentes aux grappes de combustible (nom, poids, déviations des spécifications techniques, programme chargement prioritaire, etc) viennent directement sous forme de disquettes du fabricant de combustible. Après traitement et authentification par le RAQ (Représentant pour l'assurance-qualité) proposé au contrat d'approvisionnement, ces données sont transmises pour traitement à INPHYCOM via un PC/TÉRMINAL lors de la réception du combustible en centrale.

- Tous les emplacements et déplacements de combustible neuf ou usé en centrale sont transmis soit à partir des ordinateurs portatifs lecteurs de BARCODE, soit à partir d'input direct au PC/TÉRMINAL.

- Les taux de combustion massique des grappes usées calculés à partir des FM/SIMEX lui sont également transmis via le PC/TÉRMINAL.

Toute cette information sera traitée, authentiﬁée, testée et traitée par le module INPHYCOM (Main Routine). D'autres modules dans INPHYCOM peuvent en retirer les données sous une multitude de formes. Ces informations serviront surtout à l'accomplissement des tâches de notre premier groupe (voir tableau 1).

On y retrouvera par exemple:

- Les données nécessaires aux responsables de l'approvisionnement du combustible pour la centrale:
  1) Taux de consommation
  2) Projections des besoins
  3) Approvisionnements passés
  4) Etc

- Les données nécessaires à l'analyse des performances du combustible:
  1) TCM moyens
  2) Historiques des grappes
  3) Programme "chargement prioritaire"
  4) Taux de défaillance
  5) Performances des grappes avec concessions techniques
  6) Etc

- Les données nécessaires à la gestion des stocks en centrale:
  1) Stock neuf & usé
  2) Taux d'occupation des piscines
  3) Listes détaillées des entrepôts de combustible neuf & usé
  4) Déplacements du combustible en centrale
Les données nécessaires pour rencontrer les exigences de la CCEA et de l'AIEA en matière de garanties de non-prolifération:

1) Rapports des inventaires mensuels à la CCEA/AIEA
2) Résultats des calculs de la composition isotopique des grappes
3) Liste du combustible sous surveillance et sous confinement par l'AIEA
4) Etc

Également, avec INPHYCOM, on produira un fichier qui identifie par No. de série des grappes dans le cœur et leurs positions respectives. Ce dernier sera transmis aux F/M SIMEX afin que ce module puisse attacher au combustible sous irradiation une identité propre. Cette opération ferme une deuxième boucle.

Finalement, une autre branche vient s'ajouter au cycle i.e. celle du calcul de la production de Cobalt-60 en réacteur. Depuis un certain temps, Hydro-Québec à Gentilly 2 met sous irradiation à chaque année, dans les barres de compensation, une certaine quantité de Cobalt-59. Ces B/C sont irradiées pendant un cycle d'à peu près 1 an, puis retirées du cœur et vendues à la société Nordion International Inc.

Afin d'en calculer et de facturer la quantité de Curies produits durant un cycle, un module appelé INCOGEN (INventaire du COBalt à GENtilly) puise dans les F/M SIMEX les données de flux moyen dans le cœur du réacteur. Les résultats de ces calculs serviront à notre groupe pour faire la gestion de ce Cobalt-60:

- Suivi de la production de Co-60
- Vérification du respect des normes de la CCEA
- Traitement du Co en piscine, expédition et facturation

En général, toutes les données produites par ce réseau informatisé sont recueillies par les gens de la section sur PC et, puis traitées à l'aide de logiciels tels que LOTUS 1-2-3, DBASE etc. Elles serviront à la production de rapports pertinents.

Nous voyons donc, que globalement, le deuxième objectif de notre effort d'intégration i.e. regroupé dans un seul réseau informatisé, tous les outils nécessaires..., est en bonne voie de réalisation. Depuis l'intégration des modules, au réseau IBM 4341, l'interconnexion informatisée des différents modules entre eux par PC, PC/TÉRMINAL ou par PC portatifs on peut en effet dire que les outils de travail sont intégrés et qu'ils ouvrent le cycle complet du combustible.

CONCLUSION

Nous avons démontré, à quel point le personnel préposé à la gestion du combustible était bien intégré à l'intérieur d'un groupe afin d'atteindre une efficacité optimale. Également, à quel point les outils de travail de ce groupe l'étaient aussi. L'union de ces deux facettes fait qu'aujourd'hui, la gestion du combustible à Gentilly 2 a vraiment pris son sens large et qu'elle s'effectue dans des conditions très efficaces. Les résultats obtenus jusqu'à date nous en assurent. Le temps consacré à ces tâches, dans certains cas, a été réduit par un facteur de 50 à 75%. Les données disponibles pour analyses des performances et pour la production de rapports ont augmenté non seulement en nombre mais aussi en qualité. Tous ces facteurs font que la décision d'intégrer et d'engagement de poursuivre cette voie, à mesure que de nouveaux moyens se présentent, ne fut jamais regrettée par Hydro-Québec.
Session 9:

Reactor Safety I

Chairman:

R.A. Brown, Ontario Hydro
A STUDY OF THE EFFECTIVENESS FOR RESPONSE TO LOSS OF CLASS IV POWER OF ONE STANDBY CLASS III GENERATOR OPERATING IN A CONTINUOUS OR TEST CYCLE MODE

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Abstract: This paper presents the results of a reliability study that models, evaluates and compares the effectiveness for response to loss of Class IV power (LOCLIV) of a single available Class III standby generator (SG) at an operating CANDU plant. The modelling is based upon operating and licensing considerations at Point Lepreau Generating Station (PLGS) and is applicable to cases where one of its two SG's is out of service for maintenance when a LOCLIV event occurs. The different licensing requirements for two modes of SG operation, either continuous or in a series of successive testing cycles, when the reactor is shut down or operating at power, have been modelled. Techniques have been developed to convolve the modelled probability of occurrence of a LOCLIV at a particular time instant with the probability of successful SG response and Class III system reconfiguration from the existing operational state to the required safety response state. The concepts, techniques and models developed to measure the response to a LOCLIV event by an SG in continuous or cyclic mode of operation are described in this paper.

1. INTRODUCTION

The PLGS station electrical services are classified in order of their levels of reliability requirement. Those power supplies are divided into four classes, namely I, II, III and IV that range from uninterruptible power to that which can be interrupted with limited but acceptable consequences. Very brief descriptions of the electrical power supply systems are presented in this paper. Detailed descriptions of these systems are given in the respective plant design manuals. Two 5 MWe standby diesel generator sets are provided to supply emergency power to the ODD and EVEN 4,16 KV Class III buses 5322-BUE and -BUF in the event that: (a) normal Class IV supplies to either or both buses fail, or (b) a loss of coolant accident (LOCA) occurs. Standby generators are started automatically on loss of normal Class IV power and are ready to accept load in less than one minute after receiving a start signal. Either generator can supply the required station shutdown loads. These loads are added to a demand-started SG by an automatic, selective sequencing arrangement.

The current operating policy for standby generators is to operate one SG continuously while the other SG is unavailable due to a forced or planned outage. This continuous operating procedure is compared with the alternate policy of cyclic operation of one SG when the second SG is unavailable. In this paper, the relative effectiveness of both modelled responses is compared via the response to a LOCLIV event in the case the reactor is at power.

Markov models are developed for the SG in continuous and cyclic operating modes. Procedures and methods have been proposed for deriving data, that may be input directly into the developed models, from station operating experience with electrical transients as well as from results of the standby generator routine testing programme. These Markov models have been solved using the Matrix Multiplication Technique. Mathematical methods are proposed for convolving the probabilities of the Class IV and Class III power availability for the case where the reactor is at power in order to assess the overall ability of Class III power to respond to a LOCLIV event. The case with reactor shut down is considered in a subsequent paper. The concepts, models and results of the sensitivity study comparing the projected SG responses to a LOCLIV event when in continuous and cyclic mode of operation are presented in this paper.

2. PLGS ELECTRIC POWER SYSTEM-STATION SERVICES

The different classes of power supplies are described in the following [11]. The overriding philosophy is that each class of power will be available from separate ODD and EVEN supplies and is capable of ensuring safe operation.

Class IV Power Supply: (AC) Power to auxiliaries and equipment that can tolerate long interruptions without endangering personnel or station equipment is obtained from the Class IV power supply. Complete loss of Class IV power initiates a reactor shutdown.

Class III Power Supply: Alternating current (AC) supplies to auxiliaries that are necessary for the safe shut down of the reactor and turbine when Class IV is lost are obtained from the Class III buses powered by two standby diesel generators. These auxiliaries can tolerate interruptions in power for about 3 minutes maximum duration.

Class II Power Supply: Uninterruptible alternating current (AC) supplies for essential auxiliaries are obtained from the Class II power supply. If a loss of Class III power occurs, the battery feeding an Inverter in the applicable circuit will ensure necessary power.
**Class I Power Supply:** Uninterruptible direct current (DC) supplies for essential auxiliaries are obtained from the Class I battery power supply.

3. **STANDBY GENERATOR OPERATING POLICY CONSIDERATIONS RELEVANT TO SG MODELLING**

Standby power for the Class III loads is supplied by two diesel generator sets. Each diesel generator can supply the total safe shutdown loads of the unit. The Class III shutdown loads are duplicated, one complete system being fed from each diesel generator. In the event of failure of Class IV power, the "poised" diesel generators are signalled to start automatically. The generators can be up to speed and be fully loaded in less than three minutes. When both SG's start, each generator automatically energizes half of the shutdown load through a load sequencing scheme. The normal status of the standby generators is both generators available, shut down and selected to start automatically in response to a LOCLIV or LOCA signal ready to supply power to the respective 4.16 KV Class III buses. Whenever one standby generator becomes unavailable, the other is started, synchronized and partly loaded. If both standby generators are unavailable, the reactor is required to be shut down within eight hours and the primary circuit cooled.

**Electrical system maintenance is confined to either the ODD or the EVEN sections of the system at any one time.**

State space models which represent operating constraints on one SG while the other SG is unavailable are described in the following section for the cases when the reactor is at power and shut down.

4. **MODELLING OPERATION OF A SINGLE STANDBY GENERATOR WITH THE OTHER UNAVAILABLE**

The sequence of operating states of the standby generator can be modelled as a stochastic process. In order to solve a stochastic process, it is desirable first to construct an appropriate state space diagram then to insert the relevant transition rates. Models which represent the continuous or cyclic mode of operation of one standby generator while the other SG is unavailable are presented in the following subsections.

4.1 **Reactor at Power**

4.1.1 **Modelling of the Continuous Operation Mode**

Figure 1 represents the continuous operating mode for one standby generator while the other SG is unavailable.

![Figure 1. Sketch of the Continuous Operating Mode for One Standby Generator](image)

In the continuous operating mode, one standby generator is run continuously for the entire time interval (0-T) that the other standby generator is unavailable. Figure 2 is a state space diagram for a standby generator in the continuous operation mode. State 1 and transitions 1-2 and 1-3 do not occur in the case of continuous SG operation; however, they are included in Figure 2 to indicate how this mode of operation would be treated in a manner similar to the cyclic mode if a demand start were required.

**Figure 2. Model for the Standby Generator in Continuous Operation Mode**

The basic assumption in the continuous operation model of Figure 2 is that at time \( t = 0 \), a successful start attempt is made and the SG enters State 2, i.e. the running state. The model of Figure 2 then reduces to a three state model consisting of States 2, 3 and 4. It is assumed that the SG exists in any one of the three states at any moment during the outage of the other SG. The operating SG may experience a severe extended outage which initially transfers it from State 2 to State 3 and, when repair time extends beyond eight hours, the SG transfers to State 4. The SG is transferred to State 2 from State 4 after successful repair and it finally returns to the poised state via a successful post-maintenance test in running State 2.
4.1.2 Modelling of the Cyclic Operation Mode

A sketch of the cyclic operating alternative for the SG is shown in Figure 3.

![Diagram of SG state transitions]

State 2: This is the running state. The residence time in State 2 has four constituent parts, e.g., the run time following all test start attempts, the "after-repair" post-maintenance test run time for all outages, the run time prior to each running failure and the run time in response to any LOCLIV demand. There are two possible transitions from this state, namely:

i) Transition from State 2 to State 1 after a successful test or LOCLIV response run and after a successful post-maintenance run following a planned or forced outage.

ii) Transition from State 2 to State 3 due to running failures.

State 3: This is the initial failure state. It is entered from the poised state via the dormant outage state as a result of a dormant failure or due to an annunciated failure or due to a failure while the SG is running. The total time spent in this state is the sum of all repair times. The length of any stay in this state has a maximum of 8h constrained by the Operating Policies and Principles requirement for a unit shutdown. Two transitions can be made from this state:

i) Transition from State 3 to State 2 after repair within 8h of failure occurrence.

ii) Transition from State 3 to State 4 when the required repair time exceeded 8h.

State 4: This is the extended failure state entered after an SG repair time exceeds 8h. One transfer out of this state can be made:

i) Transition from State 4 to State 2 after repair. The established plant policy is to prove SG availability via a run after a repair and then to put the SG in the poised state.

State 5: This is the dormant outage state. No repair is performed on the SG while in this state, because the need for repair has not yet been annunciated. The only transition from this state is:

i) Transition from State 5 to State 3 due to discovery of a demand failure. The SG is totally repaired in State 3 when the repair time is less than 8h or enters State 4 otherwise.

4.2 Reactor is Shut Down

Differences in licensing requirements between the reactor at power and reactor shutdown states require different state space models. A parameter known as the Recall Time plays a very important role in modelling SG operation with the reactor shutdown. Recall Time is the maximum tolerable interruption of power to critical loads. Thus a Recallable Outage

---

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State for the SG is defined as a state from which the SG can be returned to service within the Recall Time following an annunciated demand. For such successful action, no unavailability ensures.

Typical values of 0.5 h to 3 h can be used for Recall Times. There are two types of recallable outage states for the standby generators:

1. A failure state for which the repair time is less than the Recall Time.
2. An outage state for which there is a guarantee that the remaining time to complete the repair will be less than the Recall Time from any subsequent instant should a demand for the SG power arise.

The modified models for the SG in continuous and cyclic modes of operation given the reactor is shutdown are described in Subsections 4.2.1 and A.2.3.

4.2.1 Continuous Operation with Reactor Shutdown

The continuous operation model of the standby generator applicable to the reactor shutdown case is given in Figure 5.

Figure 5. Continuous Operation Model for one Standby Generator with the Reactor Shut Down

The states of Figure 5 are:

State 1: This the poised state. This state and transitions from and to it do not form part of the continuous operation model for cases where the other SG was made unavailable due to a planned or maintenance outage as the second SG would have been started first to cover off the known outage requirement. It is important to note that transitions 1-2 and 1-3 represent successful and unsuccessful start attempts that must be included in the continuous operation model for the SG when the other SG is unavailable due to a "forced" outage. In the reported studies, it is assumed that at t=0, a successful entry was made into State 2. Therefore State 1 is not part of this model.

State 2: This is the acceptable running state. The SG is running and capable of supplying the potential maximum load demand at that time. For conservatism, this is taken to be 4.5 MWe, the analyzed demand load for LOCA/LOCLIV response when the reactor has been operating at full power. The SG enters the running state after a successful start and two transitions from this state, 2-6 and 2-7 are considered.

State 3: This is the initial failure state. The repair duration must exceed the Recall Time. For example, if a failure requires six hours for repair, then the SG spends (6 - Recall Time) in State 3. If repair would require more than 12h, then the SG transfers into the extended outage state 12h after failure detection. The twelve hour value corresponds to the length of an outage shift following current plant practice. It is believed that, if repair cannot be affected within this time, further safety actions would likely be necessary.

State 4: This is the extended failure state. It is possible that the SG can experience failures which will require repair times exceeding 12h. Such failures were defined as extended failures in the reactor at power case. Here, as in Figures 2 and 4, State 4 is introduced to permit evaluation of the frequencies and probabilities that certain normally undesirable actions will have to be undertaken for cases of extended SG outages.

State 6: This is one of two Recallable Outage States. For this state, the SG can be repaired within the Recall Time following failure detection and then put back into the running state. The SG is considered to be capable of meeting heat sink and battery support requirements as the Recall Time defines the limit within which a load demand must be met. Thus no unavailability is associated with this state. However, while in State 6, the SG cannot provide power to the associated Class III bus.

State 7: This is the second Recallable Outage State. It corresponds to the final period of each outage spent in States 3 and 4. The duration of stay in State 7 is the Recall Time.

There are two kinds of unavailability associated with the SG when the reactor is shut down. One is unavailability to meet the load functional demand. It is the sum of probabilities for States 3 and 4. The other is unavailability to produce Class III power. That is the sum of probabilities of States 3, 4, 6 and 7. Only the unavailability to meet the load functional demand contributes to failure of standby Class III required response.

Recallable outage States 6 and 7 are assumed to be states from which the SG can be repaired successfully and have the Class III power supply restored with no failure to meet a load demand.

4.2.2 Cyclic Operation Model With the Reactor Shut Down

Figure 6 presents the cyclic operation model for the SG when the reactor is shut down.
Definitions of states provided in Subsection 4.2.1 apply also to Figure 6. State 5 is the Dormant Outage State as defined in Figure 4.

Transition 1-6 reflects annunciated outages whose repair time is less than the Recall Time. Transition 5-3 represents those demand start outages requiring repair time greater than the Recall Time; while transition 5-6 models demand start outages having repair time less than the Recall Time.

The "load functional demand unavailability" for the model of Figure 6 is the sum of State 3 and 4 probabilities. The "SG unavailability to produce power" in cyclic mode of operation with reactor shut down is the sum of probabilities for States 3, 4, 5, 6 and 7.

A.1 Estimation of Transition Rates

The transition rates of a state space model can be estimated from recorded data using the concept of the frequency of transfer from one state to another. For example, transitions from State i to State j can be described by [2]:

\[ f_{ij} = \lambda_{ij} P_j \]  

where:

- \( f_{ij} \) = frequency of transfer from State i to State j
- \( \lambda_{ij} \) = transition rate from State i to State j
- \( P_j \) = probability of State i

If, in a given recorded period, \( N_{ij} \) is the total number of transitions from State i to State j and \( T_j \) is the total time spent in State i, mean estimates for \( f_{ij} \) and \( P_j \) are:

\[ f_{ij} = \frac{N_{ij}}{T_j} \]  
\[ P_j = \frac{T_j}{T} \]

Substituting from Equations (2) and (3) into Equation (1) gives the estimate for transition rate:

\[ \lambda_{ij} = \frac{N_{ij}}{T_j} \]

Equation (4) was used in estimating the transition rates for all models presented in this paper.

Methods have been presented in References [3] and [4] to derive values of parameters which are compatible with the cyclic model from PLGS operating records.

4.4 Evaluation of State Probabilities

The Matrix Multiplication technique was used for determining the time-dependent state probabilities of both continuous and cyclic operation models. The Matrix Multiplication technique requires [2]:

\[ [P(t)] = [P(0)] [\lambda]^{n-1} \]  

where:

- \([P(t)]\) = Vector of system state probabilities at time \( t \)
- \([P(0)]\) = Vector of initial system state probabilities
- \([\lambda]\) = Stochastic transitional probability matrix
- \( n \) = Number of discrete, equal time steps used in the evaluation process.

In the Matrix Multiplication method, the stochastic transitional probability matrix is constructed for a small interval of time, \( \Delta t \). The actual value of \( \Delta t \) is chosen so that the probability of two or more inter-state transitions, involving either state, occurring in interval \( \Delta t \) time is negligible. After the stochastic transitional probability matrix for the time interval \( \Delta t \) has been established, the matrix is multiplied by itself repeatedly until the total time period of study has been modelled. Although the number of multiplications in practice may seem large, for a reasonable choice of \( \Delta t \), the method can give results with acceptable precision for all practical purposes. A value of 15 minutes was considered satisfactory for the SG system under consideration.

5. STATE SPACE MODEL SOLUTIONS

The state probabilities and the unavailabilities for the SG were computed for the case when the reactor is at power. PLGS site specific data for the observation period 82-07-01 to 87-12-31 were used in the computation process. When the information required for both the continuous operation and the cyclic operation models were not readily available in the PLGS data format, the model parameters were either assumed or obtained by using a data derivation approach. For example, no extended running failures are recorded in the PLGS site specific data. It was, therefore, necessary to assume some realistic frequency and duration of extended outages from the running state for the complete solution of the continuous operation model. The site specific database did not have derived residence times and transition rates for the cyclic model. Thus the data derivation approach was used to obtain compatible parameters. The data from the operating history of both PLGS standby generators were combined in the study.
The minimum and maximum repair times for running failures, recorded in the data base, are 0.2 and 6.2 hours, respectively. No outages from the running state exceeding 8h were recorded in the data base.

For the case when the reactor is at power and one SG is on planned outage, State 1 of Figure 2 does not exist. As no extended outage have been observed, the state space model of Figure 2 for continuous operation of one SG in response to a planned SG outage effectively consists of only States 2 and 3. Thus it was assumed in the study that State 1 was not part of the continuous operation model.

The index of interest, the time-dependent unavailability for the SG, can be estimated using the following equation [2]:

\[ U(t) = (\lambda_{23}/\lambda_{23}+\lambda_{32})=e^{-\lambda_{23}t+\lambda_{32}t} \]

(6)

The unavailable hours for the SG is the residence time in State 3, and is essentially the product of the probability of State 3 and the period of observation. Figure 7 shows the estimated time-dependent unavailability for the SG. Note that it reaches an equilibrium condition starting at the 10th week observation period.

To investigate the impact of repair time greater than 8h on the SG unavailability, State 4 of Figure 2 was added to the model (States 2 and 3) of the previous example. The following data was assumed to provide for a sensitivity analysis.

1. The transition rate from State 3 to State 4, \( \lambda_{34} \), was 10%, 1% and 0.1% of the transition rate from State 2 to State 3, \( \lambda_{23} \).

2. For each value of \( \lambda_{34} \), the average residence times in State 4 were 10, 100, 1000 and 2000 hours (these provided estimates for the transition rate from State 4 to State 2, \( \lambda_{42} \)).

This sensitivity analysis provides estimates of the impact of extended outages on performance measures for the system. The 3-state model for the continuous operation case which includes the extended outage state, was solved using the Matrix Multiplication approach described in Subsection 4.4. The initial probability vector used in the process was \([1 0 0]\) which signifies that at time \( t = 0 \), the SG was in the running state, i.e., in state 2 and the stochastic transitional probability matrix was:

\[
\begin{bmatrix}
1-\lambda_{23}\Delta t & \lambda_{23}\Delta t & 0 \\
\lambda_{23}\Delta t & 1-(\lambda_{23}+\lambda_{32})\Delta t & \lambda_{32}\Delta t \\
0 & \lambda_{32}\Delta t & 1-\lambda_{32}\Delta t
\end{bmatrix}
\]

The unavailability for the SG is the sum of State 3 and 4 probabilities. The unavailable hours for the SG is the product of the SG unavailability and the period of observation. The estimated SG unavailability for a one week period of observation is 2.4E-3. One important result was that for changes in \( \lambda_{34} \) and \( \lambda_{42} \) values, the variation in SG unavailability was not significant; it remains virtually unchanged for all cases investigated [3]. As stated earlier, values for the parameters of the cyclic model are not collected in the PLGS data base. Data reduction was therefore required to obtain a consistent set of model parameters. The basic assumptions made in the data reduction process were:

- cycle period = 3 days, i.e., 72 h = times between successive start attempts
- run duration, \( t_R = 3 \) h = running interval once started
- period of shutdown time, \( t_D = 69 \) h = portion of the cycle with SG poised to start auto
- Parameters required in the cyclic model are derived in References [3] and [4].

The time-dependent probability values were estimated using the Matrix Multiplication approach. The stochastic transitional probability matrix with time steps \( \Delta t \) of 15 minutes is:

\[
\begin{bmatrix}
1-\lambda_{23}\Delta t & \lambda_{23}\Delta t & 0 \\
\lambda_{23}\Delta t & 1-(\lambda_{23}+\lambda_{32})\Delta t & \lambda_{32}\Delta t \\
0 & \lambda_{32}\Delta t & 1-\lambda_{32}\Delta t
\end{bmatrix}
\]

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At time \( t = 0 \), as a result of a start attempt, the SG can transfer either to state 2 or state 3 depending on whether the start attempt is successful or not. Two vectors of initial probabilities were used in the computation process. The vector of initial probabilities based on observed data was:

\[
[P(0)] = [0.974, 0.026, 0, 0]
\]

This vector was derived from the database. There were 19 starting failures out of 733 total start attempts. The probability of starting failure is:

\[
P_{\text{SF}} = \frac{\text{Number of starting failures}}{\text{Total number of start attempts}} = \frac{19}{733} = 0.02592087 = 0.026.
\]

Therefore, the probability of a successful start is:

\[
P_{\text{SS}} = 1 - 0.026 = 0.974.
\]

The other vector of initial probability used in the calculation process reflects the design basis [5]. It was:

\[
[P(0)] = [0.95, 0.05, 0, 0]
\]

where a 5% demand start failure probability was assumed. State probabilities for an SG were computed for a one week period of observation. The sum of probabilities for states 3, 4 and 5 represents the unavailability for the SG. Sensitivity analyses were performed to investigate the impact of cycle period, test run duration and poised state duration on the unavailability of the SG. Reference [3] contains the details of the sensitivity analyses results. The results show that \( t_R \), \( t_p \) and cycle period impact on the unavailability of SG is virtually insignificant.

Table 1 compares unavailability values for the SG in continuous and cyclic operation for one week. These results apply to the case where transition rate \( \lambda_{34} \) was assumed in the continuous operation case to be 10% of the transition rate \( \lambda_{23} \) and transition rate \( \lambda_{2p} \) was assumed to be 0.1 occ/h. The length of the cycle period was 3 days with the test run duration, \( t_R \), 3 hours and the poised state duration, \( t_p \) was 69 hours.

Table 1. Unavailability Comparison for one SG in Continuous and Cyclic Operation Modes for a One Week Period.

<table>
<thead>
<tr>
<th></th>
<th>Continuous Operation</th>
<th>Cyclic Operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unavailability</td>
<td>0.00244337 (0.42 h)</td>
<td>0.01669008 (2.80 h)</td>
</tr>
</tbody>
</table>

The figures shown in brackets in Table 1 are the unavailable hours during that week. The results indicate that cyclic operation of the SG leads to higher unavailability. The periodic start up and shutdown subjects the SG in the cyclic operation mode to additional starting stresses compared with the SG in the continuous running mode. That risk is recognized by including the probability of demand start failures in the cyclic model. Another major contributor to the higher unavailability for the SG in the cyclic operation mode is dormant failure duration. For the continuous running case, only the running failures contribute to the unavailability of the SG.

The unavailability of power to the Class III loads during an SG outage is considered in the following section. There the unavailabilities of the Class IV and Class III supplies are combined to establish an overall measure of station power unavailability.

6. COMPARISON OF CLASS III POWER AVAILABILITY WITH DIFFERENT OPERATING MODES FOR THE SG

Figure 8 depicts the possible situations which can arise when the status of Class III and Class IV power supplies are combined in order to assess the reliability of electrical supply during the outage of one SG.

![Figure 8](image-url)

- **(a) \( P_{SO} \)**: Probability of successful operation of Class IV power throughout the SG outage
- **(b) \( P_{OSR} \)**: Probability of an overall successful response by Class III power to a loss of the Class IV power.
- **(c) \( P_{TL} \)**: Probability of a total loss of the Class IV and Class III power supplies.

Figure 8. Event Sequence for the Supply of Power to Required Class III Loads During Outage of a Single SG

For the \( P_{SO} \) case, no LOCLIV demand is placed on the Class III standby power to meeting plant requirements. Consequently, the probability of this case arising is independent of operating modes for the SGs.

Branch (b) of Figure 8 identifies the probability of successful response by standby Class III power to a loss of Class IV power. \( P_{OSR} \). The probability of a successful response by the standby generators is dependent upon the operating procedures for them. \( P_{SO} \) and \( P_{OSR} \) gives the probability that the Class III loads will be adequately supplied with power during the outage of the other SG. Branch (c) corresponds to a loss of Class IV power followed by a failure of the SGs to supply the Class III loads. The probability of the total loss of electrical power, \( P_{TL} \) is dependent upon the operating procedures for the SGs.

A model for a loss of Class IV power (LOCLIV) is given in Subsection 6.1. Mathematical models for the probability of a successful response by the operable Class III standby generator to a LOCLIV and also for the probability of LOCLIV and loss of Class III power are derived and detailed in Subsections 6.3 and 6.4. Class IV outage durations considered in the sensitivity studies were 1, 4, 24, 48 and 168.
hours. A summary of the values computed for P_{SO}, P_{SSR} and P_{TL} is included in Subsections 6.3 and 6.4.

5.1 Modelling of the Loss of Class IV Power

A LOCLIV power can be considered as a Poisson process. The complete expression for the Poisson distribution is given as [2]:

\[ P_N(t) = (\lambda t)^N e^{-\lambda t} / N! \]

Equation (7) recognizes and counts multiple failures but does not include the time taken to repair or replace a component when it does fail. Consequently, terms in the expression for the probabilities of two or more failures assume instant replacement or, at least, repair on a time scale short in comparison with the mean time to failure of that component. Table 2 gives the probabilities of occurrences of 0 to 3 Class IV power failures during a one week period when the reactor is at power using a failure rate for Class IV power 37.9E-06 occ/h. The latter value was derived in official station documentation [6].

Table 2. Probability of Class IV Power Failure in One Week with Reactor at Power (Class IV Failure Rate is 37.9 E-6)

<table>
<thead>
<tr>
<th># of Failures</th>
<th>Probability of Loss of Class IV Power Failure Within a 168 h period</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.99365303</td>
</tr>
<tr>
<td>1</td>
<td>0.00632679</td>
</tr>
<tr>
<td>2</td>
<td>0.00002014</td>
</tr>
<tr>
<td>3</td>
<td>0.00000004</td>
</tr>
</tbody>
</table>

Table 2 shows that the probability of having more than one failure during an SG outage time of one week is negligible. Therefore only one failure of Class IV power during the SG outage interval is considered throughout this study. The mathematical model for the first failure of Class IV power occurring in the time interval (0, t) is:

\[ P_{SF} = \lambda t e^{-\lambda t} \]

6.2 Reliability of Class IV Power, P_{SO}

The reliability or successful operation of Class IV power over the standby generator outage period can be derived from Equation (7) as:

\[ P_{SO} = e^{-\lambda t} \]

Using \( \lambda = 0.00003790 \) occ/h failure rate for Class IV power with the reactor at power, the estimated P_{SO} for a one week SG outage is then:

\[ P_{SO} = e^{-0.00003790 \times 168} = 0.99365 \]

Note that this value of P_{SO} is independent of whatever operating procedure is followed for the remaining SG to supply standby Class III power.

6.3 Probability of a Successful Response to a Loss of Class IV Power, P_{SSR}

Figure 9 depicts the occurrence of a LOCLIV while one standby generator is on outage.

![Figure 9](image-url)

For the time interval (0, T_R), i.e., before the occurrence of the loss of Class IV power, the SG is running synchronized with certain breaker and load configuration. For the duration of the LOCLIV, i.e., (T_R - T_Q), the SG must run with a different bus and load configuration. Therefore, before the loss of Class IV occurrence, the system operating status is X while following the LOCLIV in order to have an acceptable response, the configuration must be Y. The probability that the Class III system successfully survives reconfiguration (P_{SSR}) is the probability of an acceptable transition from X to Y. The point estimate for P_{SSR} was 0.855 determined in Reference [3]. To determine the sensitivity of electrical support to P_{SSR}, its value was varied between 80% to 100%.

6.3.1 A Model for the Probability of a Successful Response to a Loss of Class IV Power in Continuous Operation Case

Given the LOCLIV power occurs at time T_Q, the probability of the running SG being able to respond successfully, P_{SR} = (probability of the SG being in running state at T_Q) \times (probability of system successful reconfiguration) \times (probability of the SG surviving Class IV outage time). Mathematically it can be expressed as:

\[ P_{SR} = P_2(T_Q) \times P_{SSR} \times e^{-\lambda_SG(T_F-T_Q)} \]
Excluding the component $P_{\text{SSR}}$, Equation 8 can be derived from the reliability function $P = e^{-\lambda_S t}$ as follows:

$$P = e^{-\lambda_S t}$$

Hence, $P_2(T_0) = e^{-\lambda_S T_0}$ and $P_2(T_F) = e^{-\lambda_S T_F}$

or, $P_2(T_F)/P_2(T_0) = e^{-\lambda_S (T_F-T_0)}$

Therefore, $P_2(T_F) = P_2(T_0) e^{-\lambda_S (T_F-T_0)}$

Note that for constant $(T_F-T_0)$ intervals that start at different values of $T_0$, the exponential term is a constant, while $P_2(T_0)$ is time variant.

Including the effect of $P_{\text{SSR}}$, the quantity $P_{SR}$ reduces to:

$$P_{SR} = P_2(T_0) P_{\text{SSR}} e^{-\lambda_S (T_F-T_0)}$$

In Equation (8),

$$P_{SR} = \text{Probability of SG being able to respond successfully to the LOCLIV demand given that demand occurred at } T_0.$$  
$$P_2(T_0) = \text{Probability of the SG running at time } T_0.$$  
$$e^{-\lambda_S (T_F-T_0)} = \text{Probability that the SG runs successfully in the interval } T_0<T_F.$$  
$$P_{\text{SSR}} = \text{Probability that the Class III system successfully reconfigures to meet the demand of a LOCLIV at } T_0.$$  
$$\lambda_S = \text{Failure rate for the SG in Test Mode (a constant).}$$

$$= 0.00205456 \text{ h}^{-1}$$

The expression for the probability of a successful Class III response to a LOCLIV occurring at an arbitrary time instant $t$ during the SG outage time, $P_{\text{OSR}}$, is given by:

$$P_{\text{OSR}} = P_{\text{SSR}} \int_0^T P_{\text{LOCLIV}} (t) x$$

$$[P_2(t) x e^{-\lambda_S (T_F-T_0)}] \ldots (9)$$

where,

$$P_{\text{SR}} = \text{The probability of a successful response by Class III Standby Power to a LOCLIV that started at any instant in the interval } 0<t<T$$

$$P_{\text{LOCLIV}}(t) = \text{Probability that the first failure of Class IV occurs in time interval } dt \text{ about } t$$

$$\lambda_{IV} = \text{Failure rate for Class IV power.}$$

The integral range $(0, T)$ encompasses the SG outage duration. The expression for the probability of a successful response to a LOCLIV which could occur at any instant during the SG outage reduces to:

$$P_{\text{OSR}} = P_{\text{SSR}} e^{-\lambda_S (T_F-T_0)} \int_0^T \left( \lambda_{IV} e^{-\lambda_{IV} t} \right) P_2(t) dt \ldots (10)$$

The basic assumption made in deriving Equation (10) is that the LOCLIV duration, i.e., $(T_F-T_0)$, can be treated as a constant which may be given different values for performing sensitivity analysis. In Equation (10) $P_{\text{SSR}}$ and $e^{-\lambda_S (T_F-T_0)}$ are constant values. The integral

$$\int_0^T \lambda_{IV} e^{-\lambda_{IV} t} P_2(t) dt$$

can be evaluated utilizing the Trapezoidal Rule. Denoting $Y(t) = \lambda_{IV} e^{-\lambda_{IV} t} P_2(t)$ and subdividing the integral range $(0, T)$ into $n$ equal parts by the points $0, T_0, T_1, T_2, T_3, \ldots, T_{n-1}, T_n = T$ and letting $Y_0 = Y(T_0), Y_1 = Y(T_1), Y_2 = Y(T_2), \ldots, Y_n = Y(T_n)$, with $h = (T-0)/n$, the integral reduces to:

$$\int_0^T \lambda_{IV} e^{-\lambda_{IV} t} P_2(t) dt = h/2 \left( Y_0 + 2Y_1 + 2Y_2 + 2Y_3 + \ldots + 2Y_{n-1} + Y_n \right)$$

Reference [3] provides the computed probability values for successful operation of the standby generator with different fixed Class IV outage durations for the continuous operation model based upon an estimated failure rate for the SG operated in test mode of $\lambda = 0.00205456$ occ/h. Outage time for the other SG was 168 hours and the value of $h$ in the trapezoidal formula was taken to be 0.25 hour.

6.3.2 A Model for the Probability of a Successful Response to a Loss of Class IV Power in the Cyclic Operation Case

The mathematical model for the probability of Class III power being able to respond to a Class IV power loss in the cyclic operation mode is given by:

$$P_{\text{OSR}} = \int_0^T \lambda_{IV} e^{-\lambda_{IV} t} \cdot P_1(t) \cdot P_{SS} \cdot P_{SEQ} \cdot e^{-\lambda_S (T_F-T_0)} dt$$

$$+ \int_0^T \lambda_{IV} e^{-\lambda_{IV} t} \cdot P_2(t) \cdot P_{SSR} \cdot e^{-\lambda_S (T_F-T_0)} dt$$

$$= P_{SS} \cdot P_{SEQ} \cdot e^{-\lambda_S (T_F-T_0)} \int_0^T \lambda_{IV} e^{-\lambda_{IV} t} P_1(t) dt$$

$$+ P_{SSR} \cdot e^{-\lambda_S (T_F-T_0)} \int_0^T \lambda_{IV} e^{-\lambda_{IV} t} P_2(t) dt \ldots (11)$$
Where,

\[ P_{SS} = \text{Probability of a successful SG start from the poised state} \]

\[ P_{SEQ} = \text{Probability of successful sequencer operation. Given that an SG is poised auto, when LOCLIV occurs, the sequencer provides all required permissive signals within the design range of time intervals for the sequenced loads to operate if a process requirement exists.} \]

\[ e^{-\lambda_{SG}(T_f-T_0)} = \text{The reliability of SG operation for time interval } T_0 \text{ to } T_f. \lambda_{SG} \text{ is the auto logic SG failure rate when poised auto and } \lambda_{SG} \text{ is the test logic SG failure rate when running in a test mode.} \]

\[ \lambda_{IV} e^{-\lambda_{IV}t} dt = \text{The probability of Class IV first failure in the time interval } dt. \]

\[ P_1(t) = \text{Probability that SG is in the poised state at } t. \]

\[ P_2(t) = \text{Probability that SG is in the running state at } t. \text{ Note, this } P_2(t) \text{ differs from the Section 6.3.1 value.} \]

\[ P_{SSR} = \text{Probability of successful system reconfiguration following a loss of Class IV power and correct sequencer response.} \]

Point estimate values of \( P_{SSR} \) and \( P_{SEQ} \) which are 0.855 and 0.795, respectively, were derived from generic data \([5]\). In the present studies, however, \( P_{SSR} \) and \( P_{SEQ} \) were varied from 80% to 100% for sensitivity analyses. \( P_{SS} \) values used in the computation were 0.974 and 0.95 as employed earlier in this paper. Reference \([3]\) presents the computed probabilities of standby generator successful response to a loss of Class IV power in the cyclic operation case.

6.4 Probability of the Total Loss of Class III and Class IV Power, \( P_{TL} \)

6.4.1 Continuous Operation Case

The mathematical model for the probability of the total loss of Class III and Class IV power where the SG has been operated in continuous mode is given by the following equation:

\[ P_{TL} = \int_0^T \lambda_{IV} e^{-\lambda_{IV}t} dt \]

\[ -P_{SSR} \cdot e^{-\lambda_{SG}(T_f-T_0)} \int_0^T \lambda_{IV} e^{-\lambda_{IV}t} P_2(t) dt \] ....(12)

\[ P_{SEQ} \cdot e^{-\lambda_{SG}(T_f-T_0)} \int_0^T \lambda_{IV} e^{-\lambda_{IV}t} P_1(t) dt \]

6.4.2 Cyclic Operation Case

The model for the probability of the total loss of Class III and Class IV power in the case of cyclic operation of the SG is:

\[ P_{TL} = \int_0^T \lambda_{IV} e^{-\lambda_{IV}t} dt \]

\[ -P_{SSR} \cdot P_{SEQ} \cdot e^{-\lambda_{SG}(T_f-T_0)} \int_0^T \lambda_{IV} e^{-\lambda_{IV}t} P_1(t) dt \]

\[ -P_{SSR} \cdot e^{-\lambda_{SG}(T_f-T_0)} \int_0^T \lambda_{IV} e^{-\lambda_{IV}t} P_2(t) dt \] ....(13)

Complete results computed using Equations (12) and (13) for different \( P_{SS} \), \( P_{SSR} \) and \( P_{SEQ} \) values are presented in Reference \([3]\). Terms outside the integral sign in Equations (9) to (11) are constant values. Table 3 presents a comparison of computed values for the probability of the total loss of Class III and Class IV power in both continuous and cyclic operating modes for the available SG using point-estimate values for the probability of successful system reconfiguration (\( P_{SSR} \)) and the probability of successful sequencer operation (\( P_{SEQ} \)).

Table 3: Comparison of Calculated Values for the Probability of the Total Loss of Class III and Class IV Power

<table>
<thead>
<tr>
<th>(( T_f - T_0 ))</th>
<th>Continuous Operation</th>
<th>Cyclic Operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Class IV available to Class III Buses</td>
<td>( P_{SO} = 0.99365303 )</td>
<td>( P_{SO} = 0.99365303 )</td>
</tr>
<tr>
<td>Class III not available</td>
<td>( P_{CB} = 0.0541020 )</td>
<td>( P_{CB} = 0.0541020 )</td>
</tr>
</tbody>
</table>

Figure 10 summarizes estimated values of \( P_{SO} \), \( P_{SSR} \) and \( P_{TL} \) values for both continuous and cyclic operations of the standby generator for the case when the Class IV outage time is considered to be 2 hours.

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Class IV available to Class III Buses

\[ P_{50} = 0.99365303 \]

Class III available

\[ P_{SSR} = 0.00485515 \]

Class III not available

\[ P_{II} = 0.00014918 \]

b: Cyclic Operation \((P_{SSR}=0.974)\)

Class IV available to Class III Buses

\[ P_{50} = 0.99365303 \]

Class III available

\[ P_{SSR} = 0.00473909 \]

Class III not available

\[ P_{II} = 0.00160788 \]

c: Cyclic Operation \((P_{50}=0.95)\)

Figure 10 Criteria for Continuous and Cyclic Operation Cases Using \(P_{SEQ} = 0.795, P_{SSR} = 0.555\) and Class IV Outage Duration of 2 h.

The results shown in Table 3 and Figure 10 indicate that the continuous and the cyclic modes of operation are comparable in terms of probability of the total loss of Class IV and III power and also in terms of the probability of the successful response by the standby Class III power system to a loss of Class IV power given the point-estimate values for \(P_{SSR}\) and \(P_{SEQ}\) are used. However, if the values for \(P_{SSR}\) and \(P_{SEQ}\) approach 100%, then the continuous operation case yields significantly higher availability than the cyclic mode [3].

7. CONCLUSIONS

This paper has presented the modelling and some numeric results of a reliability study that compares continuous and cyclic operation modes of one standby generator when the other SG is unavailable. State space models were developed for cases of the reactor shut down and at power with the SG in both operating modes. For the case presented here of a one week SG outage, the cyclic mode operation of the other SG led to a 7 times higher unavailability prediction than did continuous operation. The two main reasons for this effect are the need for inclusion of a demand start failure probability and the potential for unannounced faults while the SG is in the "poised" state.

In relative terms, for the cyclic and the continuous operation cases noted here, predictions for probability of the simultaneous loss of the Class IV and Class III power and the probability of successful Class III response to a LOCLII are within a factor of two when the point-estimate values for the \(P_{SSR}\) and \(P_{SEQ}\) are used.

Further work is being done to determine realistic values for the probabilities quoted when \(P_{SSR}\) and \(P_{SEQ}\) take values representative of plant operating experience. As available site data is limited almost entirely to responses to actual Class IV upsets, it is difficult to quantify these variables. Additional efforts are also being made in the solution of the shut down reactor case. Here, the presence of a Recall Time relaxes the constraints of instantaneous response to a LOCLIV, thus providing an interesting situation where it is likely that continuous operation mode would be less "reliable".

8. REFERENCES

EMITTANCE OF ZIRCALOY-4 SHEATH AT HIGH TEMPERATURES IN ARGON AND STEAM ATMOSPHERES

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ABSTRACT

The spectral emittance of Zircaloy-4 sheath at wavelengths 1.0 and 2.3 μm was measured between 1100 and 1650°C in an argon and steam atmosphere. The results showed that at a given temperature the emittance increases rapidly from an initial value of about 0.3 in argon to a value of about 0.8-0.9 during steam oxidation. This change in emittance is associated with the color change of the fuel sheath, from bright metal in argon to black oxide in steam. An increase in the surface temperature of the Zircaloy-4 sheath led to a small decrease in emittance. A linear correlation between emittance and sheath temperature was obtained.

No significant change in emittance was observed at the tetragonal to cubic phase transition temperature of 1580°C.

INTRODUCTION

During a postulated loss-of-coolant loss-of-emergency-coolant injection (LOCA/LOECI) accident, the main mode of heat transfer between the fuel bundle and the pressure tube is radiant energy exchange. This energy exchange is controlled by the emittance of the surfaces involved. Under accident conditions, the Zircaloy-4 (Zr-4) fuel sheath is exposed to steam and hence would be rapidly oxidized at temperatures above 1000°C. This oxidation changes the emittance of the sheath surface and a knowledge of the variation of sheath emittance during oxidation is essential to properly estimate the maximum temperatures of sheath, fuel and pressure tube during these accidents. To estimate the maximum temperatures by a thermal modelling, the total hemispheric emittance is required. This study, however, focuses on the determination of the normal spectral emittance of the fuel sheath at two wavelengths as the first step in estimating the total emittance.

LITERATURE REVIEW

The emittance change due to oxidation has been the subject of many studies (1-3). Mellinger and Bates (1) found the spectral emittance of Zr-4 in an unlimited steam supply to rapidly approach a value of approximately 0.9 and decrease slowly if the specimen is held longer at temperatures above 1200°C. A study carried out by White (2) shows the spectral emittance at 0.65 μm wavelength increasing after oxidation from 0.45-0.5 to 0.7-0.78. A similar trend is shown in the Thermal Radiative Properties (TRP) handbook (3). A summary of spectral emittance vs temperature for oxidized and unoxidized Zr and Zr-4 is presented in Figure 1. In all cases the spectral emittance of oxidized Zr and Zr-4 (closed symbols) is well above the emittance of the unoxidized specimen (open symbols).

The total emittance of Zr-4 was determined by White (2) to be 0.24 for unoxidized and 0.6-0.78 for oxidized specimens (Figure 2). The total emittance of unoxidized Zr was measured to be about 0.2-0.3 (1); and is also included in Figure 2. These values are in the same range as the spectral emittances shown in Figure 1, which do not show a strong dependency on wavelength, and suggest that both oxidized and unoxidized Zr-4 behave like a grey body.

As surface oxidation proceeds, the oxide approaches the stoichiometric composition of ZrO₂. The total and the spectral emittance of commercially available stabilized ZrO₂ are reported in the literature (4-5) and are summarized in Figure 3.

FIGURE 1: NORMAL SPECTRAL EMITTANCE OF Zr AND Zr-4;

FIGURE 2: TOTAL SPECTRAL EMITTANCE OF Zr AND Zr-4;

FIGURE 3: TOTAL SPECTRAL EMITTANCE OF STABILIZED ZrO₂.
The relative amount of reflectance and absorbance is determined by surface conditions such as roughness, composition, and temperature. It may also depend on the wavelength and direction of the incident thermal energy.

It can be concluded from Kirchoff's Law for a grey surface that the spectral emittance, the total emittance and the absorptance are all equal. If the surface is diffusely emitting, this will also be true in all directions, providing the incoming radiation is also diffuse.

Thermal Emittance

The intensity of radiation emitted by a surface depends on its temperature and its emittance. The total emittance is defined as

$$\varepsilon = \frac{I_T}{I_{T(b)}}$$

where $I_T$ is the total intensity of radiation emitted by a real surface and $I_{T(b)}$ is the total intensity of radiation emitted by a blackbody at the same temperature. Similarly, spectral emittance is defined as

$$\varepsilon_{\lambda} = \frac{I_\lambda}{I_{\lambda(b)}}$$

(1)

where $I_\lambda$ and $I_{\lambda(b)}$ are spectral-band radiation intensities. The spectral emittance can be found for a surface if its true temperature, $T_s$, and apparent temperature, $T_a$, are known. The apparent temperature, which varies with wavelength, is defined as the temperature of a blackbody emitting the spectral radiation of the same intensity as the surface under consideration.

The spectral intensity of thermal radiation of a blackbody is given by Planck's equation:

$$I_{\lambda(b)} = \frac{C_1}{\lambda^5 \left(e^{C_2/\lambda T} - 1\right)}$$

(2)

From Equations (1) and (2) we obtain

$$\varepsilon_{\lambda} = \frac{I_\lambda}{I_{\lambda(b)}} = \frac{e^{C_2/\lambda T_s} - 1}{e^{C_2/\lambda T_a} - 1}$$

(3)

At temperatures between 1100°C and 1700°C and wavelengths less than 3 μm,

$$e^{C_2/\lambda T} \gg 1$$

and Equation (3) becomes

$$\varepsilon_{\lambda} = \exp \left(\frac{C_2}{\lambda} \left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right)$$

(4)

where $C_2 = hC/k = 14 380 \text{ μm} \cdot \text{K}$.

In this study, the true temperature and apparent temperatures at two wavelengths were measured and Equation (4) was used to calculate the corresponding spectral emittances of the fuel sheath.

Assuming zirconium oxide behaves like a grey body, a measurement of spectral emittance near the peak of the blackbody radiation intensity distribution will
The rate constant is a function of the temperature and time. Of those investigations, the work done by Sawatzky and Ledoux (7) most closely resembles our study, we discuss in this section the oxidation kinetics of this material.

Above the α/β transformation temperature of Zr-4 (≈ 862°C), steam reacts with β-Zr to form a surface layer of ZrO₂ and an underlying layer of oxygen enriched α-Zr. Several investigators have determined the growth of these layers as functions of temperature and time. Of those investigations, the work done by Sawatzky and Ledoux (7) most closely resembles our test geometry.

According to Sawatzky and Ledoux, the growth of both ZrO₂ and α layer thicknesses are governed by a parabolic rate law:

\[ x = \delta t^{1/2} \]  

(5)

where \( x \) is the thickness of the layer in millimetres, \( t \) is the exposure time in seconds and \( \delta \) is the parabolic rate constant in millimetres per (second)^{1/2}. The rate constant is a function of the temperature and the type of layer formed. The rate constant for ZrO₂ layer growth shows a discontinuity at 1580°C. This discontinuity is due to the tetragonal to cubic phase change in ZrO₂ at this temperature.

In the temperature range from 1100 to 1580°C the rate constants are given by

\[ \delta_{ZrO_2} = 1.59 \exp\left(-\frac{9010}{T}\right) \]  

(6)

\[ \delta_\alpha = 5.03 \exp\left(-\frac{10480}{T}\right) \]  

(7)

and

\[ \delta_{(\alpha,ZrO_2)} = 6.18 \exp\left(-\frac{9880}{T}\right) \]  

(8)

The rate constants in the temperature range 1580-1700°C are

\[ \delta_{ZrO_2} = 4260 \exp\left(-\frac{22860}{T}\right) \]  

(9)

\[ \delta_\alpha = 5.03 \exp\left(-\frac{10480}{T}\right) \]  

(10)

and

\[ \delta_{(\alpha,ZrO_2)} = 7550 \exp\left(-\frac{22730}{T}\right) \]  

(11)

In Equations (6) to (11), the temperatures are expressed in K and \( \delta \) is in mm/s^{1/2}.

Using the ZrO₂ layer rate constant at 1100°C, the time to form a layer of 2.3-μm, equivalent to the longest wavelength used in this study was calculated to be 1.04 seconds. Since emittance is a surface phenomenon, it depends on conditions not more than a few wavelengths from the surface. This indicates that a change in emittance, due to changes in oxide layer thickness, should not be expected after the first few seconds of oxidation. Any other changes in emittance, after the short initial transient, will probably be due to changes in phase or stoichiometry of the oxide layer rather than its thickness.

**EXPERIMENTAL**

A two-color Ircon-Mirage pyrometer was used in this study to determine the true temperature of the fuel sheath. The apparent temperatures at wavelengths 1.0 ± 0.1 μm and at 2.3 ± 0.3 μm were measured using a Williamson pyrometer and a Barnes pyrometer, respectively. All pyrometers were calibrated with a blackbody prior to these measurements.

**Calibration of Pyrometers**

The blackbody used for calibration was a 72-mm-long, 25-mm-diameter graphite cylinder with a central cylindrical cavity, 35 mm long and 8-mm diameter. From the opposite side of the blackbody, a cylindrical hole, 35 mm long and 6-mm diameter, was drilled, into which a V5X26 thermocouple was introduced. Thus, the blackbody cavity was separated from the end of the thermocouple by 2 mm of graphite.

The blackbody was suspended inside an insulated and argon-purged quartz tube, which was canted on the front end with a quartz foam cap. A 10-mm circular hole in the centre allowed an unobstructed view of the blackbody cavity by the pyrometers. A 25-kW RF induction unit was used to inductively heat the blackbody to temperatures between 500 to 1800°C. During the calibration procedure, a 3-mm-thick sapphire window was placed in front of the pyrometers since it would also be used in the fuel sheath emittance tests.

For pyrometer calibration, the blackbody was held at a constant temperature as indicated by the thermocouples. The voltage output from the pyrometers corresponding to the temperature of the thermocouple was recorded. This procedure was repeated at several temperatures and calibration curve equations were generated to correlate the thermocouple temperature to the voltage output of the three pyrometers.

**Emittance Test Apparatus**

A schematic of the experimental assembly used to determine the spectral emittance of Zr-4 is shown in Figure 4. It consists of a test chamber, a steam generator, a gas superheater, power supplies, pyrometers and data acquisition system.

The steam generator produces steam at a flow rate of 5 g/min and is heated to ~300°C in a superheater. Valves allow steam or argon to enter the emittance test chamber from the bottom through an alumina ceramic diffuser. The diffuser was designed to produce even gas flow inside the test chamber. The gases escape through the top.

The test chamber consists of a quartz containment tube, 165 mm high and 45 mm in diameter. Details of the test chamber is shown in Figure 5. The containment tube has a pyrometer view port, 15 mm in diameter, at a height of about 60 mm from the base. A sapphire window, 3 mm thick, covers the view port so that tests can be conducted in a gas-tight atmosphere.
The test specimen, an 82 mm long, Pickering size fuel sheath, is positioned vertically inside the containment tube. It was heated internally by a 6.3 mm graphite rod, tapered to a smaller diameter in the centre, and connected to a 10 kV.A power supply via two water-cooled copper electrodes. The three pyrometers, namely Ircon, Williamson and Barnes, were placed normal to the fuel sheath surface to view the same area of the sheath through the sapphire window.

**Test Procedure**

During an emittance test, the Zr-4 fuel sheath, cleaned previously with acetone and isopropyl alcohol, was heated internally and the temperature held constant at a value between 1100 to 1650°C. An argon atmosphere was maintained within the test chamber during the heating period and at the test temperature. The true temperature of the fuel sheath and the apparent temperatures at the two wavelengths 1.0 μm and 2.3 μm were measured with the calibrated pyrometers. Superheated steam was then introduced to displace argon leading to oxidation of the Zr-4. The changes in true and apparent temperatures were measured continuously as the oxide layer formed. During the test, the power input to the graphite rod was controlled to maintain the true temperature at the initial value. After the test, the specimen was cooled in an argon atmosphere and the thickness of the ZrO2 and α-layers formed was measured using standard metallographic techniques.

In selected tests, two calibrated Pt/Pt-13% Rh thermocouples were spot-welded to the fuel sheath 10 mm apart from each other, near the pyrometer view area and the temperature was determined. The temperature gradient across the view area was less than 0.3°C/mm. The temperatures measured by the thermocouples were within ± 1°C of those measured by the two-color Ircon pyrometer.

During the test, the three pyrometer readings were recorded at intervals of one to five seconds by a data acquisition program using a Tektronics 4052 computer. From the apparent temperature measurements and the true temperature, the spectral emittance, $\varepsilon_\lambda$, of unoxidized Zircaloy in argon and of oxidized Zircaloy in steam at wavelengths 1.0 μm and 2.3 μm was calculated using Equation (4). The results from each test consisted of a list and a plot of the temperature versus time, as well as two plots of normal spectral emittance, at wavelengths 1.0 μm and 2.3 μm as a function of time.

**RESULTS AND DISCUSSION**

As an example, Figure 6 shows the true temperature, $T_s$, and the apparent temperature, $T_a$, at 1.0 and 2.3 μm for a 1500°C test. In this test, the specimen was heated up in argon and maintained at 1500°C for about 40 s before steam was introduced. On steam injection, the true temperature rises rapidly to about 1780°C since the Zircaloy oxidation reaction is exothermic. About 10 s after steam injection, the true temperature falls to about 1240°C because of steam cooling: the inlet steam is at a lower temperature (~ 300°C) than the specimen. At this point the power supply to the graphite heater rod is adjusted to maintain the nominal test temperature of 1500°C. After 182 s, steam is displaced by argon and the power shut off. The apparent temperatures, $T_a$, follow the general trend of the true temperature, as expected.
Figure 6 shows the change in calculated emittance at a wavelength of 2.3 μm as a function of time for the same test. Emittance is about 0.27 before steam injection. On steam injection, emittance increases rapidly to about 0.88. This change in emittance is due to the change in color of the fuel sheath, from bright metallic in argon to black in steam. The black color of the oxide layer was observed after the test.

Figures 8 and 9 show a summary of the measured steady-state emittances from all the tests for unoxidized and oxidized fuel sheath as a function of temperature at wavelengths 1.0 μm and 2.3 μm, respectively. The following best-fit equations were obtained to correlate the emittance data with the temperature in °C.

At 1.0 μm, for unoxidized Zr-4 from 1400 to 1650°C
\[ \varepsilon_\lambda = 1.97 - 1.07 \times 10^{-3} T \]  
and for oxidized Zr-4 from 1300 to 1650°C
\[ \varepsilon_\lambda = 1.24 - 2.58 \times 10^{-4} T \]  

At 2.3 μm, for unoxidized Zr-4 from 1100 to 1650°C
\[ \varepsilon_\lambda = 0.47 - 1.3 \times 10^{-4} T \]  
and for oxidized Zr-4 from 1100 to 1650°C
\[ \varepsilon_\lambda = 0.89 - 3.4 \times 10^{-5} T \]

Our results were compared with literature data for similar ranges of spectral emittance. The data of Mellinger and Bates (1) for wavelength 1.5-2.5 μm for oxidized Zr-4 and those listed in the TRP (3) at 2.3-μm wavelength for oxidized and unoxidized zirconium are in good agreement with our data below 1500°C. A decrease in emittance with time to a lower value corresponding to ZrO, as reported by Mellinger and Bates (1), was not observed in our tests, probably because our tests lasted only 300 s maximum.
The emittance of oxidized Zircaloy at both wavelengths (Figures 8 and 9) does not show a significant change around 1580°C at which ZrO₂ changes phase from tetragonal to cubic lattice. This indicates that the expected emittance changes at this transition temperature are small compared to the experimental error.

Tests conducted in steam for different exposure times, from 60 to 180 s, at isothermal conditions did not show significant changes in measured emittance, indicating that an increase in the oxide layer thickness after the initial transient period does not affect the emittance value.

The thicknesses of α and ZrO₂ layers formed in the various tests were measured after cross-sectioning the fuel sheath. As an example, for a test conducted at 1300°C, the α and ZrO₂ layers formed after 86 s in steam are shown in Figure 10. The total layer thicknesses for various tests are compared in Figure 11 with the thicknesses calculated using Equations (8) and (11). Good agreement between the calculated and measured values is seen below 1450°C. Above this temperature, the measured values were generally higher than calculated. This is probably due to the significant temperature increase of the fuel sheath due to the exothermic reaction between Zr-4 and steam above this temperature. An increase in the temperature of the fuel sheath would lead to a decrease in thickness since the rate constants for layer growth increase rapidly with increasing temperature, as discussed earlier.

**FIGURE 10:** CROSS-SECTION OF AN OXIDIZED Zr-4 FUEL SHEATH AT 1300°C FOR 86 s

**FIGURE 11:** COMPARISON BETWEEN MEASURED AND CALCULATED THICKNESS OF ZrO₂ AND α LAYERS FROM VARIOUS TESTS

**CONCLUSIONS**

The normal spectral emittance of a Zr-4 fuel sheath increases rapidly on steam oxidation from a value of about 0.3 to about 0.8-0.9 in the temperature range 1100 to 1650°C. This change in emittance is associated with the change in color of the fuel sheath, from bright metallic when unoxidized to black upon oxidation in steam.

At temperatures below 1500°C, the measured emittance in this study is in good agreement with other workers. The final emittance value did not depend on steam exposure time and thus appears to be independent of oxide layer thickness. An increase in the surface temperature of the fuel sheath led to a small decrease in emittance, probably due to a change in the nature of the oxide layer formed with temperature. No significant change in emittance was observed at 1580°C, the phase transition temperature of ZrO₂ from tetragonal to cubic.

**ACKNOWLEDGMENTS**

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**REFERENCES**

ABSTRACT

Five experiments have been conducted to study the effects of pressure, liquid level and heating power on the circumferential temperature distribution and deformation of CANDU pressure tubes under stratified coolant conditions. Stratified coolant conditions in these experiments were effected by injecting near-saturated water into the heated pressure tube to replenish the water boiled off during each experiment. In all five experiments, the pressure tube ballooned at the top, resulting in near direct contact with the calandria tube. The experimental results suggest that pressure-tube deformation had a significant cooling effect on the pressure tube and therefore affected the circumferential temperature distribution.

The experimental program was funded by the CANDU Owners Group (COG).

INTRODUCTION

In some postulated Loss-of-Coolant accidents (LOCAs) with coincident loss of emergency coolant, the coolant in a horizontal fuel channel of a CANDU reactor may boil off and become stratified. [1] When this occurs, the exposed part of the pressure tube will be heated up by radiant heat from the uncovered fuel pins and steam convection. The pressure tube will then become hotter at the top while remaining close to the saturation temperature at or below the liquid level. The result is a circumferential temperature gradient around the pressure tube. If the coolant voiding rate is high, the circumferential temperature gradient will be relatively small and the pressure tube will heat up uniformly. When the pressure tube deforms, it will balloon uniformly into contact with its calandria tube and will cool down by transferring its heat to the surrounding moderator. [2] If the voiding rate is low during flow stratification, a relatively steep circumferential temperature gradient may develop on the pressure-tube wall. The pressure tube may deform preferentially at the top and may rupture prior to contact with the surrounding moderator-cooled calandria tube. The integrity of the pressure tube in these situations depends on the pressure-tube circumferential temperature gradient influenced by coolant level, power in fuel, and the fuel channel internal pressure.

Computer codes, such as CATENA [3,4,8,9], SMARTT [5], and AMPTACT [6] are capable of predicting the thermal and mechanical responses of the pressure tube during stratified flow conditions. However, these codes need to be verified against experimental data before they can be depended upon to analyze postulated reactor accident scenarios.

Two series of experiments have been performed at the Whiteshell Nuclear Research Establishment to study the thermal-mechanical response of hot pressure tubes under different stratified flow conditions, as well as to provide verification data for various computer codes. The first series consisted of four tests in which water was boiled off from the channel without being replenished. Both the experimental and numerical simulation results of this Boil-Off Test Series have been reported earlier. [5,7-9] Without replenishing the water being boiled off in the channel, the water boil-off rates in all tests were substantially higher than those expected in postulated small-break LOCA scenarios. In light of this, in the second series of tests, saturated water was injected into the heated channel at a constant rate to slow down the boil-off of water (hence "Make-Up water" test series). This second series, consisting of five tests, has been completed. The results of the first three tests have been reported elsewhere. [10] In this paper, descriptions and results of these five tests, particularly the last two, are presented.

The objectives of the five experiments were to measure the circumferential temperature distribution and deformation of heated pressure tubes as a function of heater power, coolant level, and pressure-tube internal pressure. Data from these experiments will be used to verify the computer codes mentioned earlier.

APPARATUS

The apparatus for all five tests was essentially the same, with only minor modifications from one to the other. Figure 1 shows the horizontal test section which consisted of a 2.29-m segment of a CANDU-type channel. Make-up water entered the test section at the inlet end through a connection welded to the bottom of the pressure tube. The outlet end was connected to a vertical pipe (25.3-mm ID) for steam to exit. Pressurized water from a boiler of a given flow rate was passed through a network of piping and injected into the test section. The vertical pipe at the test-section outlet was connected to a condenser and a surge tank pressurized with helium. The test-
section pressure was maintained at a desired value during the experiment by controlling the helium back pressure in the surge tank.

![Diagram of test section of the pressure-tube circumference temperature distribution experiment](image)

**FIGURE 1: TEST SECTION OF THE PRESSURE-TUBE CIRCUMFERENTIAL TEMPERATURE DISTRIBUTION EXPERIMENT**

Inside the channel, 36 electric heaters (each 2.3 m long) were grouped into three separate heater-element rings. These heaters, together with a centre tube, had the same cross-sectional dimensions as a CANDU-type 37-element fuel bundle. Each heater was made of two concentric zirconium tubes. [7] The inner tube was the heater filament and the outer tube the heater sheath. The heater filament was electrically insulated from the sheath by small alumina rings spaced 100 mm apart. During the experiment, each heater was pressurized with helium to prevent the sheath from collapsing or ballooning.

K-type thermocouples were spot-welded to the outside surface of the pressure tube and to the heater sheaths. These thermocouples were grouped in three rings at axial locations labelled R1, R2 and R3 as shown in Figure 2. Three Linear Voltage Displacement Transformers (LVDT) were installed to measure movements of the calandria and pressure tubes. LVDT 1 measured the absolute movement of the top of the calandria tubes. The gap width of the annulus between the pressure and calandria tubes was measured at the top and bottom by LVDTs 2 and 3, respectively.

![Diagram of axial locations of thermocouple rings and linear voltage displacement transducers](image)

**FIGURE 2: AXIAL LOCATIONS OF THERMOCOUPLE RINGS AND LINEAR VOLTAGE DISPLACEMENT TRANSDUCERS (DISTANCE IN mm)**

Typical locations (labels) of various thermocouples in each thermocouple-ring are shown in Figure 3. Thermocouples to measure heater-sheath temperatures (T/C 12 to 17) were only available in thermocouple-rings R1 and R3. Moreover, the R-type thermocouples (T/C 18) used in Test 3 to 5 to measure the heater-filament temperatures was only available in thermocouple-ring R3.

![Diagram of thermocouple locations on the pressure tube and inside the heated channel](image)

**FIGURE 3: THERMOCOUPLE LOCATIONS ON THE PRESSURE TUBE AND INSIDE THE HEATED CHANNEL**

The test section was immersed in a pool of water kept at 75°C. The annulus between the calandria and the pressure tubes was purged with CO₂ before each test. A garter spring was positioned around the mid-plane of the pressure tube.

**TEST PROCEDURE**

At the beginning of each test, the pressure tube was filled with water at room temperature and pressurized (1.1 or 3.9 MPa). The water surrounding the calandria tube was heated to 75°C to simulate the moderator water. The pressurized water in the pressure tube was gradually raised to the saturation temperature using six bottom heaters in both the outer and middle heater-element rings. This procedure reduced the top-to-bottom temperature gradient across the pressure tube. After the water in the pressure tube had reached saturation temperature, valves at the inlet and outlet were opened, and slightly subcooled make-up water was injected into the pressure tube.

After the make-up water has been injected into the pressure tube and all temperatures had stabilized (1-2 hours), the power supply was shut off and electrical cables to the rest of the heaters were quickly connected. Once the heater power was reestablished, the test was initiated and the transient time was referenced from this moment.

**RESULTS**

Table 1 summarizes the test parameters. In any of the tests, the pressure-tube circumferential temperature as well as heater-sheath temperature distributions along the pressure tube were quite similar. Therefore, for each test, we shall describe temperature distributions measured at only one of the axial thermocouple-ring locations. After the description of individual tests, the results will be compared with one another to examine the effect of heater power, water level (make-up water flow rate)
and pressure-tube internal pressure on the pressure-tube circumferential temperature distribution and deformation.

### TABLE 1: TEST PARAMETERS

<table>
<thead>
<tr>
<th>Test Number</th>
<th>Pressure (MPa)</th>
<th>Power (kW)</th>
<th>Flow Rate (g/s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.1</td>
<td>39</td>
<td>8.2</td>
</tr>
<tr>
<td>2</td>
<td>1.1</td>
<td>84</td>
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</tr>
<tr>
<td>3</td>
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<td>3.9</td>
<td>80-100</td>
<td>26</td>
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</tbody>
</table>

**Tests 1 to 3**

As the results of Tests 1 to 3 have been reported elsewhere [10], only highlights of these three tests are given here. Tests 1 to 3 were conducted at a relatively low channel internal pressure of 1.1 MPa. Power and make-up water flow rates are shown in Table 1. In Tests 1 and 2, a constant heater power was applied. The quasi-steady-state water level established in these two tests were 1/4 and 1/2 measured from the bottom of the pressure tube. The resulting pressure-tube temperature transients measured in Tests 1 and 2 are shown in Figures 4 and 5, respectively. In Test 3, power was increased in steps (39, 46, and 60 kW), with the make-up water flow rate remaining constant at 12 g/s. The resulting pressure-tube temperature transient is shown in Figure 6. In general, a wide circumferential temperature distribution with a large top-to-bottom temperature difference was observed in each test. However, it is obvious that the measured pressure-tube circumferential temperature transients depended on the parameters of heater power and coolant level. The effects of these two parameters will be discussed later.

The pressure tube deformed (ballooned) at the top in all three tests. The ballooning of the pressure tube affected the pressure-tube circumferential temperature, which in turn affected the rate of pressure-tube deformation. To illustrate this effect, the pressure-tube deformation measured by the LVDTS in Test 3 is also shown in Figure 6. In Period 1, with heater power = 39 kW, the top of the pressure tube reached a quasi-steady-state temperature of 628°C. There was hardly any pressure-tube deformation, probably because the pressure tube was not hot enough.

In Period 2, when the heater power was increased to 46 kW, the temperature at the top increased initially and the pressure tube started to balloon at the top. The ballooning of the pressure tube created a larger flow area inside the top of the pressure tube, enhancing the steam cooling of both the heaters and pressure-tube wall near the top. Also, the pressure tube moving away from the heaters at the top reduced the radiative heat flux incident on the inner wall of the pressure tube near the top. The radiative and conductive heat transfer to the moderator, however, was enhanced due to a reduced CO2 gap. As a result, the top of the pressure tube was cooled and the temperature at the top declined after reaching a maximum. As the temperature at the top declined, the deformation of the pressure tube slowed down (Figure 6). Hence, the pressure-tube deformation and temperature at the top of the pressure tube are mutually dependent. This cooling effect due to pressure-tube ballooning at the top was also observed in Test 1 (Figure 4).
Thermocouple 14 became uncovered at 970 s, but the centre pin (T/C 11) remained just covered for the rest of this period. It is therefore inferred that the channel was about 1/2 full after 970 s and for the rest of Period 1. The temperatures of the uncovered heaters increased to a maximum shortly after 1500 s but subsequently dipped and levelled towards the end of this period. The dip in heater-sheath temperature was due to the reduction in heater power at 1500 s. When the power was increased in Period 2, temperature of uncovered heaters increased initially. Temperatures started to decrease slightly at about 3750 s and then levelled off towards the end of Period 2, implying that the heaters were cooled. The cooling of heaters was due to pressure-tube deformation, which will be discussed later. The centrepin (T/C 11) became uncovered at about 3500 s, indicating that the channel was slightly less than half full after this instant.

In Period 3, when the power was further increased, the temperature at locations away from the top (T/C 2 and 9) increased and became higher than that of the top. Hence, the top of the pressure tube was cooled more effectively than other sectors.

Test 4

The power history for Test 4 was divided into two periods: 0 to 3086 s and 3086 to 4122 s (Figure 7). Initially, the heater power was ramped quickly to about 40 kW at 100 s. Between 100 and 1500 s, the heater power gradually increased. At 1500 s, power was intentionally lowered to maintain the water level in the pressure tube at mid-height. At 3087 s, the power was raised to lower the water level to below mid-height. The nominal power maintained in Periods 1 and 2 were 40 and 50 kW, respectively.

Figure 9 shows the measured temperature transients and movement of the pressure tube. The pressure-tube temperature transients were divided into two periods, as was the power input. In Period 1, the upper half of the pressure tube was quickly exposed to dry steam. The temperatures increased quickly, but started to decrease at approximately 1500 s. This temperature decrease was due to heater power reduction at 1500 s. The pressure-tube temperatures tended to level off towards the end of this period. A top-to-bottom temperature difference (ΔT) of approximately 300°C was developed towards the end of this period. This top-to-bottom temperature difference was considerably smaller than those observed in Tests 1 to 3. In the same period, Figure 9 also shows that the pressure tube hardly ballooned prior to 1500 s, confirming that the dip in pressure-tube temperatures at this instant was not caused by pressure-tube deformation. After 1500 s, the top of the pressure tube started to balloon quickly. The ballooning, however, slowed down after 1750 s as the pressure-tube temperatures were levelling off.

In Period 2, when the heater power was increased, temperatures near the top (T/C 1 and 2) of the pressure tube increased initially, but started to dip at about 3600 s. This decrease in temperature was due to the same cooling effect by the preferential ballooning of pressure tube at the top, as observed.
in Tests 1 and 3. This cooling effect by pressure-tube ballooning at the top is not as strong near locations away from the top, where there is a larger CO$_2$ annulus gap. The temperature measured by T/C 8 became the highest because the top of the pressure tube was cooled more efficiently.

![Figure 9: Measured Pressure-Tube Circumferential Temperatures at Thermocouple Ring R3 and Movement of Pressure Tube in Test 4](image)

In this test, a substantial circumferential temperature gradient was developed, causing the pressure tube to preferentially balloon at the top. However, no contact with the calandria tube was observed, and the test was terminated after 4200 s.

**Test 5**

The power history for Test 5 is shown in Figure 10. It is divided into three periods: 0 to 378 s, 378 to 919 s, and 919 to 1127 s. Nominal heater power attained in these three periods were 83, 89, and 99 kW. At 1008 s (end of Period 3), some heaters near the top of the pressure tube started to fail, causing the power to dip. At 1127 s, the pressure tube ruptured at the top and the test was terminated. Heater-sheath temperatures measured at axial thermocouple-ring R3 are shown in Figure 11. During Period 1, heaters near the top (T/C 12 and 13) were quickly exposed to dry steam as water in the channel boiled off. Thermocouples 14 and 15 were not exposed until Period 3. It is thus inferred that the pressure tube was slightly more than half full in Periods 1 and 2, and slightly less than half full in Period 3.

Figure 12 shows the circumferential temperature transients measured at thermocouple-ring R3 and movements of the pressure tube at thermocouple-ring R1. Shortly before the end of Period 1, temperatures near the top of the pressure tube (T/C 1) reached a maximum of 704°C before declining. Temperatures at other locations in the uncovered upper half of the pressure tube also declined after reaching a maximum. This familiar behaviour was caused by the cooling effect due to pressure tube ballooning, as the pressure tube ballooned substantially towards the end of Period 1. The maximum $\Delta T$ developed in this period was 455°C. When the heater power was increased in Period 2, pressure-tube temperatures at the top increased initially, and the pressure tube continued to balloon. The pressure tube started to contact the calandria tube near thermocouple-ring R1 at approximately 475 s as the LVDT reading at the top of the pressure tube remained constant after this instant. The contact between the pressure and calandria tubes near thermocouple-ring R1 cooled the top of the pressure tube at this location. The top of the pressure tube at thermocouple-ring R3, probably not yet in direct contact with the calandria tube, was also cooled by axial heat conduction. The temperature at the top fluctuated at first, but subsequently remained at a lower temperature of approximately 500°C. Temperatures at locations away from the top (T/C 2, 8, 3, 7) increased initially, but also decreased to a constant value after the pressure tube had come into contact with the calandria tube. However, the temperature measured by T/C 8 became the highest, implying that locations away from the top were not cooled as efficiently. In Period 3, since the pressure tube has already come into contact with the calandria tube, temperatures at the top remained constant despite the power increase. Temperatures at other uncovered locations increased because they were not cooled as effectively as the top.

![Figure 10: Power History of Test 5 (F indicating heater failure and end of experiment)](image)

![Figure 11: Measured Heater-Sheath Temperatures at Thermocouple Ring R3 in Test 5 (F indicating heater failure and end of experiment)](image)
Pressure tube failure occurred after heater failure. It was probably caused by local heating due to arcing when the heater failed.

DISCUSSION

In Tests 1 to 3 (pressure = 1.1 MPa), a large top-to-bottom AT in excess of 400°C developed on the pressure tube prior to heater failure. Pressure-tube deformation occurred in most cases, but the pressure tube neither contacted the calandria tube nor ruptured. 10] Pressure-tube circumferential temperature distribution curves for the three tests (Period 1 in Test 3 is used in comparison) are given in Figure 13. The curves shown correspond to the circumferential temperature distributions at the instant when the temperature at the top of the pressure tube has reached a maximum. Tests 1 and 3 were conducted at the same power but different make-up water flow rates resulting in different water levels in the channel. The water level in Test 1 was about 1/4 vs. 1/2 in Test 3 (Period 1). A higher peak temperature (715°C) at the top of the pressure tube (a larger AT) was observed in Test 1, which was caused by heaters being more uncovered at the lower water level. However, the circumferential temperature gradients are comparable in both cases. This shows that a lower water level results in a larger AT but not necessarily a steeper temperature gradient. Tests 2 and 3 were conducted at different heater power but comparable final water levels (both about 1/2). A larger AT, with a steeper circumferential temperature gradient, was developed in Test 2 (higher power). In this case, although the amount of uncovered heater surfaces were comparable, the heat flux from the heaters was higher in Test 2 because of a higher heater power. A higher heater power resulted in both a larger AT and a steeper circumferential temperature gradient.

Tests 4 and 5 were conducted at a relatively high pressure, 3.9 MPa, and with a channel being approximately half full. In each of these two tests, a circumferential temperature distribution developed on the pressure tube, which consequently deformed at the top. The circumferential temperature distribution, however, was affected by the heater power and the internal pressure. To illustrate the effects of these two parameters, pressure-tube circumferential temperature distribution curves for Tests 3 to 5 (Period 1 in all three tests are used for comparison) are given in Figure 14. The curves for Tests 3 and 4 correspond to the circumferential temperature distribution developed towards the end of Period 1. The curve for Test 5 corresponds to that developed at the instant when the temperature at the top of the pressure tube reached a maximum. Test 3 and 4 were conducted at comparable power and water levels but at different pressure-tube internal pressure. It appears that a higher internal pressure results in a smaller top-to-bottom temperature difference (AT) and a smaller circumferential temperature gradient. The reason is the following: first, the saturation temperature is higher at 3.9 MPa, second, the density of steam at 3.9 MPa is approximately four times that at 1.1 MPa. The higher steam density results in a higher heat capacity, cooling both the heaters and the top of the pressure tube more effectively. Moreover, the pressure tube started to balloon at a lower temperature because of the higher pressure. The cooling effect of the ballooning pressure tube limited the temperature at the top of the pressure tube. As a result, the top of the pressure tube reached a lower peak temperature. A lower temperature at the top, together with a higher saturation temperature at the bottom, results in a smaller top-to-bottom temperature difference. Since the water level is comparable, a smaller AT results in a smaller temperature gradient. A comparison between Tests 4 and 5 (both at the same pressure) shows that, for comparable water levels, a higher power results in a higher temperature reached at the top and, therefore, a larger top-to-bottom difference. Since the water levels are comparable, a steeper circum-

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ferential temperature gradient also results. This finding is consistent with that obtained from the comparison between Tests 2 and 3 for the effect of heater power.

Pressure-tube deformation near the top affects the circumferential temperature distribution and thus its own subsequent history. In all five tests except Test 2 (heaters failed too early to reveal the same phenomenon), the temperature at the top of the pressure tube declined after reaching a maximum. This occurred after the pressure tube had started to balloon at the top. Pressure-tube ballooning created a larger flow area inside and near the top of the pressure tube, enhancing the steam cooling of heaters and the pressure tube. Moreover, ballooning reduces the radiative incident heat flux on the inner wall of the pressure tube but enhances the radiative and conductive heat transfer to the moderator-cooled calandria tube. This cooling effect by pressure-tube ballooning is significant and should be taken into account in the calculation of pressure-tube circumferential temperature distribution during LOCA situations.

The pressure tube deformed in all five tests, but contacted the calandria tube only in Test 5, where both the heater power and pressure were high (power = 80 to 100 kW, pressure = 3.9 MPa). Apparently, the pressure tube was heated to a higher temperature with a high heater power in spite of the more effective cooling of denser steam at higher pressure. This, compounded by a higher internal channel pressure, resulted in the pressure tube ballooning faster and coming into contact with the calandria tube.

SUMMARY

Five experiments have been conducted to investigate the pressure-tube circumferential temperature distribution and deformation as a function of heater power, coolant level (coolant flow rate), and pressure. None of the pressure tubes failed as a result of the circumferential temperature gradients induced by the stratified flow. The pressure tubes all ballooned preferentially at the top, resulting in near or direct contact with the calandria tube and were cooled by the moderator. Results of the five experiments lead to the following conclusions:

1. A lower coolant level results in a larger top-to-bottom temperature difference (ΔT) but not necessarily a steeper temperature gradient on the pressure-tube wall.
2. A higher heater power results in both a larger ΔT and steeper circumferential temperature gradient.
3. A higher internal pressure results in both a smaller ΔT and circumferential temperature gradient.
4. Pressure-tube deformation has a significant cooling effect on the pressure tube and therefore affects the circumferential temperature distribution. Modelling of this effect is essential in the calculation of circumferential temperature distribution and pressure-tube deformation in LOCA situations.
5. Pressure tube ballooned at the top in all five tests. Only in Test 5 where both the heater power and channel pressure were high did the pressure tube come into contact with the calandria tube.

Data collected in these experiments will continue to be used to verify computer codes like CATHENA, SMARTT and AMPTRACT.

ACKNOWLEDGEMENTS

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REFERENCES


EXPERIMENTS TO DETERMINE THE THERMAL-MECHANICAL RESPONSE WHEN MOLTEN ZIRCALOY-4 FLOWS ONTO A BALLOONED PRESSURE TUBE

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ABSTRACT

An experimental program has been set up to investigate the thermal-mechanical behaviour of a fuel channel when molten Zircaloy-4 (Zr-4) from a fuel bundle flows onto a ballooned pressure tube. The first two tests of the series were carried out with a pressure of 3 MPa inside the pressure tube and subcooling on the calandria tube of about 20°C. Results show that the calandria tube dries out directly under the Zr-4 melt but quickly rewets within 22 s. The maximum calandria tube temperature measured directly under the molten Zr-4 droplet was 900°C. This paper describes the experimental setup and summarizes results from these first two tests.

INTRODUCTION

During a postulated Loss-of-Coolant Accident (LOCA) with impaired emergency cooling, decay heat in fuel bundles may raise fuel sheath bundle end plate and end cap temperatures above the threshold for the exothermic Zircaloy/steam reaction. Temperatures resulting from this exothermic reaction may exceed the melting point of unoxidized Zircaloy-4 (Zr-4). Experiments performed at Whiteshell Nuclear Research Establishment (WNRE) (1) and Westinghouse Canada Inc. (2) have shown that molten Zr-4 from the fuel sheaths will not flow onto the pressure tube because Zr-4 with dissolved oxygen wets uranium dioxide (UO2). Also, for the moderate bundle heat-up rates expected in a LOCA, end plates and end caps would oxidize before flowing onto the pressure tube. However, bundle experiments at Westinghouse Canada Inc. (2) showed that for fast heat-up rates, end plates and end caps can potentially melt and flow onto the surrounding pressure tube.

An experimental program has been set up to study the heat transfer between the molten Zr-4 and the fuel channel. The program's main objective is to study the thermal-mechanical output when molten Zr-4 flows onto a ballooned pressure tube. The data collected in these experiments will be used for modelling purposes using the two-dimensional computer code MINI-SMARTT. (3) The objective of this paper is to present results from the first two tests of this experimental program.

The experimental program was funded by the CANDU Owners Group (COG).

MATERIALS AND METHODS

Experimental Apparatus

The apparatus consisted of a 1580-mm long section of Zr-2.5 wt% Nb pressure tube mounted inside a 1730-mm long Zr-2 calandria tube (Figure 1a). The calandria tube was surrounded by heated, non-flowing water in an open tank. The top surface of the calandria tube was covered by at least 400 mm of water throughout the experiment. The walls of the tank were equipped with plexiglass windows for viewing.

The heater was a 570-mm long, 38-mm diameter graphite rod concentrically located inside the fuel channel assembly (Figure 1b). The heater was equipped with a tapered hole containing a funnel-shaped graphite crucible. The crucible holds the Zr-4 slug and forms a pouring funnel for the molten material.

FIGURE 1: SCHEMATIC OF TEST RIG USED FOR THE MOLTEN ZR-4/PRESSURE TUBE INTERACTION EXPERIMENTS; (a) SIDE VIEW AND (b) CROSS SECTION A-A
Power was supplied by a 500 A D.C. power supply. Power supplied to the heater assembly was determined by measuring current and voltage at the outer ends of the stainless-steel bus bars. It was assumed that the power lost to the bus bars was small compared to power generated in the graphite as the resistivity of graphite is approximately 10 times that of stainless steel.

In total, 29 thermocouples (T/C) were used to monitor the temperatures of the test section: 24 on the outer surface of the calandria tube, and 5 on the inner surface of the pressure tube. A cross section through the test section shows the location of various thermocouples located at the test section centreline (Figure 1b). A schematic of the thermocouple grid attached to the outside surface of the calandria tube, directly under the graphite crucible, is shown in Figure 2.

The thermocouples on the inner surface of the pressure tube were held in place by Zircaloy shims spot welded over the ends of the thermocouples. These shims served as thermal radiation shields for the thermocouples. Of the five thermocouples inside the pressure tube, one was placed in a 0.6-mm diameter hole drilled at a 45° angle into the pressure tube wall under the graphite crucible. The other four thermocouples rested on the pressure tube wall, one under the graphite crucible and the other three slightly offset from the drop zone.

Along with the above thermocouples, two others were used to measure water temperature in the tank surrounding the calandria tube. These thermocouples were located 50 mm away from the calandria tube.

The entire test was recorded using a video camera located outside the Plexiglass window installed in the moderator tank wall. A mirror was positioned under the test section so that any boiling on the underside of the calandria tube could be recorded.

**Experimental Procedures**

The following procedure was used for the tests:

1. The inside of the pressure tube was purged with argon. The argon pressure inside the pressure tube was maintained at 3 ± 0.02 MPa.
2. The water surrounding the calandria tube was heated to either 76°C (first test) or 80°C (second test).
3. The annulus between the pressure and calandria tubes was purged with argon (first test) or CO₂ (second test). The purge flow was high initially, but reduced to near stagnation just before the start of the test.
4. Power to the test section was turned on and ramped up to 50 kW. Power was kept near this level until the pressure tube ballooned into full contact with the calandria tube. Full contact was determined by watching the video of the test section.
5. After the pressure tube had fully ballooned into calandria tube contact, the power was ramped to produce the molten Zr-4 droplet.
6. Power was decreased and the test terminated after the droplet had landed on the pressure tube and any dryout caused by the molten Zr-4 had revet. The time of the droplet/pressure tube contact was determined by watching the output of the thermocouples directly under the graphite crucible.

**Experimental Results**

For the sake of brevity the results from the second test will be discussed in detail and results from the first test summarized.

Heater power and pressure tube temperature transients from the second test are presented in Figure 3. In this test, power to the heater was increased from 0 to 49 kW over 290 s and kept near this level for 70 s. During this period, pressure tube wall temperatures increased at a ramp rate of 9.3°C/s (Figure 3b). The pressure tube ballooned into full contact with the calandria tube when pressure tube temperatures reached 790°C. Upon contact, pressure tube inner surface temperatures rapidly decreased. Following pressure tube/calandria tube contact the calandria tube underwent a short period of patchy dryout between 327 and 339 s (Figure 4). The maximum recorded calandria tube temperature in this period was 510°C. The pressure tube temperatures leveled off at approximately 650°C for this short period, then decreased to as low as 320°C. The 12-s period of patchy dryout for a contact temperature of 770°C and a subcooling of 20°C was consistent with previous contact boiling tests.

After contact, when the calandria tube returned to stable nucleate boiling, power to the heater was ramped up to 73 kW. At 428.2 s a molten Zr-4 droplet was recorded to have dropped onto the pressure tube. This was shown by a spike in the output from the thermocouples located directly under the graphite.
crucible (T/Cs 27 and 28) (Figure 3b). Post-test examination determined that the droplet was 19.1 g, 8.4-mm thick and roughly 26 mm in diameter (averaged).

Heat-up rates on the underside of the calandria tube caused by the molten Zr-4 droplet were as high as 590°C/s (Figure 5). These heat-up rates were greatest directly under the molten Zr-4 contact area and decreased with increasing distance from the drop zone. Maximum calandria tube temperatures measured during this thermal transient were estimated at 900°C and occurred 3.5 s after the droplet contacted the pressure tube. This maximum was an estimate as the thermocouples used on the calandria tube were set up using an amplifier with a maximum output that corresponded to an 800°C temperature.

Three-dimensional temperature plots (Figures 6 and 7) and two-dimensional isothermal contour plots (Figure 8) were developed to help visualize the thermal transients on the calandria tube. These plots were produced using readings from thermocouple locations 1 to 20 (Figure 2) and the following:

1) Symmetry about the Zr-4 drop zone was assumed; and
2) The SAS spline subroutine G3GRID [4] was used to interpolate between data points, thus smoothing out the profiles.

The plots show the formation of a hot spot directly under the Zr-4 droplet that lasted 22 s. Moderator temperatures during the second test remained constant between 80 and 81°C.

The video record of the test indicated that the pressure tube first came into contact with the calandria tube near the top midplane. Contact then spread over the rest of the heated zone. The video clearly indicated that the calandria tube was in stable nucleate boiling after pressure tube contact. Upon closer examination of the video a small dryout patch (25 mm in diameter) was observed when the molten Zr-4 droplet contacted the ballooned pressure tube. This small dryout patch rewet after 22 s but left a discoloured stain on the calandria tube surface.
Important results from the two tests are summarized in Table 1 for easy comparison. The results from both tests indicate that a 20-g molten Zr-4 droplet landing on a ballooned pressure tube has negligible effect on channel integrity, when the pressure tube is internally pressurized to 3 MPa.

![Figure 8: Two-dimensional isothermal contour plots for the underside of the calandria tube, directly under the molten Zr-4 droplet (second test). The number in brackets is elapsed time since droplet landed on pressure tube.]

<table>
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<tr>
<th>Test No.</th>
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</table>

![Table 1: Summary of test parameters and results from the first two tests of the molten Zr-4/pressure tube interaction series.]

**CONCLUSIONS**

The following conclusions can be drawn from these experiments:

1. The flowing of molten Zircaloy onto a ballooned pressure tube, internally pressurized to 3 MPa, resulted in a short period of dryout on the calandria tube followed by a quick rewet. The size of the dryout patch corresponded to the size of the melt.

2. A 19.1-g molten Zr-4 droplet landing on a ballooned pressure tube, internally pressurized to 3 MPa, had no noticeable effect on channel integrity.

3. No noticeable deformation of the calandria tube occurred as a result of the molten Zr-4.

**REFERENCES**


OVERPRESSURES AND TIME SCALES ASSOCIATED WITH HYDROGEN COMBUSTION

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ABSTRACT

Since 1978 an experimental program has been underway at the Whiteshell Nuclear Research Establishment to provide fundamental combustion data and large-scale model verification in support of analysis of postulated post-accident hydrogen combustion situations in CANDU® containments and calandria. The extent of the hazard from accidental hydrogen combustion is determined primarily by the magnitude and duration of the combustion overpressure. This paper reviews the factors contributing to combustion pressure development and the effectiveness of mitigation by venting, and presents some illustrative experimental results.

INTRODUCTION

Postulated accidents in CANDU® reactors lead to potential hydrogen combustion situations in the calandria and the containment (1,2). The concerns associated with hydrogen combustion are (a) threat to the calandria and containment integrity, (b) threat to the safety equipment, and (c) potential pressure-driven increase in the release of activity from containment to the outside atmosphere through leak paths. The extent of these threats is directly influenced by the magnitude of the overpressures and their duration (3). Analyses of these threats requires verified tools to predict combustion pressure transients.

An experimental program in support of these analyses, under way at the Whiteshell Nuclear Research Establishment (WNRE) since 1978, has provided fundamental combustion data and large-scale verification of engineering models. This paper reviews the factors contributing to combustion pressure development and mitigation, illustrated by experimental results from the Containment Test Facility (CTF) at WNRE.

The pressure rise associated with a confined combustion is due to the expansion of gases heated by the chemical reaction. If the deflagration speed is small with respect to the speed of sound, the combustion pressure is bounded by the adiabatic, isochoric, complete combustion (AICC) pressure, which can be determined from equilibrium thermodynamic calculations. The AICC pressure is the highest for stoichiometric mixtures and decreases upon dilution with air and steam (4), Figure 1. The actual peak combustion pressure is lower than the AICC pressure (see Figure 2) if there is incomplete combustion or heat loss (non-adiabatic) or venting (non-isochoric) during the burn. Burning rates are thus inextricable from the consideration of combustion pressure development. Only very fast, complete burns will result in a peak pressure approaching the AICC pressure.

EXTENT OF COMBUSTION

The AICC pressure is reduced in proportion to the fraction of residual unburnt combustible. Incomplete combustion is common for marginally flammable mixtures or incompletely mixed volumes. Typically, low fractions of hydrogen are burned at hydrogen concentrations close to the upward propagation limits (nominally 4 - 5% H₂ in dry air); complete combustion occurs for hydrogen concentrations approaching the downward propagation limits (- 8 - 9% in dry air) (see Figure 3). For mixtures between the upward and downward limits, the igniter location has a large effect on the extent of combustion, with bottom ignition producing the most complete burning (5).
Initial turbulence greatly increases the extent of combustion for mixtures with a hydrogen concentration between the upward and downward limits (5). For instance, it increased the extent of combustion from 26% for quiescent mixtures to 83% for mixtures of 6% H₂-air (see Figure 4). Also, the greater extent of combustion observed for these mixtures in large volumes is thought to be related to turbulence arising during the combustion process. But turbulence quenches combustion at lower hydrogen concentrations.

**Burning Rate**

The burning rate is defined as the mass rate of consumption of the combustible,

\[ m_u = \rho_u S_u A_f \]

where \( A_f \) is the average flame surface area normal to the flow, \( \rho_u \) is the unburnt gas density, \( S_u \) is the laminar burning velocity and \( \phi \) is a turbulence factor (6). The burning rate, and thus the instantaneous pressure rise, depends on the initial thermodynamic and gas dynamic conditions of the mixture (composition, temperature, pressure, turbulence), and the boundary conditions (the size and geometry of the confinement, topography and details of obstructions).

**Thermodynamic Conditions**

The minimum burning rate is that corresponding to the laminar burning velocity (i.e., \( \phi = 1 \)). The laminar burning velocity is controlled by the rate of chemical reaction and the rate of heat and species diffusion, and hence, depends only on the initial thermodynamic conditions of the mixture. It is a fundamental quantity widely used as an indicator of mixture's "sensitivity". There exists a good experimental consensus for laminar burning velocities of H₂-air mixtures. The influence of steam on H₂-air laminar flames has been examined in detail at UNRE. Measured laminar burning velocities of H₂-air-steam mixtures, Figure 5, show the sensitivity of burning velocity to H₂ composition and the lowering of burning velocity with steam dilution. Steam has a distinct catalytic effect on burning velocity that partially offsets the effect of its increased heat capacity, which lowers flame temperature and, consequently, the burning velocity (7). The presence of steam lowers the equilibrium combustion temperature and, hence, the peak pressure (8).
Turbulence

Turbulence present in real systems will always result in a higher burning rate compared to laminar burning. Turbulence contributes to combustion pressure by increasing the burning rate and the extent of combustion. It increases the burning velocity by increased transfer of heat and active species to the unburnt gas and increased flame surface area due to flame wrinkling. Intense turbulence can quench burning (9). Turbulence may be present initially, such as in the vicinity of fans, or may arise during the combustion process. Typically, spherically expanding laminar flame fronts become turbulent owing to flow and diffusion instabilities caused by the large density difference between burnt and unburnt gases. As this is illustrated by Figure 6, observed flame speeds of quiescent mixtures are consistently higher than the laminar prediction.

Lean H₂-air flames are particularly prone to these instabilities. Burning velocities of H₂-air-steam mixtures have been recently measured at WNRE as a function of composition and turbulence intensity. Figure 7 illustrates the effect of initial turbulence on the burning velocity of H₂-air mixtures. With increasing intensity of turbulence, the burning velocity increases to reach a maximum beyond which it quenches (10,11). In Figure 7b, turbulence quenching is evident in mixtures containing less than 10% H₂. For initial turbulence in H₂-air mixtures at atmospheric temperature and pressure, φ has a maximum of about 16 (11), corresponding to burning velocities greater than 30 m·s⁻¹ and flame speeds over 200 m·s⁻¹ for stoichiometric mixtures. Confined combustion-driven flow over repeated obstacles can produce greater turbulence intensities (12), which can lead to even greater flame speeds for near-stoichiometric mixtures. Severely accelerated flames are discussed later in connection with transition to detonation.

VENTING

Postulated post-LOCA (loss-of-coolant accident) hydrogen combustion in reactor vaults would be vented to adjacent containment volumes or to the vacuum building (1). Venting relieves the peak overpressure and shortens its duration. The effectiveness of venting depends on the size of the vent, its location with respect to the ignition source and the burning rate. Venting can, however, cause gas dynamic instabilities (13) that contribute to flame acceleration. Prediction of vented combustion pressures requires an in-depth understanding of feedback mechanisms and a rigorous verification on a meaningful scale.

At WNRE, using a 6.3-m³ sphere vented combustion behaviour of H₂-air-steam mixtures has been investigated as a function of mixture composition, fan turbulence, vent diameter and ignition location (14,15). Results have been compared with predictions of a one-dimensional analytical model VENT (16).

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Generally, it was observed that venting was effective in reducing the peak pressure at low hydrogen concentrations and large vent areas, i.e., conditions for which combustion times ($t_c$) were longer than the characteristic vent times ($t_v$, $t_c/t_v > 1$). Vent effectiveness ($\epsilon$) is defined in terms of peak pressures:

$$\epsilon = \frac{(P_a - P_v)}{(P_a - P_t)} = f(t_c/t_v)$$

where $P_a$ is the AICC pressure, $P_v$ is the peak pressure obtained with venting and $P_t$ is the ambient pressure. Figure 8 compares the combustion pressure histories for three vent sizes for a low burning rate mixture ($10\% H_2$, air), covering a range for the elementary vent parameter, $t_c/t_v$, of 1.0 to 10.0. Figure 9 shows peak pressures observed over the range of 5 to 20% $H_2$ in air and the relief provided by venting.

A more fundamental vent parameter, $A/S_o$, proportional to $t_c/t_v$, was defined by Bradley and Mitcheson (17), where $A$ is the ratio of vent area to surface area of the vessel and $S$ is a non-dimensional flame speed.

A conservative estimate of vent effectiveness ($\epsilon$) is obtained with the following equations:

$$\epsilon < 0.01, \ \epsilon = 0$$

$$0.01 < \epsilon < 0.5, \ \epsilon = 0.46 + 0.1 \ln(\epsilon/S_o)$$

$$0.5 < \epsilon < 10, \ \epsilon = 0.54 + 0.22 \ln(\epsilon/S_o)$$

The above empirical correlations for vent effectiveness are for peak pressures only. Estimates of the rates of pressure rise require a more detailed calculation.

The VENT model, in use at WNRE since 1982 (16), estimates the peak pressures and the rates of pressure rise. The model assumes a spherical flame front, subsonic flame velocity (i.e., uniform pressure throughout the vessel) and uniform venting in all directions. The calculation is based on conservation of mass and prescribed rates of burning and venting.

The predictions of the VENT model have been matched with experimental results from the CTF using two values of turbulence factor ($\phi$), one prior to the onset of instability (due to vent opening) and the other after the onset of instability. Figure 10 shows a comparison of experimental results and theoretical predictions for 15% $H_2$-air vented combustion. Pre-vent values of $\phi$ range from unity to 1.6 and are comparable to the flame-induced burn rate enhancement observed previously with unvented combustion. Post-vent values of $\phi$ range from 1.3 to 3.4.

The maximum vent opening in the CTF is 45 cm in diameter, corresponding to a dimensionless vent size ($A/V^{2/3}$) of 0.05. Values of interest in containment analyses are in the range 0.1-0.2. The above results suggest that instabilities that develop in large-scale systems, as shown in the investigations with hydrocarbons (18), may not be observed in small-scale systems. At the present time preparations are underway at WNRE for larger-scale (10 times) $H_2$-air experiments in an open-air facility to verify scaling laws with the $A/S_o$ ratios of interest.

Accelerated Deflagrations and Detonations

Combustion waves are classified as either deflagrations or detonations. "Explosion" is an ambiguous term and should be avoided. Until this point we have considered only deflagrations. This
mode of combustion covers a vast domain of combustion behaviour in which flame propagation is by diffusion of heat and active species. Deflagrations are typically, but not necessarily of subsonic velocity. A detonation, by contrast, propagates as a chemically supported shock wave and is supersonic with respect to the unburnt gas. Because of the high velocity, equalization of pressure in the surrounding volume does not occur. Global detonations in containment are not considered a possibility, owing to insufficient amount of available hydrogen, but the possibility of detonations in unmixed local volumes of a few cubic metres cannot be ruled out. The pressure across the detonation wave is capable of causing local damage to equipment or material in its path.

Direct initiation of detonation requires a strong ignition source, unlikely to be present in containment. However, if the mixture is capable of sustaining a detonation, and a deflagration is initiated in such a mixture, mechanisms exist for deflagration-to-detonation transition (DDT). A useful measure of detonation sensitivity is the detonation cell width. An exhaustive study of detonation cell widths with a predictive theoretical model has been completed at WNRE (19).

Obstacles in the flame path can produce turbulence in the unburnt gas flow. Large and fine-scale turbulence in the wake of the obstacles cause flame stretching and turbulent burning. Confined combustion-driven flow over repeated obstacles can produce very high flow velocities and turbulence intensities, causing much higher flame speeds than that produced by initial turbulence (18). If the mixture is not sufficiently reactive, the flame will quench as a result of excessive flame stretching and rapid cooling by turbulent mixing. Otherwise, the flame will continue to be accelerated. While there is no defined limit of initial and boundary conditions for flame acceleration to occur, flame acceleration for insensitive mixtures is limited by quenching mechanisms. In large-scale tests with a geometry designed to produce acceleration, significant flame acceleration was not observed below 12% hydrogen in air (20).

For sensitive mixtures, it is possible to have accelerated deflagrations propagating at several hundred metres per second, with strong associated shock waves and exhibiting detonation-like properties but with the structure of a deflagration (21). The shock waves compress and preheat the unburnt gas and the flow generated in the unburnt gas creates intense turbulence. Under such conditions, DDT has been observed. A condition for DDT in a shock-obstacle collision has been identified (22) whereby the transient shock heating of gases is sufficient to initiate detonation. The incident shocks associated with a fast deflagration may become reinforced (focused) in some obstacle configurations and produce transition (23).

The capability to predict whether transition will occur for a given set of initial and boundary conditions has not yet been achieved. The establishment of detonation cell widths and of some transition conditions at WNRE marks significant progress towards that goal. In the absence of complete predictive capability, (24) approximate methodologies have been devised, classifying the gas mixtures and confinement geometries in terms of likelihood of producing transition.

SUMMARY

The factors contributing to the magnitude and duration of hydrogen combustion pressure have been reviewed. The pressures for slow combustion are bounded by the theoretical constant volume combustion pressure (AICC). The theoretical peak pressures are not achieved if there is incomplete combustion, heat loss or venting during the burn. Thus, the burning rate determines the time available for cooling or relief by venting. Burning rates are strongly influenced by turbulence. Turbulence may be initially present or produced during combustion by flow instabilities or interaction with obstacles. Severely accelerated flames can produce conditions whereby deflagration-to-detonation transition becomes possible. Venting provides relief of combustion pressures if the time required for venting is comparable to the burn time.

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10th Annual Conference

OTTAWA, ONTARIO
JUNE 4 - 7, 1989, juin 4 - 7

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10th Annual Conference
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Proceedings
Volume 3
Foreword

As in previous years, the Annual Canadian Nuclear Society Conference was held in conjunction with the CNA Conference. This year, to recognize the Fiftieth Anniversary of the Discovery of Nuclear Fission, a special plenary Symposium, co-sponsored by the Canadian Nuclear Society, Canadian Nuclear Association, and other Canadian Learned Societies, was held as part of the CNS Annual Conference. The proceedings of the Special Symposium are published as Volume 1 of the Proceedings of the 10th Annual Conference of the Canadian Nuclear Society.

Volumes 2 and 3 contain the proceedings of the 17 technical sessions from the 10th Annual Conference of the Canadian Nuclear Society. We are pleased to include several papers which had been accepted but which, because of travel difficulties, were not available for presentation at the Conference.

The papers for these proceedings were prepared on standard forms supplied by the Canadian Nuclear Society and are generally published as submitted by the authors. Responsibility for the content of each paper rests solely with the author.

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The success of the 10th Annual Conference was due not only to the efforts of authors and other participants at the Conference, but also to the assistance of the numerous CNS members and others who gave of their time to participate in reviewing the large number of papers received. The support of the CNA organizing committee under the chairmanship of R. Veilleux also contributed significantly to the success of this conference. The secretarial assistance of Mrs. V. Mussell is also gratefully acknowledged.

P.J. Fehrenbach, T.J. Jamieson
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Comparison of Homogeneous and Two-Fluid Simulations of a Large-Break LOCA in a CANDU
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ABSTRACT

The reactor cooling system that evolved in the development of AMPS (Autonomous Marine Power Source) was intended in the first instance to contribute inherent reactor safety in nuclear marine propulsion.

The system, believed to have potentially broader application in reactor design, is described generically. Among the key features of the system is assured passive cooling in accident scenarios, and self-regulation of the interaction between the hot circulating primary coolant and the reserve coolant under normal conditions.

INTRODUCTION

The AMPS nuclear electricity generating plant [1,2], currently under active development [3,4,5] by the ECS Group of Companies, incorporates a reactor of new generic design belonging to the emerging general class of reactors strongly identified with inherent safety [6]. One requirement of this reactor class is a passive decay heat removal system. In this paper, the basis of the passive cooling system of the AMPS reactor is described within the broad context of inherent reactor safety.

The first goal of the AMPS development [7,8,9] has been to make available for small and medium-sized submarines the true air-independence capability that only nuclear power can provide. This necessitates the creation and adaptation of the technologies of inherent reactor safety specifically for the marine environment; in fact, nuclear power plants employing such technologies may prove to be the only practicable route to achieving this goal. Although the evolution of the AMPS concept is conditioned by the specific requirements of this first application, it is anticipated that many aspects of the particular design approach of the AMPS reactor may be applied to advantage in less constrained design environments as well.

INHERENT SAFETY

The phrase "inherent safety" is now commonly used in nuclear reactor design to designate a reactor concept as having facilities for both passive reactor shutdown and passive decay heat removal. The word "passive" is used to denote that such facilities will assuredly function to protect the reactor plant from damage in the face of an extended range of design basis accidents, independently of whether short-term mitigative action can be served by either active safety systems or operator intervention. To assure that the passive systems are sufficiently endowed with the attributes of reliability and availability, and that these attributes are readily identifiable to consumers, the passive systems are generally based on physical principles which are "intrinsic" or "process-inherent" to the normal functioning of the reactor system. Thus, for example, passive reactor shutdown based on the direct negative influence of core temperature increases on neutronic multiplication may be credited as process-inherent. On the other hand, a passive shutdown cooling system in which coolant circulates through the core by natural convection may be considered to fall short of being truly passive if it relies on the opening of a check valve to permit the onset of convective cooling.

In any case, "inherent safety" is not a quality to be assigned to a particular reactor design, system or subsystem with the expectation that it can receive special acceptance by the public, the customer or the regulating authorities. Rather, inherent safety is a design approach which may be adopted in a given nuclear power project for the purpose of best satisfying (or perhaps uniquely satisfying) the particular set of requirements that may arise from new technological and sociological circumstances of the particular application.

It would appear that there exists a general perception that to embark on any nuclear project involving new technological concepts is to invite an extremely long development program which in the end may not be successful. However, most of the reactor concepts adopting the inherent safety approach are built up largely of components that are proven through many years of experience in existing reactors. Moreover, the remaining novel aspects of such reactors are drawn from well-understood technologies such as neutronics and thermalhydraulics which, consistent with the overall inherent safety approach, can be applied conservatively within the boundaries of current understanding. In fact, the minimizing of technological, as well as environmental and other risks, is itself a goal of the inherent safety approach.

Most of the reactor concepts fitting the inherent safety description have been proposed during the past decade. They have ranged in potential application from central plants for electricity generation and district heating, to special-purpose reactors for local heating applications and marine propulsion. In terms of reactor type, they have included pressurized water reactors, unpressurized water reactors, high-temperature gas reactors, and liquid metal fast reactors and have
originated and been promoted in a variety of countries including Canada, Federal Republic of Germany, Sweden, Switzerland, and U.S.A. Most of the concepts advanced [10] have been in the small to medium size reactor category. This correlation stems partly from necessity; during the trend towards smaller unit capacities, inherent safety offers a means of controlling costs through simplification of equipment and procedure. It is also a matter of opportunity; plants of lower output capacity are generally more amenable to inherent safety in design.

THE AMPS REACTOR

Beginning in 1985, the AMPS reactor, as a major component of the Autonomous Marine Power Source, became a subject of development [7] by ECS Power Systems Inc. Given its position at the low end of the scale of nuclear electric generating capacity, and the severe design constraints imposed by its compact marine operating environment, AMPS was deemed a prime candidate for the application of inherent safety principles.

Because of the unique circumstances of its destined application, the AMPS reactor concept could not be based directly on any specific previous concept. An appreciation of the objectives and the means of achieving inherent safety goals in earlier and concurrent concept developments, such as SLOWPOX [11], PIUS [12], and TPS [13], provided stimulus. As will be seen in the present paper, however, among the distinctive inherent safety features of the AMPS reactor cooling system are its barrier-free segregation of the circulating primary coolant from the constantly available (passive) reserve coolant, and its ability to self-regulate the maintenance of this segregation in normal operation by passive means without compromising the passive cooling operation when it is needed in accident conditions.

AMPS COOLING SYSTEM PRINCIPAL COMPONENTS

The AMPS cooling system evolved as an essential feature of reactor designs intended specifically for small scale marine use. However, the AMPS system may be viewed as more broadly applicable, especially in other applications where a high degree of inherent safety is needed. Consequently, the AMPS cooling system is described in the present paper in a generic form only. It is described, in first instance, in its most elementary form with reference to the reactor plant depicted in the schematic diagram of Figure 1. The ready adaption of the AMPS scheme, in advantageously meeting special application requirements, will be demonstrated with reference to AMPS reactor variations exemplified schematically in Figures 2 and 3.

Primary Circuit

In the reactor plant shown in Figure 1, the primary heat transport circuit comprises the equipment components normally found in a typical reactor cooling system of the closed circuit variety. Included are the reactor core with inlet and outlet plena, located inside the reactor vessel. Also included are the inlet and outlet ducts (with their associated hydrodynamic ports), connecting by means of inlet and outlet conduits, through penetrations in the walls of the

reserves tank to connect with external components, principally the main circulating pump and combinations of heat exchanging components, such as preheaters, evaporators, steam generators, or simple heat exchangers. The part of the primary circuit that embraces these latter components could be located within the reserve coolant tank along with the reactor vessel and connecting ducts and conduits, if required for safety reasons in a particular design. The solid arrows in Figure 1 indicate the normal path of coolant flow in the primary circuit during normal operation.

Reserve Coolant Tank

The reserve coolant is maintained at suitably low standby temperatures with the aid of reserve tank coolers (not shown in the diagrams) which transport to the external environment any residual heat transferred from the primary circuit to the reserve tank during normal operations, and also the decay heat transferred to the reserve tank by natural convection in the passive cooling circuit, denoted by the broken arrows in Figure 1, following emergency shutdown. The reserve tank coolers may operate on either active or passive principles but, for consistency with the passive safety principles, the capacity of the passive component of reserve tank cooling would have to be adequate for the safe removal of decay heat over an indefinitely long period following reactor shutdown.

Over the range of operating and shutdown conditions anticipated for an AMPS reactor as typified in Figure 1, the coolant circulating within the primary circuit will be at temperatures considerably higher than that of the reserve coolant maintained in the reserve coolant tank. Thermal insulation of a suitable design is shown attached to the primary circuit boundaries shared with the reserve coolant tank. The insulation enhances the energy delivery efficiency of the reactor and facilitates the maintaining of the reserve coolant at a standby temperature sufficiently low to support safety objectives.
Ports and Their Basic Properties

The elements of the primary heat transport circuit not ordinarily found in a reactor cooling system, but which are a key feature of the AMPS system, are the inlet and outlet hydrodynamic ports, located in the circuit just upstream from the inlet duct and downstream from the outlet duct, respectively. The hydrodynamic ports are shown in basic form in Figure 1. Although the ports may be viewed as series components of the primary circuit, they represent no physical barrier to flow in the passive cooling circuit. This is because the hydrodynamic ports have been developed to exhibit specific hydrodynamic properties when correctly incorporated in the design of the reactor primary heat transport circuit. The most important of these properties are that the ports: (i) support a high degree of continuity of primary circuit flow (axial in the port) at normal reactor operating conditions, (ii) offer large resistances against any tendency for either combining or dividing branch flows to develop through the lateral slots, relative to the axial flows, as is particularly required when the orientation or operating parameters of the reactor deviate inadvertently from normal, and (iii) offer relatively small resistance to large branch flows of either kind in severely abnormal or accident conditions, such as when the main circulating pump has stopped.

Pressurizing Arrangements

The primary heat transport system pressure boundary includes the reserve coolant tank and those components of the primary circuit indicated in Figure 1 to lie outside the boundary of the reserve coolant tank. The reactor of Figure 1 is depicted as being maintained at an elevated operating pressure by means of a pressurizer connected to the reserve coolant tank. If the reactor is to be operated nominally at atmospheric pressure, either the pressurizer or the reserve tank itself is left open, or otherwise vented to atmosphere. The so-called (open) pool-type reactors automatically fall into this latter category.

The principal objective of pressurizing primary coolant in any reactor plant is to permit the transport of heat at elevated temperatures, in the interest of extracting high quality heat for efficient thermodynamic conversion or other use, while retaining the advantages of the coolant in its liquid state. In the reactor plant of Figure 1, only the coolant circulating in the primary circuit has a requirement for elevated temperatures in consideration of plant thermodynamics. Therefore, if the ports provide suitable operational segregation between the circulating coolant and the reserve coolant, the latter may be maintained at relatively low temperature. Ideally, with the support of the reserve tank coolers, it is maintained at temperatures much lower than the saturation temperature corresponding to atmospheric pressure. This arrangement for the reserve coolant temperature to remain low not only limits the quantity of superheated coolant that potentially would be released in the event of an accidental depressurizing of the cooling system, but the associated thermal ballast enhances the passive cooling system’s potential in mitigating accidents.

Because the hydrodynamic ports provide at all times barrier-free interfaces between the reserve tank and the primary circuit, the operating pressures of these two parts of the system are intimately related. This fact has several design consequences: (i) the degree of pressurization of the primary system is unimportant in a description of the basic normal functioning of the cooling system, (ii) whether the pressurizer is connected to the reserve coolant tank or to the primary circuit is optional, (iii) the nominal operating pressure of the entire heat transport system is required to be conservatively in excess of that required to the prevent boiling at the point of the hottest coolant within the primary circuit, and (iv) in the special case of the reactor not being pressurized to higher than the ambient pressure, the reactor is constrained to operate with core temperatures not exceeding the saturation temperature at atmospheric pressure.

AMPS COOLING SYSTEM FUNCTION

The above-mentioned properties of the hydrodynamic ports are manifest in the AMPS reactor system as represented in its most basic form in Figure 1. A description of the operation of the cooling system of this plant follows.

Normal Operation

During normal operation a steady flow is maintained through the primary circuit by the continuous operation of the circulating pump. The pumping head equals the algebraic sum of the pressure changes across individual circuit components, including (i) the heat exchanger components located externally to the reserve cooling tank, and (ii) the components forming that part of the circuit delineated by the solid arrows and shown residing inside the reserve coolant tank.

Hydrodynamic Ports in Normal Operation. To provide constant availability of passive cooling by process-inherent means during normal reactor operation, the hydrodynamic ports are designed to present at no time any physical barrier against the flow of coolant between the primary cooling circuit and the reserve coolant tank. However, to avoid the undesirable transport of either thermal energy or radioisotopes from the primary circuit into the reserve tank during normal operation, it is important to reduce to acceptable levels the tendency for such exchange flow, and also the magnitude of such flow, should it occur. The limiting of exchange flow is essential also in preserving the option of safety shutdown induced by the onset of passive cooling, should that option be desirable in future AMPS reactors. In this option it is important to inhibit neutron absorbing nuclides dissolving in the reserve coolant from displacing any of the dilute coolant of the primary circuit prematurely.

Design Requirement for Zero Exchange Flow. The tendency for such exchange flow is reduced to acceptable proportions, in the first instance, by the incorporation of the port concept in the design of the primary circuit in accordance with a specific design requirement. The requirement, applying only to that part of the circuit within the reserve coolant tank of Figure 1, is stated as follows: During normal reactor operation, the net pressure from all effects accumula-
tive around the passive cooling circuit delineated by the broken arrows in Figure 1, and tending to support net flow around this circuit, must be zero.

The basic parameters of the primary circuit that must be taken into account in designing for zero net accumulative pressure are the primary circuit mass flow rate, the coolant temperature profile in the primary circuit, the bulk temperature of the coolant of the reserve coolant tank, and the pressure drop due to the primary circuit mass flow through the core and the other components in the flow path from the inlet port to the outlet port. These parameters are determined as the normal operating design values, selected primarily on the basis of optimal energy production efficiency and safety in the intended application.

Criterion for Zero Net Pressure. More specifically, the criterion for meeting the requirement of zero net accumulative pressure may be expressed in terms of the values of the axial flow areas associated with hydrodynamic ports. The flow areas are chosen so that the change in dynamic pressure (velocity head change), associated with a constant mass flow passing from the vicinity of the inlet port to the vicinity of the outlet port, is equal to the deficit by which the net elevation pressure, assessed accumulatively around the passive cooling circuit, fails to match the resistive pressure losses associated with the same constant flow of primary coolant through the core. The correct assessment of the net elevation pressure is based on the vertical port separation L as indicated in Figure 1, the between-port coolant temperature profiles (and hence the density profiles) along paths through both the reactor vessel and the reserve tank. The resistive losses are due largely to the hydrodynamic energy dissipated in the core itself.

Base Case. It is clear that within the stated criterion for meeting the zero net accumulative pressure requirement, a great deal of flexibility is available in the thermohydraulic design of reactors based on the one depicted in Figure 1. For example, if space is not a consideration and the temperatures in the profile extending through the core and between the two ports are considerably higher than those that are characteristic of the bulk of the reserve coolant, as will be the case for the reasons of thermal efficiency and reactor safety already discussed, the requirement of zero net accumulative pressure may be met without prescribing any net change in flow area. In this case, the design criterion reduces to that of the net elevation pressure being equal to the resistive pressure losses along the flow path through the core and between the ports. This criterion is met by the suitable selection of the vertical separation L between the ports.

Restricted Space. In a different example, space limitations are assumed to impose a restriction on the reactor plant and hence on the vertical separation L between the ports in the reactor depicted in Figure 1. For a prescribed disposition of normal operating temperatures throughout the system, this reduction can be permitted, while preserving the zero net accumulative pressure requirement, by choosing the axial flow area of the outlet port to be sufficiently larger than that of the inlet port to create a net dynamic pressure decrement in the flow passing from the inlet port to the outlet port. This decrement constitutes additional driving pressure to compensate for the reduction in the net elevation pressure associated with the shortening both the hot and the cold hydrostatic columns operative around the passive cooling circuit.

Accident Conditions

As already implied, a principal objective of the AMPS cooling system is to achieve a passive cooling system capability which is constantly available for deployment by process-inherent means and capable of satisfactorily mitigating incipient reactor accidents. To this end, the hydrodynamic ports are designed to present no physical barrier against the flow of coolant between the primary cooling circuit and the reserve coolant tank. The basic design principle for avoiding the potentially deleterious effects of such an open arrangement on normal reactor operation has been outlined, and further refinements of their functions in this regard will be described in a later section.

Loss of Primary Circuit Flow. In the event that pumped flow through the primary circuit of the reactor cooling system shown in Figure 1 suddenly ceases during the course of normal reactor operation, the resulting drop in axial coolant flow through the hydrodynamic ports leads to a loss of the dynamic pressure change that is maintained during normal operation. The net accumulative pressure around the passive cooling circuit becomes non-zero and dominated by elevation pressures differences which are enhanced momentarily by the rising temperature of coolant within the core. In these circumstances, the reactor core becomes a component of the now operative passive cooling circuit and, following some reactivity-related transient oscillations, caused in part by the sudden entrainment through the inlet port of the relatively cool reserve coolant within the passive circuit, a relatively steady flow supported entirely by natural convection becomes established.

Loss of Heat Sink and Loss of Regulation. In the event of loss of capacity of the heat sink to the primary circuit, the resultant return of the circulating primary coolant to the core at temperatures significantly elevated above normal raises the average temperature of the effective thermosyphon hot leg (now of length L) to above-normal values, and forces the passive circuit to operate in parallel with the pumped circuit. This behaviour ameliorates the core coolant condition due to the increased core flow and, more importantly, to the combining of cool reserve coolant with primary circuit flow at the inlet port. A similar chain of events results, in the near term, following a loss of regulation event in which, for a time, the reactor undergoes a significant inadvertent increase in reactor power. The response of the system to still further cases of mismatch between the heat source and heat sink give rise to similarly mitigating sequences.

Loss of Coolant. The arrangement of Figure 1, in which the primary circuit and the reserve coolant tank are operated at vastly different temperatures but are intimately connected in regard to pressure, enhances the safety performance of the reactor in the event of
a primary circuit pipe break. This follows since only the relatively small fraction of the total primary inventory which lies within the primary circuit itself is at a temperature exceeding the saturation temperature at one atmosphere and is therefore subject to rapid vaporization and accompanying energy release at the instant of the break. The bulk of reserve coolant inventory remains as cool liquid immediately following the break and serves to continue the core cooling function after entering the core region through the ports and eventually establishing a steady passive cooling flow.

In the special case in which a reactor is non-pressurized relative to atmospheric pressure, and the maximum coolant temperature must be regulated not to exceed the saturation temperature at this pressure, a pipe break is followed by a very orderly progression of events due to the absence of vaporization of any coolant liquid, and culminates in the steady operation of the passive cooling circuit in the dissipation of residual reactor power production.

Post Shutdown Cooling. Post shutdown decay heat removal is provided by assuring in the design that through-core passive cooling flow is sufficient to maintain the fuel temperature at values below the safe design value, even when the reserve coolant temperature rises to the level necessary to support operation of the passive reserve tank cooler. Beyond the described accident scenarios which draw the passive cooling system into operation, the AMPS passive cooling system, as described, responds with adequate safety by process-inherent means to the accident situation initiated by the loss of all electrical power to plant systems, i.e., by a plant black out.

Reactivity Reduction. In the sampling of accident scenarios discussed above, in illustration of the response of reactor cooling system to accident conditions, no particular reference was made to either reactivity reductions or reactor shutdown action instigated early in the event scenarios. It is clear in all examples given, however, that the assurance of early reactivity reduction is essential in arriving at a detailed thermal design which will maintain the safety of the reactor following all credible initiating events. Furthermore, the provision of an early shutdown will clearly extend the period of time that may be allowed to pass before human intervention is required to assure long term safety following an accident. Such reactivity-limiting mechanisms are not prescribed as a part of the AMPS reactor cooling system, but it is clear that if the reactivity reduction or shutdown action is inherent in the processes which are intimately related to the accident condition, then reactivity-limiting mechanisms can be seen to be consistent with inherent safety aspects of the AMPS cooling system.

Therefore, two types of reactivity-limiting mechanism, both of which are generally understood, are mentioned here as alternatives which might be incorporated in a reactor plant based on the AMPS cooling system. The first is based on negative reactivity power coefficients associated with either the expansion of the coolant or the effect of fuel heating on either resonance absorption or neutron thermalization. Such mechanism may be adopted in the detailed design of a reactor such as the one in Figure 1, through the selection of fuel type. The AMPS reactors advanced to date have followed this approach through the adoption of U-238,4 in the core design to yield a large prompt negative reactivity response to increases in fuel temperature reflecting degradation of the core cooling requirement.

The second mechanism is less direct and is initiated by the onset of passive cooling which, in turn, is triggered in response to the initiation and development of the accident scenario. This reactivity reduction mechanism may be implemented in the detailed design of a reactor by arranging for the permanent addition of heavily concentrated neutron absorber in solution with the reserve coolant and placing neutron absorber only in mild solution with the primary coolant, in the manner characteristic of the SECURE-PIUS [12] reactor. The reactor, in this case, is regulated by actively varying the concentration of absorber in the primary circuit. Under normal operating conditions the solutions in the two levels of concentration in the reactor are kept from mixing with one another, as a byproduct of the arrangement described earlier to keep the same two categories of coolant from mixing, for reasons of both thermal efficiency and prudent radiological management. The additional requirement of maintaining the two concentrations of solution in fairly strict isolation from each other may in turn require an elaboration of operating protocol and the adoption of the somewhat more refined variants of the hydrodynamic ports which will be discussed later with reference to the Figures 2 and 3.

Operating Protocol

Startup. In terms of strictly thermal hydraulic considerations, the reactor cooling system of Figure 1 does not place any formidable restrictions on the operating protocol of the reactor plant. In a typical start up scenario, the primary circuit may first be turned on, in which case there will be a significant bypass exchange flow in which some circulating primary coolant enters the reserve tank through the inlet port, while an equivalent flow of reserve coolant passes into the primary circuit through the outlet port. As the reactor power is increased after being brought to criticality, the primary circuit warms more quickly than does the reserve coolant and, by the time the primary circuit reaches the normal operating temperature, it will have ceased altogether to exchange with the reserve coolant. A less likely alternative start up procedure is to first bring the reactor to an operating power level while relying on the convective flow of the passive system for core cooling. As the pumped cooling circuit is brought on line, the ingress flow of the passive cooling is gradually reduced to zero and the normal operating conditions are reached.

Continuity of the Exchange Flow Function. During the course of extended normal operations of the reactor depicted in Figure 1, a finite amount of exchange flow may occur inadvertently due to fluctuations in the operating parameters which lead to momentary departures from the zero net accumulative pressure requirement. It is an advantage of the AMPS cooling system that there is a continuity in the behaviour of the exchange flow deviation from the null point, as the values of the operating parameters vary about the precise combinations of values that provide null exchange. Such advantage is manifest in the
amenable to control of exchange flows to arbitrarily small values, whether by active control of the operating parameters or by the arrangements for passive compensation, to be described, particularly when such flow components arise in superposition due to different sources of perturbation.

**HYDRODYNAMIC PORTS (HDP's)**

**Basic Hydrodynamic Port Structure**

Beyond the imposition of the zero net accumulative pressure criterion on the primary circuit design, through the proper selection of the port axial flow areas as described previously, additional factors which limit the magnitude of exchange flow between the primary circuit and the reserve tank during normal operations are vested in the design of the hydrodynamic ports themselves. A simplified representation of a most basic version of the hydrodynamic port, suitable without further refinement in many applications, can be seen in Figure 1. This port constitutes an approximation of a continuous pipe made up of two flow-area defining ducts, joined axially by a branching section opening composed of a series of several slot-defining plates alternating with spacers which give rise to the circumferential exchange flow slots.

**HDP Function**

During normal operation, when the requirement of zero net accumulative pressure is nominally satisfied, essentially all flow through the port is axial and is substantial in magnitude due to primary pump operation. Under these conditions, branch flow tends to be discouraged on the basis that the particles of coolant involved in any such branch flow would experience high momentum changes, whether combining with or dividing from the main axial flow by passage through the slots.

The purpose of the slot and plate structure of the branch openings is to require the largest possible momentum change for the "typical" particle of coolant involved in branch flow. Thus, the port structure described is seen to be supportive of first two of the previously mentioned important properties of the hydrodynamic ports, namely, (i) the support of primary circuit flow continuity in normal reactor operating conditions, and (ii) the offering large resistances to both combining and dividing branch flows when the operating principal parameters inadvertently deviate somewhat from nominal values.

**Shutdown Cooling.** The third of the previously mentioned important properties of the hydrodynamic ports, namely, the offering of relatively small resistances to large branch flows of either kind in severely abnormal or accident conditions, is characteristic of the hydrodynamic port described, for the following reason. In abnormal circumstances, such as during the impairment of the primary pump, safe mass flow requirements through the core are considerably reduced due to (i) the inlet temperature being now determined by the reserve coolant temperature which is maintained at values greatly depressed relative to operating core inlet temperature, and (ii) the expected reduction in reactor power due to a suitable provision for reactor shutdown in accident conditions. The relatively small mass flow requirement and the resulting acceptability of relatively low passive cooling flows in abnormal conditions, makes tolerable the residual resistance to branch flows negotiating the slot-defining plates of the ports.

**HDP Variations.** The detailed specification of the hydrodynamic ports based on the principles of operation just outlined may be carried out by making suitable selections of the variables such as the axial flow area, the number and thickness of the slots, and the sizes of the plates and spacers may be selected to suit the capacity and the performance specifications for a reactor following the form of the reactor of Figure 1 and designed for a particular application. It is an advantage of the port concept and the manner of its implementation that the ports can be relatively simple of structure and, therefore, affords great flexibility in adaptation to a variety of reactor configurations. Nevertheless, the important properties of the ports alluded to earlier can be enhanced by implementing any one number conceivable refinements, two of which are described here in some detail.

**HDPs With By-Pass Modification.** A specific modification of the basic hydrodynamic port depicted in Figure 1 is now described briefly. The objective of this modification is to enhance, relative to the simple port depicted in Figure 1, the previously-mentioned second important property, namely, the offering of large resistances to both combining and dividing branch flows when the normal operating parameters inadvertently deviate somewhat from nominal values, while leaving essentially unaltered the branch flow area available to the shutdown passive cooling mode.

The modified hydrodynamic port may be described as the basic structure of the hydrodynamic port depicted in Figure 1, modified so that an array of bypass passages connect the area defining ducts independently of the exchange flow slots. The details of the bypass structure and the theory of their operation will be presented elsewhere. In the meanwhile, a brief theory of their operation is given in a companion paper [5] in this volume, reporting on the experimental studies of the hydrodynamic properties and the systemic behaviour of both the simple and the modified versions of the HDPs.

Briefly, the role of the bypass passages is to provide the opportunity for either the diversion or the recirculation of a portion of the axial flow through the port, according to whether there is a combining or a dividing branch flow as the result of the positive or negative axial pressure drops induced by the branching flows. As it turns out, the resulting reduction or increase in axial flow produces a trimming of the dynamical pressure within the port which is always in the sense that tends to cancel the initiating tendency for branching. When there is no branching, there is no exchange flow, and the modified port behaves in a manner identical to the simple port, i.e., it offers minimal resistance to the axial flow.

It is an important feature of the modified hydrodynamic port that the effects at work in opposing the departures from the zero net accumulative pressure requirement (giving minimal exchange flow between the primary circuit and the reserve coolant tank) act
promptly, i.e., in the order of the through-port transit time. This immediacy of response is of particular importance in counteracting the influences of periodic inertial forces on mobile reactors, in which the effective gravitational constant which determines the elevational pressures operating within the system may be subject to severe time variation.

Adequacy of Ports Without Shrouds. The discussion to this point has focused on two principal features of the invention which provide for the restraint of exchange flow between the primary circuit and the reserve tank in the absence of a physical barrier separating the two. These features may be embodied in a variety of reactors, of which one example has been described with reference to Figure 1. For many applications of such reactors, these two features alone may be sufficient. It is recognized, however, that without further measures to limit exchange flow, such flow may persist at levels unacceptable in other applications.

Anti-Convective Shroud Function

Residual Exchange Flows. The principal reasons for the persistence of exchange flows, in spite of the two features mentioned, are two-fold. The first is port related and results from the fact that a finite amount of local exchange flow will always be present in the ports as defined. The local exchange flow is the result of small internal pressure differentials caused by both the dynamic and the elevational pressure effects. Such differentials may be experienced in progressing longitudinally from one slot to the next, or when in sampling pressures while progressing vertically across the port when axis is oriented more or less in line with the horizontal. These effects give rise to local circulation of coolant into and out of the slots, even when the net branch flow in the port may have been cancelled due to the earlier-described design measures. The effects are aggravated by the inevitable hydrodynamic roughness near the slots and plates of the port interior, and the severe temperature gradient in the vicinity of the inescapable thermal interface between the circulating coolant and the reserve coolant.

The second reason for persistent exchange flow is system related. Even when the earlier-described design measures succeed in minimizing the pressure differentials of the system that drive exchange flows, in the first instance, or counteracting such pressure differentials as they may arise, in the second, such measures cannot in principle eliminate the exchange flows absolutely. Therefore, in reactor applications which require that exchange flows be restricted to very small values, or indeed eliminated, a supplementary mechanism to complete the suppression of exchange flows within the reactor system is necessary. An active mechanism that regulates one or more of the main system parameters to eliminate instrumentally detected or anticipated residual exchange flow within a cooling system, such as that described to this point in reference to Figure 1, may be devised. However, for reasons of reliability, universal applicability, and harmony with the basic objectives of passive operational control as well as passive safety, a passive mechanism is preferred to an active one in the performance of this function. In addition, the attribute of maintaining the ideal operating conditions by passive means have proven particularly valuable in addressing the design of plants subjected to accelerated or rotational motion, as will be considered in a later section of this paper.

Remedying Residual Exchange Flows. The phenomena of local exchange flow associated with each hydrodynamic port of the system, and the exchange flows residual in the reactor cooling system as a whole, are addressed through the introduction of a single attachment applied to each hydrodynamic port of the system. The attachment, examples of which are shown in the AMPS cooling system variants of Figures 2 and 3, is referred to herein as the anti-convective shroud (ACS). The anti-convective shroud is a passive device with the capacity to suppress the local exchange flows in the port to which it is attached, and to accumulate incipient residual exchange flow within the system flow in such a manner that the pressure differentials that drive the residual flows can be completely compensated.

Anti-Convective Shroud Structure. The anti-convective shroud is essentially a hood which opens downward and encloses the entire structure of the hydrodynamic port to which it is attached. Under normal operating conditions, the shroud provides a largely isothermal, quasi-static zone of coolant generally enveloping the port. This zone of coolant continually adopts the approximate temperature of the coolant flowing axially within the port, by virtue of the stratification grid made up of vanes placed vertically in the downward opening and the thermally insulating material placed on the remaining surfaces of the shroud. Since the temperature of the circulating primary coolant, and hence the temperature of the quasi-static zone, is considerably hotter than that of the reserve coolant, a large thermal gradient must exist in the vicinity of the downward opening. The purpose of the stratification grid is to inhibit the formation of local convective circuits in the vicinity of the opening, thereby facilitating the orderly formation and maintenance of thermally stratified layers of coolant ranging from the nearly primary circuit temperatures at the top of the grid, to typically reserve coolant temperatures at the bottom of the grid.

Containment of Local Exchange Flows. It was implied in the earlier discussion that, in the absence of special measures such as the anti-convective shrouds, local exchange flow through the slots of a given hydrodynamic port would exist as the result of small internal pressure differentials caused by both the dynamic and the elevation pressure effects. In the presence of the quasi-static zone of coolant around the port, however, significant elevation pressure differentials cannot exist, since the circulating primary coolant inside the port and the coolant of the quasi-static zone outside and along the port are now essentially of the same temperature. Moreover, while the dynamical pressure differences occurring within the port may result in some local exchange flow through the slots of the port, such exchange generally involves coolant of the single temperature and is contained within the shroud. Thus, the anti-convective shroud can be viewed in part as a passive device with the capacity to suppress the local exchange flows caused by dynamic and elevation pressure differentials in the port to which it is attached.
Suppression of Systemic Exchange Flows. It was also implied in the earlier discussion that, in the absence of special measures such as active primary parameter controls, or the passive compensating effects of the anti-convective shrouds, some residual exchange flows would persist in the reactor cooling system as a whole. Key points of behaviour of the anti-convective shroud which enable it to play a key role in limiting the extent of such residual exchange flow are described in the paragraphs immediately following.

If one considers the vertical temperature profile of the coolant located in the stratification grid of the anti-convective shroud exemplified in either Figure 2 or Figure 3, a transition in coolant temperature will be observed in passing from the top of the grid where the temperature is that of the local primary coolant, to the bottom of the grid where the temperature is nearly that of the reserve coolant. In a somewhat idealized example in which a reactor cooling system incorporating such a port is operating with absolutely zero net exchange flow passing through the port, the temperature profile in the grid may be characterized by the location of the median temperature plane located, say, midway between the upper and lower extremes of the grid.

If, however, the system should experience changes in operating parameters such that a small branching flow of primary coolant enters the anti-convective shroud, then a net downward displacement of the median temperature plane in effect occurs. This shift in the elevation of the median temperature plane amounts to a lengthening of the column of hot coolant extending below the port. As this column constitutes a component of the passive cooling circuit, which was defined earlier in the discussion on the criterion for zero exchange flow, the lengthening of this column contributes to the change in the net elevation pressure evaluated around the passive cooling circuit. Thus, the branch flow initiated by a disturbance in the operating parameters contributes to the creation of a component of driving pressure in the passive cooling circuit.

The remarkable feature of the anti-convective shroud in the configuration indicated is that, when such a shroud is added to each of hydrodynamic ports used in the cooling system arrangement exemplified in Figure 1, the induced change in the net elevation pressure induced in the manner described above is always in the sense which tends to cancel the initiated branch flow, regardless of its cause. As can be determined by a moment's reflection, this observation applies regardless of whether it is the inlet port or the outlet port that is involved, or whether the initiating disturbance gives rise to an inflow from the primary circuit into the shroud, as in the above example, or to an outflow from within the shroud into the primary circuit, in which case the median temperature plane shifts up rather than down in the effecting the correct compensation.

In summary, moderate disturbances in the primary system operating parameters cause the anti-convective shrouds to accumulate residual exchange flows within their quasi-stagnant zones in such a way that the resulting elevation pressure differentials cancel the original tendency for the exchange flow. Thus, for a wide range of normal operating scenarios, residual exchange flows may be virtually eliminated. For more extreme disturbances, the induced exchange flows may persist, but at flow rates reduced by the presence of the anti-convective shrouds to values that may be acceptable in many reactor applications.

CONFIGURATIONAL VARIANTS

The Range of Variants

To demonstrate the flexibility with which the AMPS reactor cooling system may be implemented, two alternatives, advanced design variations based on the basic version of Figure 1, are presented. These are represented in Figures 2 and 3, and are intended to cover a range of design environments, including, for example, power reactors in which there may be restrictions on exchange flows and space allowance, or research reactors in which non-pressurization and easy core access may be essential. The case of how the AMPS cooling system may be configured to deal with the mobile marine environment is also examined.

The Second Design Variation

(Restricted Space and Core Access)

The second design variation of the AMPS cooling system is shown in the reactor plant depicted in Figure 2. The cooling system of this plant has all of the essential attributes, and performs all of the basic functions, ascribed to the reactor plant of Figure 1. In addition, the reactor represented in Figure 2 includes two anti-convective shrouds, each attached to an inlet or outlet hydrodynamic port. These shrouds, and their associated stratification grids, enhance the basic performance of hydrodynamic ports in their prescribed roles in the manner described in some detail in the previous sections. Therefore, during normal operations, the reactor plant depicted in Figure 2 may be expected, even in the presence of significant operating disturbances and in the absence of actively compensating control equipment, to exhibit very little

Figure 2: Schematic representation of a reactor incorporating a second version of the AMPS cooling system addressing height, vertical access, and tolerance to operational disturbances.
exchange flow between the primary circuit and the reserve coolant tank.

Building upon the basic functions of the anti-convective shroud and the stratification grid, as described previously, the compensating actions of these attachments on a reactor system as a whole, in response to inadvertent variations in the normal operating parameters, may now be readily understood with reference to Figure 2.

Self-Regulation of Normal Operation. Suppose the reactor plant of Figure 2 is operating initially under a system of nominal operating parameters which satisfy the zero net accumulative pressure requirement for zero exchange flow. It may be recalled that the criterion for meeting this requirement is that the change in the dynamic pressure associated with a constant mass flow passing from the inlet port to the outlet port must equal the deficit by which the net elevation pressure, assessed accumulatively around the passive cooling circuit, fails to match the resistive pressure losses associated with the primary mass flow through the core. The important point to be noted here is that, in assessing the net elevation pressure in terms of the essentially the weight difference between a predominantly hot and a predominantly cold column, the height of both such columns is now effectively the vertical distance \( h \) between the median temperature planes falling at approximately the mid-elevation points in the stratification grids of the two ports. \( h \) is now the operative hydrostatic height in satisfying the zero accumulative pressure requirement, as opposed to the height \( L \) which, in the absence of shrouds and grids as for Figure 1, was taken as the operative height. A further important point is that the magnitude of \( h \) can vary in response to the receipt of small exchange flows within the shrouds, and such variation is the basis of the mechanism for compensating inadvertent departures of the cooling system from the zero net accumulative pressure requirement, which give rise to these flows initially.

Now suppose that a tendency develops within the operating system to produce ingress exchange flow, in which case an incipient combining branch flow occurs in the lower port and a corresponding dividing branch flow occurs in the upper port. In the reactor depicted in Figure 2, such a tendency may result, for example, from an inadvertent reduction in pump speed, an increase in circulating primary coolant temperature, or an effective (inertial) increase in the gravitational constant due to upward acceleration of the reactor. As the slight ingress exchange flow progresses, the median temperature plane in the grid of the inlet port rises and the median temperature plane in the grid of the outlet port falls, in the manner already described. The consequent contraction in \( h \) corresponding to the reduction in the elevation head that is necessary to cancel the cause of the initiating tendency for ingress exchange flow.

Similarly, a developing tendency within the operating system to produce a bypass exchange flow results in a protraction of \( h \) corresponding to the increase in the elevation head that is necessary to cancel the initiating cause of bypass exchange flow.

The foregoing operational scenario shows, by example, how the cooling system of the reactor depicted in Figure 2 provides the process-inherent means for regulating the cooling system during normal reactor operations in such a way that the potentially undesirable side-effects of the passive shutdown system on normal operations are automatically and continuously curtailed. This situation contrasts with that for other reactors having passive shutdown cooling systems which, as in the present invention, are deployed by process-inherent means. Such other systems [12,13] depend on sophisticated sensors and active control systems to avoid undesirable side-effects.

Space Restrictions. The representation of the reactor of Figure 2, also exemplifies the adaptation of the invention to applications which place severe restrictions on the size of the plant, while requiring passive cooling to become available as needed through inherent processes. In addressing such requirements, the vertical inter-port distance \( h \) (shown in Figure 2) has been made shorter than that which would be required to support the normal through-core flow rate by natural convection alone. Consequently, the zero net accumulative pressure requirement, as stated with reference to the passive cooling circuit, is met by making the axial flow area of the outlet port somewhat larger than the area of the inlet port. Moreover, in the instance of lessening of the requirements on the size of the plant, the inlet port and the inlet duct are oriented vertically with its anti-convective shroud and stratification grid arranged symmetrically about the port axis. It should be noted that the individual components depicted in Figure 2 are not drawn to scale and no relative size or configurational relationships among the various components should be inferred literally from the Figure.

Passive Cooling. The passive cooling circuit of the reactor of Figure 2 is readily seen to include the anti-convective shrouds and the stratification grids among its components. Accordingly, the thermosyphon hot leg, operative during convective core coolant exchange with the reserve coolant, consists of the upper plenum and the outlet duct leading to the outlet port, plus the upper shroud and grid. The effective vertical height of the thermosyphon hot leg is therefore approximately equal to \( L \), less the vertical distance from the middle of the upper port down to the median temperature plane of the upper grid. Correspondingly, the effective vertical height of the cold leg is approximately equal to \( L \), less the sum of \( h \) and the vertical distance from the middle of the lower port down to the median temperature plane of the lower grid. The layers of insulation shown attached to the shrouds, as well as to the other components of the primary circuit, are consistent with the roles of the shrouds and grids as components of the passive cooling circuit.

Core Accessibility. The reactor plant depicted in Figure 2 also exemplifies applications in which vertical access to the reactor core must remain available for purposes of shielding placement, control and shut off mechanism deployment, experimentation (as in the case of a research reactor), and refuelling and general maintenance. Access to the top of the reactor core is retained in the design of the subject reactor primarily by offsetting the outlet duct immediately as leads away from the outlet plenum. This arrangement
The reactor vessel of the plant in Figure 2 is shown in an abbreviated form. In this example, the primary coolant baffle is a demountable, thermally insulated barrier which is essential during normal operation to prevent the primary coolant from mixing with the reserve coolant, but is obviously not exposed to very large pressure differentials. In some applications, combining the baffle with reactor core shielding may be advantageous.

Access for servicing such facilities, as well as for performing refuelling operations, is obtained through the hatch in the case of a closed or pressurized reactor, or simply through the open surface of the coolant in a pool-type reactor. For either type of reactor, it may be advantageous in the design to abandon this component and to extend the reactor vessel to a much higher level than shown, or even to the location where it joins with the hatch (in a closed reactor), or breaks the coolant surface (in a pool-type reactor).

The Third Design Variation
(Accelerations and Rotational Motions)

A third variation of the AMPS cooling system is shown in the reactor plant of Figure 3. The cooling system of this plant has all of the essential attributes, and performs all of the basic functions, which were generally ascribed to the reactor plant of Figure 2. In addition, this version includes a multiplicity of inlet and outlet hydrodynamic ports, each coupled hydraulically to a corresponding inlet or outlet plenum, by means of an inlet or outlet duct. Leading in directions generally away from the reactor core, the inlet and outlet ports couple also to their respective inlet and outlet manifolds. The manifolds, in turn, couple to the inlet and outlet conduits which connect with the remaining components of the primary circuit lying outside the reserve coolant tank. Each of the hydrodynamic ports shown in Figure 3 is fitted with an anti-convective shroud and a stratification grid.

Role of Multiple Ports. The plant shown in Figure 3 typifies applications in which the reactor cooling system, as well as performing in the manner described with reference to the plants of Figures 1 and 2, also meets the requirements of (i) passive shutdown cooling being always available, regardless of the orientation of the reactor with respect to gravity, and (ii) the normal operation of the reactor being tolerant of various types of dynamic motion and net displacement, both rotational and translational, imposed within specified limits on the reactor plant as a whole. The basic operational details whereby these requirements may be met are now given. In proceeding, an understanding of the operation of the first and second versions is presumed.

Normal Operation. The primary coolant of the reactor in Figure 3 is circulated during normal operation according the pattern depicted by the arrows. The flow to the core is delivered in more or less equal shares by the several inlet ducts, and each share is received by a duct as an axial flow transmitted by the corresponding inlet port. Similarly, the flow out of the core is received in more or less equal shares by the several outlet ducts, and each share is transmitted by a duct to become an axial flow in the corresponding outlet port.

The definition of normal operating conditions, in the application environments for which the reactor variation shown in Figure 3 is suitable, includes an alignment of the central axis of the reactor to coincide more or less with the vertical, and an absence of motion of the platform on which the reactor is mounted. These stipulations for normal operations are in addition to those relating to the key system design parameters.

The operation of the reactor of Figure 3, under normal conditions as defined above, may be described in terms essentially the same as those used previously in describing the reactors of Figures 1 and 2. For example, any tendency towards exchange flow between the circulating primary coolant and the reserve coolant is minimized for all three reactors because the axial flow areas of the hydrodynamic ports are chosen so that the zero net accumulative pressure requirement is fulfilled. This requirement may be restated more aptly, however, for the specific case of the multiple port configuration of Figure 3: At normal operating conditions, the net pressure from all effects accumulative around each of the many identifiable passive cooling circuits, and tending to support net flow around any such circuit, must be substantially zero. A passive circuit, in this context, can be defined as any closed path in the system which includes the reserve coolant tank, and the two ducts connecting with any arbitrarily chosen pair of hydrodynamic ports, including pairs of inlet ports, pairs of outlet ports, and pairs formed of any combination of one of each.

As in the case of the second version shown in Figure 2, the sizing of the port areas for the third version shown in Figure 3 is based on Lq, the operative...
hydrostatic height now defined as the vertical distance between the inlet and outlet groups of median temperature planes, as well as the nominal values of the key operating parameters.

Either of the two designs of hydrodynamic port previously described in reference to Figure 1, or indeed a further design, may be chosen for application in the reactor of Figure 3, depending on the degree of resistance needed against the tendency toward exchange flow. In the meanwhile, the anti-convective shrouds eliminate or isolate the effects local exchange flows in the multiplicity of ports, and the self-adjusting capability of the median temperature planes, occurring within the various stratification grids, serve to automatically eliminate, or limit, incipient exchange flows. Additional sources of tendency toward exchange flow arise in the configuration of the passive cooling system of Figure 3, however. These may be due to the presence of minor non-symmetries in the manufacture of the various hydrodynamic components relating to the multiplicity of ports and associated flow paths. The disposition of this source of exchange flow tendency will be addressed later, along with the effects of various types of dynamic motion and net displacement imposed on the reactor plant by its operating environment.

Passive Cooling Operation. The multiplicity of passive cooling circuits addressed in connection with the maintenance of ideal operating conditions under normal (upright) conditions, are also the key to the main feature of the third version reactor, namely, the continuous availability of the passive cooling function, regardless of the orientation which the reactor may assume with respect to gravity. The operation of the provisions for passive cooling may be readily understood with reference to the plant depicted in Figure 3.

Consider the operation of the passive cooling system, following the failure of the circulating pump and the general stoppage of flow in the part of the primary circuit external to the reserve coolant tank, while the reactor, for the time being, remains upright. Under these conditions, the mass flow through the core reduces to the level which can be supported by natural convection in the passive cooling circuits. In this, more or less equal flows from the reserve coolant tank enter the inlet ports through the shrouds of the inlet ports, pass through the core, emerging into the reserve tank as equal flows through the shrouds of the outlet ports. Apart from the multiplicity of flow paths, the general behavior of the system under these conditions is essentially the same as for the reactors of only two ports as in the second version of Figure 2.

Consider now the case in which the reactor's physical orientation departs to an arbitrary degree from the upright position. It may be inferred from Figure 3 that, regardless of any such orientation, a variety of viable passive cooling circuits is always available for natural convection. Such circuits were defined, in the discussion on the requirements for zero exchange flow under normal operating conditions, as including the reserve tank, and two of the ducts connecting with any pair out of the multiplicity of hydrodynamic ports.

Plant Motion Effects. As just described with reference to Figure 3, the main feature of the third version reactor plant is the availability of effective passive cooling at all times and in all circumstances, including all physical orientations of the reactor plant. Such facility for passive cooling would be of limited value, however, if the presence of such facility were to degrade significantly the efficiency of normal operations, or if the measures required to avoid such degradation were to compromise the operability of the plant. Moreover, the operating environment that requires a passive cooling capability at all physical orientations, is likely to be a dynamic environment (as on board ship) of which the effects on normal operation have to be accommodated, in addition to the effects of inadvertent variations in the normal operating parameters as already discussed. The manner in which the normal operation of the reactor of Figure 3 is tolerant of the various types of dynamic motions and net displacements, is now described.

Rotational Displacement. Consider first an incremental rotational displacement of the reactor in Figure 3, operating initially with an upright orientation under ideal normal conditions of no exchange flow. Assuming that the rotation is clockwise about an axis corresponding to a normal to the plane of the paper of Figure 3, the distortion of the original flow symmetries due to the changes in elevation heads results in a dividing branch flow at the left-most outlet port, and a combining branch flow at the right-most inlet port. The resulting movements of the median temperature planes in the grids of the two ports, due to the accumulation within the shrouds of minor amounts of exchange flow, causes the value of \( L_q \) as it pertains to the two ports in question, to tend to shorten toward its original value, even though an increased vertical separation has occurred between these two ports. Similarly, the occurrence of combining branch flow in the right-most outlet port, and dividing branch flow in the left-most inlet port, tends to lengthen the value of \( L_q \), as it pertains to these two ports, to its original value, even though the two ports now physically have less vertical separation. Thus it may be seen for this simple example of rotational displacement that, by the interaction of incipient residual exchange flows with the anti-convective shrouds and stratification grids, the parameter \( L_q \) self-adjusts within the incrementally rotated system to satisfy the hydraulic requirements for zero exchange flow with respect to all four ports simultaneously. On extending this line of reasoning to more general circumstances, it may be concluded that the parameter \( L_q \) will automatically adjust to the value satisfying the zero flow requirement with respect to all hydrodynamic ports in the system simultaneously, even when the plant is subjected to arbitrarily complex rotational displacements.

Limits to Self-Compensation. At certain limits of reactor inclination, it is apparent within the above line of reasoning that the range of useful self-adjustment of \( L_q \) in the inhibition of exchange flow becomes exhausted. Such a limit occurs when the plant inclination becomes so great that the stratification grids of two diametrically opposite ports can no longer both intersect a single horizontal plane. It is apparent, also, that the limit of inclination in radians is approximately equal to the ratio of the (vertical) grid length to the diametrical separation of the said ports. Therefore, for a given basic reactor configuration, the range of inclination within which
the reactor plant can remain relatively free of exchange flow is proportional to the length of the stratification grid in the anti-convective shroud of each hydrodynamic port. While lengthening the grids in the design of a reactor for a given mobile application would increase the reactor's immunity to exchange flow, the indiscriminate use of such measures could compromise the effectiveness of the reactor's passive cooling system. An appropriate balance between immunity to exchange flow and the effectiveness of the passive cooling may be achieved with the design of a reactor for a particular application.

Self-Compensation For Mechanical Aberration. It was stated earlier that any tendency towards exchange flow between the circulating primary coolant and the reserve coolant may be minimized by choosing the axial flow areas of the hydrodynamic ports so that the zero net accumulative pressure requirement is fulfilled. This statement implies that, particularly in the third version reactor, a certain quality of manufacture is achievable. The quality of manufacture would have to be such that the total mass flow would be shared identically by all ducts, and the specified axial flow areas of the hydrodynamic ports would precisely materialize in the manufacturing process, to the extent that the zero net accumulative pressure requirement would be truly satisfied simultaneously for all passive cooling paths to be found in the system. Since it may be impractical, however, to manufacture the hydraulic equipment to such precision, it may appear that some form of post-assembly adjustment might be in order, and mechanisms for such adjustment might be suggested. However, it should be pointed out that, through the line of reasoning used previously to show that the anti-convective shrouds and stratification grids automatically adjust system elevation pressures to inhibit exchange flow in the face of disturbances in both the primary parameters and changes in reactor orientation, it may be shown that tendencies toward exchange flow arising from the practical limitations in component manufacturing are likewise compensated, within reasonable limits, by the self-adjusting properties of the shroud-equipped hydrodynamic ports. It follows, as a corollary, that any minor in-service changes in the geometry of the hydraulic equipment, due to corrosion, deposition, or deformation associated with aging, are likewise compensated automatically.

Self-Compensation For Dynamic Motion. In consideration of the effects of dynamic motions on the integrity of normal reactor operation, two types of motion are important in the operating environment anticipated for the reactor of Figure 3. The types of motion are (i) rotational oscillations of limited amplitude about horizontal axes, and (ii) vertical translational oscillations of limited magnitude.

Rotational Oscillations. Consider first the rotational oscillations. Here we consider only rotations about horizontal axes located approximately at the level of the reactor core. Such rotational oscillations, typified as a rocking motion about an axis through the reactor core and normal to the paper in the representation in Figure 3, can be considered to potentially create exchange flow in two ways, i.e., through inertial (accelerative) effects and through displacement effects, which will be dealt with in turn.

Inertial Effects in Rotation. Since the rotations are defined to be about the core, the forces of centrifugal acceleration on the coolant in the various ducts will tend to cancel. However, the scooping effect, as the opening of an anti-convective shroud accelerates into the reserve coolant, tends to produce an exchange flow of the "inlet port to inlet port" variety, and similarly for the outlet ports. While such an inertial effect persists, an incipient exchange flow actually occurs and results in the displacements of the median temperature planes in directions that tend to cancel this inertial effect. Since the inertial effect reverses direction for the second half of the oscillation period, the displacement of the median temperature planes will be reversed. If this reversal takes place before the ranges of travel in the stratification grids become exhausted, as will be the case ideally, no net exchange flow will be experienced as a result of the scooping effect.

Rotational Displacements. The strictly displacement effects of rotational oscillations may be analyzed using a similar line of reasoning to the one used previously in addressing the effects of net (static) displacements. In the case of very slow oscillations, the very same criterion for inhibiting exchange flow applies as for static displacements, namely, the maximum rotational displacement from the upright orientation of the reactor in radians should not exceed the static limit, namely, the ratio of the length of the stratification to the diametrical separation of the outlet or the inlet ports. As the frequency of the oscillations speeds up for a specific reactor configuration, however, there will be an increase in the maximum amplitude of oscillation that can be allowed before a net exchange flow occurs. The permissible amplitude increases because the incipient exchange flow rate is limited by the resistance of the ports to branch flow, and the capacity of each anti-convective shroud reverses direction for the second half of the oscillation period corresponding to the ship being on the crest of the wave, there will be a tendency for bypass exchange flow in the reactor cooling system. On the other hand, for that part of the cycle corresponding to a depression, there will be a tendency for ingress exchange flow in the system. Whether these tendencies lead to actual periodic exchange flows between the primary circuit and the stratification depends on a number of factors in common with those that were considered in connection with the rotational oscillations, such as the resistance of the hydrodynamic ports to branch flow, and the capacity of each anti-convective shroud, which determines the fraction of the range of the stratification grid that is traversed by the
advancing median temperature plane before it recedes in the second half of each cycle. If the ports, shrouds and grids are specified so that, for the entire spectrum of anticipated oscillation frequencies, wave amplitudes do not exceed the values that correspond those which push the median temperature planes beyond the range provided by the grids in a single half-cycle, then such motions need not result in any actual exchange of primary coolant with the reserve tank.

CONCLUDING REMARKS

A detailed generic description has been given of the functioning of the AMPS reactor cooling system in a broad range of potential reactor design and operating environments. While the emphasis has been directed towards the special circumstances of the mobile marine environment, it is evident that many of the features of the AMPS system may be implemented to advantage in various land-based applications of both power and research reactors.

Several reactors based on the AMPS principles, and ranging in electrical power from 100 kWe (1.5 MWt) [1] to 1,700 kWe (10.8 MWt) [9], have been advanced in design and in various aspects experimental verification. The cooling system, as described in this paper, is being subjected to extensive thermal hydraulic testing [3], which has already confirmed much of the operation of the AMPS cooling system. Some of the earlier results have been reported [3,5]. Given the described advantages of the AMPS cooling system, the forthcoming results of further testing are expected to be of great interest to reactor thermal hydraulics in general.

REFERENCES


[10] See, for example, the proceedings of The First IAEA International Seminar on Small and Medium-Sized Nuclear Reactors, Lausanne, 1987, and published as Nuclear Engineering and Design, 102, 1988.


MAPLE-MNR: PRELIMINARY THERMALHYDRAULICS STUDIES

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ABSTRACT

McMaster University has outlined the incentives for replacing the present 30-year old McMaster Nuclear Reactor (MNR) core with a core of MAPLE design. Briefly, such a renewal of MNR would provide experimental facilities with significantly higher thermal neutron fluxes. This involves a relatively low cost, since the present MNR cooling and auxiliary reactor systems would be retained. In this paper, preliminary thermalhydraulics studies are presented.

INTRODUCTION

The McMaster Nuclear Reactor (MNR) is a 5 MW, MTR-type, open-pool research reactor. It consists of a rectangular matrix core which contains the fuel assemblies and special sites for the irradiation of samples. The core, its support structure, and the control and reactivity mechanisms are suspended in the pool from a moveable manbridge spanning the pool. Cooling water is drawn downward from the reactor pool through the core to the primary cooling system via an outlet in the pool floor.

It has been proposed to replace the MNR with a new, Canadian designed, 5 MW reactor core known by the acronym MAPLE (Multipurpose Applied Physics Lattice Experimental). This reactor is identified as MAPLE-MNR to distinguish it from the 10 MW MAPLE-X reactor being designed by Atomic Energy of Canada Ltd. (AECL) for installation at its Chalk River Nuclear Laboratories. The MAPLE-MNR reactor is closely patterned on the MAPLE-X reactor. It is intended to install the MAPLE-MNR and its control system into the existing biological shielding and containment building, using the existing cooling and heating systems, etc. This will significantly reduce the design and construction costs, and may keep licensing requirements to a minimum. In this work, some highlights of the preliminary thermalhydraulic studies will be presented based on the proposed MAPLE-MNR core. These are: (1) Numerical flow simulations of the chimney inside the existing MNR swimming pool by the MAPLE3D code [1]; (2) Numerical flow simulation of the primary cooling system using the SPORTS-M [2]; and (3) Identification of possible cooling modes under loss-of-forced flow conditions.

MNR PRIMARY COOLING SYSTEM

In the proposed implementation of a MAPLE reactor core into the MNR reactor system, all of the in-pool components would be replaced. The original primary and secondary cooling systems would be retained, with the exception of components which may need to be renewed (such as the main pump) or slightly modified.

Figure 1 shows a schematic of the primary cooling system which is used in the present MNR, and will be used in the proposed MAPLE-MNR. Elevations of the primary system components are not reflected in this figure - the bottom of the heat exchanger is actually at core level.

Demineralized light water coolant passes from the core to a holdup tank, where short-lived radionuclides, such as N-16 are formed by activation of the coolant, decay. After a holdup time of 5 to 10 minutes, the coolant is pumped through two heat exchangers and back to the core. The coolant passes through several monitors for operational as well as safety control, such as a flowmeter and temperature sensors.

In contrast to the MNR and MAPLE cooling systems, the MAPLE-MNR system is an open loop in which the chimney water drains directly into the holdup tank before being pumped back through the heat exchangers.

MAPLE MNR REACTOR

Figure 2 shows the schematic of the MAPLE-MNR reactor core and chimney, which are connected by the coolant inlet and outlet piping to the original MNR primary cooling system. The same pool outlet (directly below the core) and inlet (pool floor a few meters away) as used with the MNR would be retained. Most of the primary system piping is encased in concrete for radiation shielding and for the protection of pipe integrity.

In Figure 3, a side view of the pool inlet piping is shown. A characteristic feature of the MAPLE design is the 10% bypass flow that is required to prevent coolant that has passed through the core (and as a result contains radionuclides) from reaching the pool surface. The bypass flow is pulled down the chimney, and is routed with the core discharge into the primary cooling system piping via the chimney suction outlets.

![Figure 1 Schematic of primary cooling system of McMaster Nuclear Reactor.](image-url)
The bypass flow is drawn off the inlet flow through a vertical pipe, as indicated in Figure 3. There is also a safety check valve located along this pipe. The check valve is closed under normal operating conditions and opens when the primary system pressure drops below a given point according to the setting of counterweights. This is an important safety feature which will give pool water access to the core under reduced flow conditions, so that natural convection may take place.

Under reactor shutdown conditions, the level of radioactive decay heat that must be removed can be up to 6% of the maximum reactor power. Removal is either by normal forced circulation through the primary cooling system, or by natural circulation if the primary system is disabled.

A tank of heavy water ($\text{D}_2\text{O}$) surrounds the fuel to provide moderation and reflection of neutrons. When the reactor is operating, this $\text{D}_2\text{O}$ doughnut is especially rich in thermal neutrons. For that reason, the $\text{D}_2\text{O}$ doughnut is penetrated by several experimental beam tubes. The $\text{D}_2\text{O}$ tank will require an independent cooling system, since it absorbs about 0.25 MW at 5 MW operation from heating by neutrons and gamma-rays. The moderator system must be closely monitored for buildup of tritium.

The MAPLE core consists of 36-pin fuel assemblies for use in standard lattice sites, and 18-pin assemblies for use in reactivity control sites. The bundles proposed for the MAPLE-MNR are 60 cm long and contain 19.7% $\text{U}-235$ in a $\text{U}\text{SiAl}$ alloy dispersed in an aluminum matrix and sheathed with finned aluminum.

The reactivity control for the MAPLE reactor core consists of control rods and shut-off rods. These rods are held in position by hydraulic cylinders or electromagnets which release the rods in the event of a power failure. The rods are enveloped by shroud tubes that are permanently positioned within the chimney.

**THERMALHYDRAULICS OF THE PRIMARY COOLING SYSTEM**

The major concern was whether or not the present primary cooling water pump, heat exchangers and piping system could meet the requirements of the new reactor. As part of this study the existing primary heat transport system attached to a MAPLE was modeled with the SPORTS-M thermal hydraulic code [2,5].

SPORTS-M is a computer code that performs steady state and transient thermal hydraulic analysis of piping networks. The governing equations used in the code are the conservation equations of mass, momentum and energy together with the equation of state for a one-dimensional transient flow of a homogeneous two-phase mixture. A heat transfer package and radial heat conduction module are coupled to the hydraulic modules. The heat transfer package contains correlations that address all the heat transfer regimes of a boiling curve. The correlations are coupled with a radial heat conduction model for a single fuel pin that solves the transient heat conduction equation by a fully-implicit finite-difference scheme.

Table 1 lists the segment names used in the MAPLE-MNR model and the description of the portion of the primary cooling system that the segment models. Figure 4 shows the idealization of the MAPLE-MNR primary cooling system. A "*" in the segment name indicates a wild character that represents the flow path. The characters chosen to represent the three flow paths are: "D" for the reactivity control sites, "D" for the 36-pin driver fuel site and "A" for the zirconium irradiation module sites. Details of the hydraulic modeling can be found in Ref. [3].

Axial power distributions for the driver and reactivity control sites were obtained from Smith [5].
A summary of the calculated MAPLE-MNR steady state thermal-hydraulic parameters is given in Table 2. The numerical predictions indicated that the onset of nucleate boiling (ONB) for the hottest part in each fuelled segment is predicted to be within maximum allowable levels.
Figure 5 shows an estimated system resistance, head loss vs. flowrate, curve both before and after the additional pipework has been added. It can be seen that the new piping does not add significantly to the present system resistance. The original MNR pump head loss vs. flow curve has been superimposed in Figure 5 for illustration purposes.

Figure 5 indicates that the primary water pump, providing it is brought up to its full specification with the addition of a new impeller, would be more than capable of handling the flow requirements of the MAPLE reactor at 5 MW power. The fact that there is no increase in flow required also suggests that the existing holdup tank will be adequate to provide the delay times necessary and, as there is now, leave plenty of capacity in hand. Figure 6 shows the holdup tank water level plotted against system flow for a delay time of 300 s. The present MNR 2 MW operating flow, the predicted MAPLE-MNR 5 MW operating flow calculated by SPORTS-M, and the original MNR 5 MW flow are shown for comparison purposes. Since the tank is built between the inner and outer walls of the reactor building, modification to it would be an expensive proposition. However, as can be seen from Figure 6, this is unlikely because there is sufficient spare capacity to meet the minimum delay requirement of 300 s for flow rates up to nearly 450 l/s. This is approximately 400 percent greater than the flow predicted by SPORTS-M for the MAPLE-MNR operating at 5 MW.

In summary, the results indicate that the flow areas and fluid velocities in the system do not exceed those that we have now for 5 MW operation. The factor of safety in terms of the ONR is approximately 30 percent so that the probability of fuel dryout is low.

CHIMNEY DESIGN AND POOL DEPTH

As mentioned previously, one of the few major changes in the new MAPLE-MNR reactor compared to the existing MNR is the reversing of direction of the core cooling water flow.

Having made the decision for upward flow it is then necessary to prevent the short-lived radionuclides from rising to the pool surface where the high energy gamma radiation could become a hazard.

This has been accomplished using a chimney design, Figure 2, which draws off the upward cooling outflow directly as it leaves the core. A bypass flow (approximately 10 percent of total flow) enters the pool directly and is then drawn from the pool down the chimney where it recombines with the core flow near the suction outlets. The verification of this concept has been handled by the MAPL3D code [5]. This study was undertaken to confirm that a reduced chimney height, to 2600 mm from 3000 mm, was sufficient to confine the outflow from the core within the chimney. To demonstrate this, flow patterns in the chimney and the reactor pool were calculated under a variety of conditions. The effects of chimney height, pool diameter, suction outlet elevation, bypass flow ratio, boundary velocity magnitudes, and nonuniform velocity profiles at the core outlet were examined.

The major assumptions and idealizations made in the simulations were:

1. the turbulent viscosity was calculated at each node based on the local values of the velocity and density, \( \rho \), and the core-exit diameter as a length scale from:

\[
\mu_t = 0.01 \rho v_{exit} D_{exit}
\]  

This model was to account for mixing of the three interacting streams (core, suction and countercurrent flows) in the chimney and satisfied the scaling laws derived in [11]. As shown in reference [11] the model predicted the experimental results from the 1/5-scale MAPLE hydraulic facility at the CNS 10th ANNUAL CONFERENCE, 1989, 10-17
Figure 7  Predicted flow patterns in chimney and pool with (a) no bypass flow, (b) 10% bypass flow.

Figure 8  Predicted flow patterns inside chimney with 10% bypass flow with nonuniform core velocity and the flow velocities at the boundaries.

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AECL-RC Whiteshell Nuclear Research Establishment reasonably well.

(2) the chimney, the core and the pool in MAPLE-MNR were represented as coaxial cylinders in cylindrical coordinates such that the actual flow areas of each were maintained.

(3) the flow was at steady state and isothermal. The isothermal assumption was found to be reasonable for high-flow conditions, since the buoyant force was relatively small in comparison with other forces from the core, suction and bypass flows.

(4) components and piping obstructing the flow in the chimney and pool were not included in the model. Since the suction outlets are located above the top of fully withdrawn position of the control/shutdown shroud tubes in MAPLE-MNR, predicted flow patterns were considered representative above this position. This assumption was considered reasonable since the high-flow core-jet velocity entering the chimney decayed very little between the top of the reflector tank and the suction outlets.

MAPL3D was used to model the volume above the top of the reflector tank, since the flow patterns in the chimney and pool were the main items of interest. In view of the symmetry of the flow, it is sufficient to model only one quadrant of the system. Thus, two boundaries of the calculation domain consisted of symmetric planes. The inlet flow conditions were explicitly specified at the core exit and the annulus between the inside of the chimney and pool wall and the outside of the reflector tank wall such that global symmetric planes. The inlet flow velocity fields were set to zero for the simulations. The converged steady-state solution was independent of the specified initial conditions.

The initial flow velocity fields were set to zero for the simulations. The converged steady-state solution was independent of the specified initial conditions.

To examine the effect of bypass flow, a case was simulated under no bypass flow conditions. The predicted flow patterns in the chimney and pool under no bypass flow conditions are shown in Figure 7a. Figure 7b shows the plot of velocities which were scaled in relation to their magnitudes. A comparison with no-bypass flow-to-bypass-flow ratio of 10% reveals that the flow patterns near the suction outlets are similar between the two cases. However, without bypass flow, the fluid in the chimney "leaks out" through the central region of the chimney and returns down through the inner edges of the chimney to make up the mass flow balance. The leaking velocity was predicted to be small and was of the order of 10 to 10-2 m/s (compared to the core velocity of 2.5 m/s and the suction velocity of 5.2 m/s).

The effect of flow velocities at the boundaries with a nonuniform core-velocity profile is shown in Figure 8. To show the effect of velocity magnitudes at the nonuniform flow boundaries, the flow velocities at the boundaries were increased to three times. A bypass flow ratio of 10% was used. The velocity magnitudes were predicted to change, but the flow patterns were unchanged regardless of the magnitudes of velocities at the boundaries.

From the numerical results, the following conclusions were drawn:

(1) the 2.6-m chimney height is sufficient for full containment of the core jet within the chimney at a bypass flow ratio of 10%;

(2) when the core jet is contained in the chimney, the vortex stretching height is unaffected by chimney height, pool diameter or suction outlet elevation (only within limits);

(3) similar flow patterns in the chimney result regardless of the magnitudes of the velocities at the boundaries for both uniform and nonuniform velocity profile at the core exit;

(4) a nonuniform core-velocity profile results in a higher vortex stretching height than a uniform core-velocity profile for a bypass flow ratio of 10%;

(5) under conditions of no bypass flow, a small amount of the core jet leaks out of the chimney top but it returns back into the chimney top.

In summary, the results indicate that a 2600 mm chimney height is adequate for full containment of the core outflow. Additionally these results provide a high degree of confidence in the chimney concept since downward flow was shown to exist under a broad variety of conditions.

ASSOCIATED SAFETY CONSIDERATIONS

Some detailed analysis has already been carried out regarding safety features of the MAPLE reactor. However, the integration of a MAPLE core into an exiting reactor is unique, and warrants an independent study. In Ref [6] the feasibility of such an integration was reviewed from a safety standpoint. Reactivity and cooling related accident scenarios are considered, and recommendations are made for future safety studies of the proposed MAPLE-MNR reactor. These recommendations take into account recent studies that have been carried out elsewhere for MAPLE reactors, as well as the particular needs of the MNR host system.

Abnormal reactivity and cooling conditions include loss of control, loss of power and/or pumping, and loss of moderator and/or coolant. The primary concern in these scenarios is core overheating with the subsequent release of highly radioactive fission products. The MAPLE design has the advantage that the direction of flow is conducive toward natural circulation of coolant through the core. Furthermore, the core design ensures negative temperature and void reactivities.

The recommendations for the next stage of study include:

- the modelling of natural circulation of MNR pool water to ensure adequate passive cooling of core heat;
- the investigation of subchannel flow conditions such as the point of onset of significant void and the critical heat flux;
- the modelling of operational transients such as loss of power;
- the study of effects by the D2O moderator system on the core reactivity and cooling.

These and many other detailed studies would be carried out during the licensing and safety research study period of the proposed MAPLE-MNR.

CONCLUDING REMARKS

Preliminary results show that:

(1) the onset of nucleate boiling (ONB) for the hottest part in each fuelled segment was predicted to be within maximum allowable levels for the present range of the flow rate;

(2) the existing primary cooling water system is adequate to handle the MAPLE-MNR cooling requirements up to an operating power level of 5 MW;

(3) the preliminary design of the chimney on the MAPLE-MNR reactor has been shown to prevent short-lived radionuclides from rising to the top of the pool. As, well, it acts as the support for the reactivity control and shutdown mechanisms.
The pool natural circulation is expected to be a major cooling mode under loss-of-forced flow conditions in the primary cooling system.

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ABSTRACT

A design feature being considered for the MAPLE-X10 research reactor is a flap valve located in the pool on the primary coolant line to the inlet plenum. When forced circulation of the coolant is not possible under some accident conditions, or when the decay power is too small to maintain natural circulation through the primary cooling system, this valve would provide a path for the primary coolant to the pool. This would allow the core to be cooled by natural circulation of the coolant in the pool.

The CATHENA computer code was used to determine whether the flap valve would be needed to help cool the core during an upset condition in the MAPLE-X10 reactor. The results of the study indicate that the flap valve would not be required as a safety device, since core cooling was predicted to be adequate during the transient. As an alternative, a permanent hole could be used in the location of the flap valve. This hole would serve the functions of both the bypass line and the flap valve.

INTRODUCTION

MAPLE (Multipurpose Applied Physics Lattice Experimental) reactors are a new class of research reactors developed by Atomic Energy of Canada Limited [1]. They are light-water-cooled research reactors with an open-chimney-in-pool arrangement. The MAPLE class of research reactors is designed to generate a maximum thermal power ranging from 1 to 30 MW.

The MAPLE-X10 reactor is the MAPLE prototype being built at the Chalk River Nuclear Research Laboratories. The reactor together with the layout of the reactor core is shown schematically in Figure 1. For normal full-power operation of the MAPLE-X10 primary cooling system (PCS), water from the core enters the chimney from the bottom and pool water enters the chimney from the top. A pump draws the water from the chimney and delivers it to the heat exchanger. Most of the flow is then directed to the core via the inlet plenum to cool the fuel. The remainder, called bypass flow, returns to the top of the chimney via the pool and mixes with the core flow in the chimney. For normal shutdown operation, decay power is removed by a natural circulation flow through the PCS. One design being considered for the MAPLE-X10 reactor includes a flap valve located in the pool on the line to the inlet plenum. When the decay power becomes too low to sustain a natural circulation flow via the PCS, or if the PCS becomes disabled, a new natural circulation path through the pool via the flap valve is formed to ensure long-term cooling of the core.

This paper examines whether the operation of a flap valve is needed to assist core cooling during an upset condition in the MAPLE-X10 reactor. The pump discharge line outside the pool was assumed to sever completely and instantaneously. The resulting flow network becomes hydraulically complex since several flow paths in the PCS result from the pipe rupture. During this transient condition, the operation of this flap valve was assessed to determine the core flow and thus the capability of core cooling. Three cases were studied using the CATHENA code [2] with constitutive relationships suitable for MAPLE conditions:

(i) the flap valve is opened when the pressure on the pool side is higher than that on the piping side at the valve,

(ii) the flap valve remains closed, and

(iii) a design combining the functions of the bypass line and the flap valve.

This paper describes briefly the CATHENA code and the idealization of the MAPLE-X10 reactor used in the study. The results predicted by CATHENA are described in detail.

THE CATHENA CODE

CATHENA (Canadian Algorithm for Thermalhydraulic Network Analysis) [2] is a computer code, which uses a one-dimensional, two-fluid model to describe the steam-water flow in a pipe network. Auxiliary models are included to describe the behavior of pumps, valves, abrupt area changes, discharge through breaks, reactor kinetics, and control systems. Although the code has been developed primarily to analyze postulated loss-of-coolant accident scenarios in CANDU® nuclear reactors, the code is applicable to a wide range of problems, including the MAPLE reactor.

The hydrodynamic equations for mass, momentum and energy are solved for each phase (liquid and vapor), resulting in a six-equation model. The equation of state, which defines the phasic densities as functions of phase pressure and enthalpy is required for closure of these equations. In addition, constitutive relationships required in the hydrodynamic equations to specify the rates of phase change, interfacial and wall-to-fluid momentum transfer, and interfacial and wall-to-fluid heat transfer have been formulated from information obtained in the literature. These relationships depend on flow regime, which is predicted using a flow-regime map. A heat conduction equation is solved for temperature distribution within a pipe wall or fuel pin (or simulator), and is coupled with the hydrodynamic equations at the wall-to-fluid interface. A full boiling curve is provided to model convective, boiling and condensation heat transfer processes at the interface. Details of the constitutive relationships are outlined in [2].

The governing equations are represented by finite-difference equations written on a staggered mesh where
pressure, phase enthalpies and void fraction are evaluated at mesh centers, and phase velocities are evaluated at mesh boundaries. The numerical scheme is based on a one-step, semi-implicit method in which the solution is not transit-time-limited [3]. A time-step controller in CATHENA automatically selects the next time step based on accuracy consideration. Conservation of mass is achieved using a truncation error correction technique similar to that used in RELAP5/MOD2 [4].

CATHENA is capable of modelling the non-equilibrium effects of subcooled boiling at low pressures and low temperatures, which would occur in MAPLE during an upset condition. Heat transfer correlations were chosen from the literature to be applicable at the MAPLE operating conditions (low pressure, high subcooling, and high velocity), and were implemented in the code. The correlations used are outlined in [5]. The accuracy of these correlations are being verified using data from the single-pin heat transfer experiments being performed at the Whiteshell Nuclear Research Establishment [6,7,8]. The bundle heat transfer experiments being performed at the Chalk River Nuclear Laboratories [9] will be used to investigate the effect of the pin spacing on the correlations. Both experimental series will also provide void fraction data to verify the void model used in the code.

IDEALIZATION OF MAPLE-X10 REACTOR

The CATHENA idealization of MAPLE-X10 is shown in Figure 2. This figure is drawn so that the relative elevations of the reactor components are displayed.
Reactor Core

The MAPLE-X10 core has 19 fuel channels: 12 hexagonal and 7 circular channels. The hexagonal channel is loaded with a 36-pin driver fuel assembly, while the circular fuel channel is loaded with a 12-pin Molybdenum target assembly. Six of the circular channels are used to deploy the control/shutoff absorber tubes.

The startup-core conditions [10] were used as a basis of the thermal loading of the core. The axial heat flux distributions in each site were also represented in the fuelled section. The radial heat flux distribution in a fuel bundle in a channel was small and thus neglected. To consider channel-to-channel power distribution and different hydraulic conditions, the core was modelled to represent five distinct parallel channels: (1) six low-power driver sites, (2) six high-power driver sites, (3) three Molybdenum sites with control assemblies, (4) four aluminum non-fuelled sites in the shutoff and central site positions, and (5) the gap between the fuel channels.

The heat transfer from the finned fuel pin was modelled using an equivalent diameter, which conserves the fuel mass. However, the channel hydraulic diameters and flow areas were based on the fuel pin including fins. The temperature-dependent material properties of fuel and sheath were used.

The flow resistance coefficients in the 36-pin and Molybdenum channels were obtained from the pressure drop data [11].

Heat Exchanger and Process Water System

The plate-type heat exchanger was modelled as 52 parallel passes. Each pass had the surface area and mass of one heat exchanger plate. The boundary conditions at the secondary-side of the heat exchanger were established to remove the 10 MW power from the primary side at steady-state conditions. The flow and inlet temperature of the process water on the secondary-side remained constant during the entire transient.

Reactor Trip and Decay Power

The reactor was assumed to trip when the flow in the piping to the inlet plenum dropped below 80% of initial steady-state flow. A reactor trip delay time of 0.5 s was applied. The core decay power curve used was obtained from [12]. It was assumed that decay power remained at 2% of the initial power after 18 s into the transient.

Break and Equipment Room

Figure 3 illustrates the modelling of a guillotine break at the pump discharge line in the equipment room. An instantaneous break was initiated by fully closing the valve on the normal flow path and by fully opening the valves leading to the equipment room at time zero.

The equipment room was assumed to be sealed and thus prevented loss of primary fluid into the environment. The volume in the equipment room was modelled as large vertical pipes with one end open to the atmosphere. The volume and height were estimated such that hydrostatic head could be properly represented.

Air Entrainment into the PCS

The primary pump continued to operate during the transient, and drew pool water into the equipment room through the chimney top. When the pool water level reaches the top of the chimney, water level in the chimney drops rapidly. As a result, an air/water mixture in the chimney was pumped into the PCS. The air was modelled as steam but with no condensation.

RESULTS AND DISCUSSION

Initial steady-state operating conditions at full power of 10 MW were simulated prior to a transient analysis. The primary pump delivered a head of about 30 m of water. The initial core and bypass flow rates were about 250 kg/s and 27 kg/s, respectively. Core inlet and outlet temperatures were 36.7 and 45.0°C, respectively.

For all three cases, a guillotine break was assumed to occur instantaneously at the pump discharge line in the equipment room. The transient calculations were completed up to 1000 s from the break initiation. The three cases differed only in modelling of the flap valve:

<table>
<thead>
<tr>
<th>Case</th>
<th>Descriptions</th>
</tr>
</thead>
<tbody>
<tr>
<td>(i)</td>
<td>the flap valve was opened when the pool side pressure at the valve exceeds the inside piping pressure.</td>
</tr>
<tr>
<td>(ii)</td>
<td>the flap valve was closed.</td>
</tr>
<tr>
<td>(iii)</td>
<td>an alternative design of a permanent hole, which is located at the same elevation of the flap valve. The flow resistance was adjusted to direct the same bypass flow (10% of the total flow) into the pool for initial steady-state conditions.</td>
</tr>
</tbody>
</table>

Figures 4 to 10 compare the results obtained for the three cases.
Figure 4 shows the predicted short-term (0 - 10 s) and long-term (0 - 1000 s) temperature histories of the peak sheath and fuel centerline, and of the core-exit coolant in the 12-pin Molybdenum channel. Fuel in the 12-pin channels had the highest heat flux in the core. For all cases, the low flow trip (80% of initial steady-state flow) was detected within 0.1 s. The peak temperatures occurred around the trip initiation time. The predicted peak temperatures at the fuel centerline and the sheath were approximately 210 and 150°C, respectively, and are within 3°C for all cases. For all cases, no steam formation was predicted due to the presence of high subcooling in the core during the entire transient.

As shown in Figure 4, the temperature histories of the alternative design of a permanent hole are predicted to be very similar to those of case (ii). This is expected since the bypass hole of case (iii) is hydraulically equivalent to the bypass line in case (ii) since both flow resistances were adjusted to direct the 10% bypass flow into the pool. For cases (ii) and (iii), small second peaks were predicted at about 2 s as the core flow reversed. The high flow resistance of the hole limited pool water from entering into the PCS through the hole. Since the flow into the PCS was small, the PCS behaved like it was closed.

Figure 5 shows the predicted short-term (0 - 10 s) and long-term (0 - 1000 s) flow velocity transients in all fuel channels. For all cases, the flow velocities in all channels were nearly the same during the transient. As shown in Figure 5(i), the core cooling was predicted to be maintained in two stages. Initially, until the pump starts to degrade, little flow in the core was maintained by the pump from the pool via the flap valve. During this period, the flow drawn into the PCS via the valve was split into the core by the pump head, and into the equipment room by siphoning. As a result, the core flow did not reverse during the transient. Later, as the pool water level was lowered in the chimney, the core flow oscillated.

As shown in Figure 5(ii) and 5(iii), during the period of about 2 to 22 s in the transient, the core was cooled by the downward flow. As the pool water level continued to drop between about 22 and 94 s, the core flow was also reduced. This is caused by a near balance of the pump head (upward through the core) and the siphoning head (downward through the core). After about 140 s when the pool water was drained part way down into the chimney, the core flow was reduced to about the same as in case (i), and was again oscillatory. For later stages of the transient, more core flow was delivered into the core for cases (ii) and (iii) than case (i) (Figure 5).

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**Figure 4:** Peak fuel sheath and centerline, and core-exit coolant temperatures in 12-pin Molybdenum fuel

---

**Figure 5:** Predicted flow velocity transients in all fuel channels
Figure 6 illustrates the event sequence during pool draining. This figure shows the liquid levels and flow directions in the PCS at 40 and 140 s. At 40 s, the liquid level in the equipment room reached the broken pipe, and resulted in the initiation of siphoning by refilling the empty pipe downstream of the break. At 140 s, oscillatory flow commenced in the PCS.

Figure 7 shows the directions and magnitudes of the flow across the flap valve for the three cases. For case (i), the pool water was drawn into the PCS via the valve up to 88 s into the transient. Later, the PCS flow entered the pool bottom via the valve. Figure 7(ii) shows that no flow entered the pool since the flap valve was closed. Figure 7(iii) shows that very little flow was exchanged via the hole between the pool and the core inlet line due to the high flow resistance across the hole.

Figure 8 shows the head delivered by the pump and the void fraction at the pump suction for the three cases. For all cases, during initial few seconds of the transient, void was predicted at the pump suction. This was caused by the suction pressure reduced by "flashing" due to the increased flow in the suction line. The void condensed soon after the reactor trip as the coolant temperature at the suction was further reduced. After about 140 s, the void fractions were predicted to oscillate between about 0.2 and 0.4 for all cases. The transient behavior of the pump head and the void fraction at the suction were predicted to be similar for all three cases. The pump head was predicted to be oscillatory due to the air entrained from the chimney in later stages of the transient.

As shown in Figure 9, pool water was initially drained by pumping at the upstream end of the break and by siphoning at the downstream end of the break. Thus, pool water was lost quite rapidly into the equipment room for all cases. For all cases, the pump initially withdrew much larger flow than initial steady-state flow due to the much reduced flow resistance in the new pumping circuit resulting from the break. During this period, the majority of the flow was drawn from the pool via the chimney top rather than from the core. Initially, the pool water was pumped out via the upstream end of the break and also siphoned out into the equipment room via the downstream end of the break. For case (i), part of the pool water which entered the PCS via the flap valve, was siphoned into the equipment room via the break. As shown in Figure 9, the water was siphoned out more for case (i) than cases (ii) and (iii) during this period. This process was similar for all cases, and continued until the pool water level was lowered down into the chimney. After about 140 s, the pump head degraded due to the air entrained from the chimney.
Subsequently, the pump continued to provide flow into the core by establishing the hydrostatic head difference between the chimney and the equipment room.

Figure 10 shows collapsed liquid levels in the pool, equipment room and chimney for all cases. The bottom of the chimney was used as a datum for the levels. After about 140 s, the water levels became quasi-steady for all cases. Then, the pump maintained the hydrostatic head between the collapsed levels of the chimney and room level. This head was the driving force through the core for siphoning. In each case, little flow was exchanged between the pool and the PCS although there were connections between them through the bypass, flap valve and/or hole.

CONCLUSIONS

A study using CATHENA was undertaken to assess the requirement for a flap valve in the MAPLE-X10 research reactor. The results show that:

- adequate core cooling is maintained even in the absence of the flap valve, provided that the reactor is tripped. Thus, the flap valve would not be required as a safety device. As an alternative, a permanent hole, which combines the functions of both the bypass line and the flap valve, could be used;
FIGURE 7: MASS FLOWS AT FLAP VALVE AND IN INLET LINE TO THE CORE

FIGURE 8: PUMP HEAD AND VOID FRACTION AT PUMP SUCTION

FIGURE 9: MASS FLOWS INTO EQUIPMENT ROOM FROM UPSTREAM END AND FROM DOWNSTREAM END OF THE BREAK
there is no flow reversal in the core when the flap valve is opened on pressure differential, but the flow reverses early in the transient when the flap valve remains closed or when a permanent hole is used. When quasi-steady liquid levels are established in the chimney and the equipment room, less flow enters the core for the case with the flap valve opened on pressure differential than the cases with the flap valve closed or with a permanent hole; and

- the PCS behavior and the resulting fuel temperature behavior with the permanent hole are nearly the same as those with the flap valve closed.

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EXPERIMENTAL DESIGN VERIFICATION
OF THE HYDRODYNAMIC PORTS FOR
THE AMPS REACTOR

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ABSTRACT

The ECS-designed hydrodynamic ports (HDPs) are components of the Autonomous Marine Power Source (AMPS) reactor heat source, and are located in the core inlet and outlet legs of the primary heat transport circuit. Thermalhydraulic analysis and experimental design verification of the HDPs are necessary to optimize the design and to demonstrate operational and safety objectives. This paper describes (i) experimental determinations of HDP pressure loss coefficients for a number of HDP designs, (ii) comparisons of these values with theoretically derived coefficients, and (iii) the experimental verification of preferred inlet and outlet HDP designs. For the verification test, HDPs were installed in a single pair of legs of the AMPS test facility and measurements of the net exchange flow through the HDPs were made under various operating conditions. These measurements demonstrated that HDPs can operate with zero exchange flow at AMPS design conditions. The measurements also showed that at off-design conditions the HDPs restrict exchange flow to satisfactorily low levels. Transient data confirmed that the exchange flow is self-correcting, and decreases to zero under a wide range of operating conditions. The steady state values of exchange flow predicted using an in-house developed computer code agree favourably with those measured. The self-correcting feature of the exchange flow is also predicted using the code.

INTRODUCTION

The development of the AMPS by the ECS Group of Companies, and the supporting thermalhydraulic experimentation program have been described by Hewitt [1] and Gray et al. [2]. Important components in the AMPS primary heat transport system (PHTS) piping are the hydrodynamic ports (HDPs), which are located in the core inlet and outlet legs. They provide openings both above and below the reactor core for unobstructed exchange between the coolant in the PHTS and that in the reserve coolant tank (RCT). The purpose of the HDPs is twofold:

1. To permit sufficient flow of water from the RCT to cool the reactor after normal shutdown for decay heat removal, and during postulated accident conditions. This flow is maintained using natural thermosyphons without the benefit of pumps or pressurized injection systems.

2. To convey pumped coolant into and out of the reactor core assembly during normal operation while limiting exchange flow between the PHTS and the RCT, thereby minimizing losses in overall plant thermal efficiency.

During normal operation, the passage of the PHTS coolant is in the form of through-flow along the center axis of the HDP assembly. Under these cooling conditions, the exchange flow between the PHTS and the RCT water is minimized by balancing the total pressure drop through the reactor core assembly against the static head of water in the RCT. On departure of the AMPS system from normal operating conditions, HDP branch flows circulating into and out of the RCT are established by natural convection. The transition from forced to natural circulation is accomplished by purely passive means, an important safety feature of the AMPS design.

In order to design HDPs that would perform satisfactorily, comprehensive pressure loss characteristics of typical HDPs were determined theoretically and experimentally both for through and branch flows. The pressure loss characteristics were obtained in the form of the through-flow and branch flow loss coefficients as a function of flow split. Loss coefficients are of greatest interest at flow splits in the range of 0 to 5 % and at 100 %. Following the pressure loss characteristics testing, appropriate designs for the inlet and outlet HDPs were chosen for the design verification tests. These tests provided verification of basic design principles and thermalhydraulic computer codes prior to the start of the final phase of testing which was also defined in reference [2].

THEORETICAL ANALYSIS OF HDP PRESSURE LOSS CHARACTERISTICS

Basic HDP Design

The HDPs incorporate a series of slot openings in the inlet and outlet legs of the reactor PHTS to allow exchange flow between the core and the reserve coolant tank water. An insulated housing called the anti-convective port (ACP) is attached to the HDP, surrounding the exchange flow slots. The purpose of the ACP is to prevent direct mixing of the hot pumped coolant from the exchange flow slots with the relatively cool water in the reserve coolant tank. For
the purpose of performing a simplified theoretical analysis of pressure loss characteristics, HDPs can be treated as piping tees with guide vanes in the branch lines. The guide vanes ensure that combining branch flows enter the run at right angles. Figure 1 shows a tee with a steady branch flow, \( W_3 \), combining with the through-flow, \( W_1 \). The following equations can be written for the control volume shown, for combining flow,

\[
W_2 = W_1 + W_3 \quad \text{(continuity)} \quad (1)
\]

\[
(P_1 - P_2)A = W_3U_3 - W_1U_1 \quad \text{(momentum through run)} \quad (2)
\]

where \( P \), \( A \), and \( U \) are the pressure, flow area and velocity, respectively, for the locations shown. Bernoulli's equation, with appropriate loss coefficients \( k \) and ignoring pipe friction losses, can be written for the tee as follows:

\[
P_3 + pu_3^2/2 = P_2 + pu_2^2/2 + k_{bc} pu_2^2/2 \quad \text{(branch flow loss)} \quad (3)
\]

\[
P_1 + pu_1^2/2 = P_2 + pu_2^2/2 + k_{te} pu_2^2/2 \quad \text{(through-flow loss)} \quad (4)
\]

where \( p \) is the fluid density, and the loss coefficient subscripts 'te' and 'bc' refer to the through-flow and branch flow, respectively, for combining flow. Using equations 1, 2, and 3, and assuming that for small branch flows \( P_3 = P_1 \), the combining branch loss coefficient can be written as,

\[
k_{bc} = -1 + 4q - (2-(A_2/A_3))^2 q^2, \quad \text{where } q = W_3/W_1 \quad (5)
\]

Combining equations 1, 2 and 4, the through-flow loss coefficient as a function of flow split, \( q \), can be written as,

\[
k_{te} = 0 + 2q - q^2 \quad (6)
\]

For small flow splits \( q < 5 \% \), the quadratic terms in equations 5 and 6 are not significant.

For dividing flow it is more difficult to derive a theoretical expression for the branch loss coefficient, \( k_{bd} \). It is defined by the following expression,

\[
P_1 + pu_1^2/2 = P_3 + pu_3^2/2 + k_{bd} pu_3^2/2 \quad (7)
\]

For zero branching flow (i.e., \( W_3 = 0 \)), and assuming \( P_1 = P_3 \), then

\[
k_{bd} = 1 \quad (q = W_3/W_1 = 0) \quad (8)
\]

Because the pressure distribution across the branch line at its junction with the through-flow line is not known, the variation of \( k_{bd} \) with flow split cannot be determined from simple momentum considerations.

An analysis similar to that shown for combining flows can be done for the through-flow loss coefficient for dividing flow, i.e.,

\[
W_3 = W_1 + W_2 \quad (9)
\]

\[
P_1 + pu_1^2/2 = P_2 + pu_2^2/2 + k_{td} pu_2^2/2 \quad (10)
\]

Combining equations 2, 9 and 10, the through-flow loss coefficient equation for dividing flow can be written as,

\[
k_{td} = 0 - 2q + q^2, \quad \text{where } q = W_3/W_1 \quad (11)
\]

It can be seen that for small values of \( q \) the loss coefficients in equations 5, 6, 8 and 11 are independent of branch flow area and not significantly affected by the quadratic terms. Around \( q = 0 \), the combining branch loss coefficient in equation 5 varies linearly with flow split, i.e., \( dk_{bc}/dq = 4 \).

The change in branch loss coefficients \( k_{bc} \) and \( k_{bd} \) with flow split is an important design parameter of HDPs. Under normal AMPS operating conditions, exchange flow should be zero. However, under certain circumstances (e.g., small changes in pumped flow or reserve tank water temperature), there may be a tendency for either ingress or bypass exchange flow. Ingress exchange flow enters the reserve coolant tank through the branch slots of the outlet HDP and leaves through the branch slots of the inlet HDP. Bypass exchange flow enters the reserve coolant tank through the branch slots of the outlet HDP and leaves through the branch slots of the inlet HDP. Ingress would be limited by increases in \( k_{bc} \) and \( k_{bd} \) for the inlet and outlet HDPs, respectively. Bypass would be limited by increases in \( (k_{bd} - k_{bc}) \) and \( (k_{bc} - k_{tc}) \) for the inlet and outlet HDPs, respectively. It can be seen that, in either case, large positive values of the linear coefficients in \( k_{bc} \) and \( k_{bd} \) would be desirable for restricting exchange flow at low flow splits.

A method of raising the linear coefficients is described in the next section.

**Improved HDP Design**

As discussed previously, the magnitude of the linear coefficients in the correlations for \( k_{bc} \) and \( k_{bd} \) can be regarded as measures of the performance of the
HDPs as exchange flow suppression devices during normal operation. The coefficients can be increased by adding a bypass pipe around the branch port of a basic HDP, as shown in Figure 2. The following analysis for combining flow for this HDP with bypass (HDP/BP) shows the effect of the addition of the bypass line.

The pressure drop through the run must be equal to the pressure drop through the bypass line, i.e.,

\[
\frac{(W_b^2 - 2W_1W_b)}{\rho A_1^2} + \frac{1}{\rho} \frac{2(W_1-W_b)W_3^2}{A_1^2} = \frac{1}{2\rho} \frac{W_b^2}{A_b^2}
\]

where \(W_b\) is the flow through the bypass line. Solving this equation for \(W_b\) as a function of \(W_a\), we find that for small values of \(W_3\), \(W_b \approx W_3\) over a wide range of \((A_1/A_b)\). Assuming that \(W_b = W_3\) and solving the following two equations for \(k_{bc}\):

\[
P_3 + \frac{1}{\rho} \left(\frac{W_1-W_b}{A_1}\right)^2 = P_2 + \frac{1}{\rho} \left(\frac{W_2}{A_1}\right)^2 \quad \text{(momentum)}
\]

\[
P_3 - P_2 = \frac{-1}{2\rho} \left(\frac{W_3^2}{A_3}\right)^2 + \frac{1}{2\rho} \left(\frac{W_2^2}{A_1}\right)^2 + \frac{k_{bc}(W_2^2)}{2\rho} \left(\frac{W_2}{A_1}\right)^2 \quad \text{(modified Bernoulli)}
\]

yields:

\[k_{bc} = -1 + 8q + ((A_1/A_3)^2 - 8)q^2 \quad (12)\]

It can also be shown for dividing flow that the flow in the bypass line is in the reverse direction, \(W_b = W_3\) (for small \(W_3\)), and that,

\[k_{bd} = 1 + 4q + 2q^2 \quad (13)\]

A comparison of the equations for \(k_{bc}\) and \(k_{bd}\) for an HDP/BP (12 and 13) with the corresponding equations for a basic HDP (5 and 8), shows that the linear coefficient for \(k_{bc}\) has increased from 4 to 8, and the linear coefficient for \(k_{bd}\) has increased from 0 to 4. This means that at low flow splits, the pressure difference needed to drive exchange flow into an HDP/BP is greater than for a basic HDP.

**EXPERIMENTAL DETERMINATION OF PRESSURE LOSS CHARACTERISTICS**

Component tests were undertaken at Stern Laboratories Inc. in Hamilton, Ontario to experimentally determine the detailed pressure loss characteristics of several proposed designs of inlet and outlet HDPs. HDPs were tested with and without bypass lines in order to compare their loss characteristics over the entire flow split range.

**Test Facility**

The flow loop used for testing each HDP consisted of a circulating pump, heat exchanger, testing tank, test section, and the necessary piping and control valves. Figure 3 is a schematic of the flow circuit used for testing outlet HDPs with combining flow. Dividing flow was obtained by closing valve 2 and opening valve 5. A somewhat different flow circuit was used for testing inlet HDPs.
Each HDP to be tested was mounted in a horizontal, cylindrical steel tank which was itself full of water. For the combining flow shown in Figure 3, the exchange flow passed through valve 2, into the test tank and entered the HDP through the exchange slots. The through-flow, WA, passed through valve 4 to a length of pipe which was welded to a coupling, into which the through-flow tube was inserted. The coupling/through-flow tube joint was sealed with an O-ring. The upstream tap (1) was located in the coupling, whose internal diameter was equal to that of the through-flow tube. Downstream of the HDP was an extension piece 150 mm long with an inside diameter equal to that of the through-flow tube. The downstream pressure tap (2) was connected to the extension piece 75 mm downstream of the HDP. The downstream piping passed out of the tank, and was connected to the return line immediately upstream of valve 3. The branch tap (3) was located in the wall of the tank at the same elevation as the centreline of the HDP through-flow tube. The flow area of the tank was large enough that the dynamic pressure at tap 3 was negligible.

The pressure transmitters used to calculate the loss coefficients — DP12, DP13 and DP32 — were measured with Rosemount differential pressure transducers. Consistency checks of the pressure measurements were easily made by comparing DP12 with the sum of DP13 + DP32. Agreement within 1% of the largest of the three measurements was considered acceptable. Pressure transmitters were also used to measure the pressure differences across the orifice plates for flow calculations. Because of the wide branch flow range (0.25 to 35 kg/s), three different orifice plates were used to measure each of the flows WB and Wc. Comparison of the flows measured by orifice plates B and C was easily made by diverting the pumped flow through valves 2 and 5. Again, agreement within 1% was considered acceptable.

Temperature measurements of the water in the loop were needed to calculate fluid densities used in flow calculations and in loss coefficient calculations. Type T (constantan-copper) thermocouples were used to measure the temperature of the flow upstream of each of the three orifice plates. The uncertainty in density due to uncertainty in temperature measurements was negligible.

The output voltages from the thermocouples and pressure transmitters were monitored by a computerized data acquisition system (DAS). An on-line FORTRAN computer program was used to convert the digital output from the DAS to temperatures and pressures in engineering units and to derive the flow rates and loss coefficients. This program was run on an HP-1000 computer which scanned the DAS output at a user-specified rate. At steady state conditions the voltages were averaged over a user-specified time interval (typically 15 s or 30 s). These average voltages and the resulting derived data in engineering units were stored on magnetic tape for processing later.

Test Procedure

For each design tested, the HDP was mounted in the tank with the sense lines connected. The tank was filled with water, the air vented from the loop and the sense lines and zero flow pressure checks completed. Using the smallest range orifice plates (0.4 to 2 kg/s) for measuring WA and Wc, the loss coefficients at small flow splits (q < 5 %) were measured. For dividing flow (WA = 35 kg/s, Wc = 0, Ws ≥ 0), valve 5 was adjusted to maintain the desired branch flow (Wb). Recordings of all temperatures and pressures were taken and the flows and loss coefficients (kbd, kbc) calculated. Similarly, combining flow (WA + Wc = 35 kg/s, Wb = 0, Ws ≥ 0) loss coefficients (kcc, kce) were determined for various values of Ws, controlled by valve 2. Large orifice plates (7 to 35 kg/s) were used to measure WA and Wc at flow splits between 20 % and 100 %.

The medium range orifice plates (2 to 9 kg/s) were used to measure WA and Wc at flow splits between 5 and 25 %. These orifice plates were also used for determining loss coefficients for 100 % dividing and 100 % combining flows at flow rates in the range 2 to 9 kg/s.

Results and Discussion

Outlet HDP (without Bypass). The performance of a basic HDP to be used in an outlet leg of the AMPS reactor vessel was measured. The test results for this HDP are presented in Figures 4 and 5. Figure 4 shows the branch loss coefficients, kbd and kbc, versus flow split for dividing and combining flows, whereas Figure 5 shows the results for through-flow loss coefficients, kdt and ktc. The results derived from analysis of the momentum equation for combining branch and through-flow loss coefficients (equations 5 and 6) are also plotted in Figures 4 and 5, respectively.

The following equations, whose coefficients were derived from regression analysis, are used to describe the data at low flow splits.

![Graph of Branch Loss Coefficients for Outlet HDP](image-url)
Because it is easily shown that \( k_{bc} = k_{tc} - k_{bd} \) at zero exchange flow, i.e.,

\[
k_{bc} = 0.20 - 1.07 = -0.87.
\]

The slopes of the loss coefficient expressions at zero exchange flow are comparable to those predicted previously from momentum considerations (equations 5, 6, 8 and 11).

Outlet HDP with Bypass (HDP/BP). The HDP described above was modified to include bypass passages around the exchange flow slots. The modifications allowed a bypass flow to pass into and out of a circumferential slot between two circular plates. A through-flow hole, equal in diameter to the through-flow pipe of the HDP, was bored through the centre of each plate. One of these bypass passages was placed upstream of the exchange slots, another downstream of the slots. They were connected by five tubes which ran parallel to the through-flow tube. In this way bypass flow could pass radially into one passage, pass through the 5 tubes, and re-enter the through-flow radially out of the other passage.

The measured branch loss coefficients for this HDP/BP are presented in Figure 6. The expressions for \( k_{bd} \) and \( k_{bc} \) versus flow split as derived from momentum considerations (equations 12 and 13) are also plotted.

The following equations, the coefficients of which were obtained from regression analysis, represent the branch loss coefficient data at low flow splits (q ≤ 5%):

\[
\begin{align*}
k_{bc} &= -0.72 + 6.6q \\
k_{bd} &= 0.98 + 1.6q
\end{align*}
\]

When we compare these expressions for HDP/BPs with those for basic HDPs we see that the linear coefficient for \( k_{bd} \) has increased from 0 to 1.6, while the linear coefficient for \( k_{bc} \) has increased from 4.0 to 6.6. This has significance for the AMPS design because for the same through-flow area, HDP/BPs are more resistant to small changes in exchange flow than HDPs without bypass. Alternatively, HDP/BPs with larger through-flow diameters than basic HDPs can be used, allowing larger passive cooling flows.

The linear coefficients in the correlations for the loss coefficients are not the only characteristics of HDPs which are important. The values of \( k_{bc} \) and \( k_{bd} \) at 100% exchange flow are also important because they affect the rate of natural circulation of reserve coolant through the core when the pumped PHTS flow is zero. The flow is combining for the inlet HDPs and dividing for the outlet HDPs in this case, so \( k_{bd} \) and \( k_{bc} \) should be as low as possible for the inlet and outlet HDPs, respectively. Since the values of \( k_{bc} \) for the baseline HDP and the HDP/BP at 100% exchange flow are 1.2 and 1.4, respectively, it appears that the addition of the bypass pipe caused an undesirable increase in both loss coefficients, but particularly in \( k_{bd} \) at 100% exchange flow. Additional experimental results (not reported here) indicate that the increase in \( k_{bd} \) can be largely eliminated by minor design modifications.

Inlet HDP and HDP/BP. The pressure loss characteristics of inlet HDPs were also measured. Compared to outlet HDPs, the configuration of the inlet HDPs is more complicated and not amenable to simple theoretical analysis. However, the measured loss characteristics for inlet HDPs were similar to those for outlet HDPs, and the addition of bypass passages to the inlet HDPs made a significant increase in the rate of change of branch loss coefficients around zero exchange flow.

For application in AMPS, it is found that the HDP/BP design can be optimized to provide a resistance to...
exchange flow under normal operating conditions, while allowing sufficient passive cooling for off-normal and shutdown conditions.

HDP DESIGN VERIFICATION TESTS

These tests were performed to demonstrate that the HDPs could operate with zero exchange flow under normal operating conditions, and that exchange flow could be restricted effectively at significant departures from normal reserve tank temperatures and pumped flow rates.

Test Facility

The HDP verification tests were performed using the full-scale reactor vessel core assembly and piping system described by Gray et al. [2], and used in previous hydraulic resistance and CHF tests. As shown in Figure 7, HDPs were installed in one of the four pairs of inlet and outlet legs with interconnecting shrouds and piping to simulate the reserve coolant tank, which was not actually used for these tests.

The outlet HDP/BP described previously was mounted downstream of the outlet nozzle. The 150 mm long extension piece used in the loss characteristics tests was fastened downstream of the outlet HDP/BP. In order to simulate HDPs immersed in the reserve coolant tank and to contain any exchange flow, cylindrical ACP shrouds were designed which fit around the inlet and outlet HDP/BPs. A 45° elbow was welded to the side of each shroud and oriented so that when the shroud was in place, the mouth of the elbow opened vertically downward. The inlet and outlet ACP shrouds with their elbows were connected by a 4" pipe as shown in Figure 7. This pipe, referred to as the cold leg, acted as a static water column duplicating the hydrostatic head of the reserve coolant tank. Near the bottom of the cold leg was a 4" ball valve which could be closed to stop any exchange flow. Water feed lines entered the cold leg above and below the ball valve. These were used to fill the cold leg with water at controlled temperatures which varied between 15°C and 50°C. In order to simulate a wide range of reserve coolant tank densities corresponding to temperatures in the range 0 to 70°C, with 90°C cooling water in the cold leg, a pump was used to inject water through a 3/4" pipe vertically downward into the 4" cold leg. The pump suction was connected to the cold leg in the horizontal section so that the net addition of water by the injection pump was zero.

The flow of water in the cold leg was measured using an ultrasonic flowmeter. The ultrasonic transducers were mounted on a length of 1" pipe in the cold leg above the injector discharge. The ultrasonic flowmeter was calibrated for flow rates in the range 0 to 0.5 kg/s. The uncertainty in mass flow rate at 0.17 kg/s was about 1.5 %. The pressure difference in the cold leg was measured using two pressure taps; one just below the elbow on the outlet ACP shroud, the other just below the elbow on the inlet ACP shroud. Six thermocouples were mounted in each of the two ACP shrouds to measure the vertical temperature distribution. These temperatures were used to calculate fluid densities and hydrostatic pressures, which when added to the pressure difference in the cold leg, gave the total inlet-to-outlet HDP pressure difference in the cold leg. This was then converted into an effective cold leg water density.

Test Procedure

With the main pump flow rate at 36 kg/s and the fluid temperature nominally at 90°C both in the PHTS circuit and in the cold leg, the steady exchange flow in the cold leg (bypass) was measured with the injector pump off. The cold leg pressure difference and the pumped mass flow rate were also measured and recorded.
The injector pump was turned on and its control valve opened until a steady bypass flow in the cold leg of 0.15 kg/s was obtained. Again, the cold leg pressure difference, pumped flow rate and exchange flow were measured and recorded. The control valve in the injector circuit was opened in steps so that steady bypass flow in the cold leg with water of the desired temperature. The injector pump was turned on and its control valve opened until a steady bypass flow in the cold leg was obtained. Although the injection flow was not measured, its effect on the cold leg pressure difference was measured.

Rudimentary transient testing of the HDP/BPs was also performed. In these tests, the injector pump was not used and its control valve was shut off. The initial conditions for the transient were obtained by filling the cold leg with water of the desired temperature. This was done through the feed lines above and below the ball valve (which was closed). With the main pumped flow conditions set at 36 kg/s and the water temperature at 90°C, the transient data recording was started and the ball valve opened. Transient tests initiated by rapid changes in pumped flow were also completed.

Results and Discussion

The injector pump was very effective in generating conditions in the cold leg which duplicated the effect of higher density cold water. The greater the injector flow, the greater the tendency for ingress, i.e., for water to enter the cold leg through the outlet HDP and leave through the inlet HDP. Conversely, the greater the pumped flow, the greater the tendency for flow to bypass the core, i.e., to enter the cold leg from the inlet HDP and to leave through the outlet HDP.

Results from the steady state tests are shown in Figure 8 which is a graph of flow split versus the static-to-dynamic pressure ratio. Flow split in this experiment corresponds to exchange flow in the actual AMPS configuration. It is defined as the ratio of cold leg flow to main pumped flow, positive for bypass, negative for ingress. The static-to-dynamic pressure ratio is directly proportional to the fluid density difference between the cold leg and the pumped flow through the core, and inversely proportional to the dynamic pressure of the pumped flow. For a nominal pumped flow of 35.7 kg/s and pumped coolant temperature of 90°C, the static-to-dynamic pressure ratio can be expressed as an effective cold leg density. This density, as well as the equivalent cold leg temperature, are also shown on the abscissa. It can be seen that the system operated with zero exchange flow at an effective cold leg temperature of 45°C (990.2 kg/m³). At an effective cold leg temperature of 60°C (1000 kg/m³), the ingress flow was 2% of the pumped flow. At 60°C cold leg temperature (977.7 kg/m³), the bypass flow was 1%.

Accurate modelling of the AMPS thermalhydraulic circuit is an important requirement. Specifically, the model must be capable of predicting exchange flows that would occur under various conditions of pumped flow and reserve tank temperature, using the less characteristics of the inlet and outlet HDPs (or HDP/BPs), and of the reactor vessel. Using the computer code TRACT (Transient Core Thermalhydraulics) — developed at ECS — which models the steady state and transient response of single phase thermalhydraulic networks, exchange flows were calculated for the conditions observed during the steady state design verification tests. The predicted results are included in Figure 8, and as shown, there is excellent agreement between the measured and predicted values.

The results of a transient run are shown in Figure 9, which is a plot of exchange flow split versus time for an initial cold leg temperature of 28°C, and a constant pumped mass flow of approximately 36 kg/s. Just prior to 50 seconds into the transient, the ball valve in the cold leg was opened. The initial ingress flow peaked at 0.9% of the pumped flow and then decreased over time to less than 0.1% by 150 seconds. The ingress was reduced by the decrease in cold leg density caused by the 90°C water entering the cold leg through the outlet HDP. The flow split transient calculated using the TRACT computer code is also plotted in Figure 9 and shows the initial rise in the exchange flow and its subsequent decrease. This phenomenon of self-limiting exchange flow was also observed for initial bypass flows (not shown here).
As the hot water entered the cold leg immediately below the inlet HDP, the effective density of the water in the cold leg was raised.

The self-limiting exchange flow feature was also observed during transients initiated by changes in pumped flow. The results of a typical test are shown in Figure 10, where pumped mass flow and ingress exchange flow are plotted for the same time period. Between 0 and 120 s, the mass flow was decreased from 35 to 21 kg/s. During this time, the ingress flow rose to a peak of 2.5 % of the total pumped flow and then decreased to 0.1 % by 300 seconds. Again, this reduction of the exchange flow is caused by a decrease in effective cold leg density, due to the ingress of hot water.

ADDITIONAL HDP/BP TESTING

The performance of the HDP/BPs will be investigated further during the AMPS system tests which are currently underway. For these tests, an electrically-heated, full-scale prototype of the AMPS reactor vessel has been placed inside a reserve coolant tank with HDP/BPs installed in all four pairs of inlet and outlet legs. Included in these tests will be the addition of ACP shroud extensions to further explore the self-limiting exchange flow feature.

CONCLUDING SUMMARY

The addition of bypass passages to the basic HDPs is an effective method of minimizing increases in exchange flow during normal reactor operation. It has been demonstrated that the reactor can operate with zero exchange flow under normal operating conditions of pumped flow and reserve tank temperature. At higher and lower reserve tank temperatures, the HDP/BP design was such that the exchange flows were kept within acceptable values. A further interesting result was that, over a wide range of reserve tank temperatures and pumped flow rates, the exchange flow was self-correcting; i.e., the exchange flow decreased towards zero in a timely manner. A further conclusion of these tests was that the steady-state operation of the reactor thermalhydraulics is faithfully modelled using an in-house computer code specifically developed for the AMPS design.

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REFERENCES:


Session 11:

Reactor Decommissioning & Waste Management

Chairman:

E. Rosinger, AECL-WNRE
INTRODUCTION

Decommissioning of a nuclear powered generating station can be defined as the removal of the facility from service at the end of its useful life, and the transformation of the facility into an out-of-service state in a manner that provides adequate protection for the health and safety of the decommissioning workers, the general public and for the environment, all within the regulatory requirements of the Atomic Energy Control Board (AEBC).

The design life of a CANDU 600 MW(e) reactor is a minimum of 30 years. Due to a variety of factors impacting on plant capacity and the economics of changing components, the life is expected to be in excess of 40 years. There will be considerable economic incentive to operate it as long as is reasonably possible. Therefore, it is reasonable to assume that decommissioning would begin some 40 years or longer after the in-service date.

The decommissioning process which involves decontamination, dismantling, radiological protection, physical security, and waste disposal activities must be funded by the utilities which have operated these facilities. While these activities typically occur at the end of the operating lives of these facilities, the substantial sums of money required necessitate that an estimate of the cost of decommissioning be made early on, and that mechanisms be put in place to assure the future availability of adequate funding.

Since these funds are to be collected from the rate payers, utilities must have in place a decommissioning cost estimate that has been prepared in a reliable, credible and defensible manner. Moreover, as these cost estimates must also withstand regulatory authority scrutiny, there is an additional incentive to prepare these estimates in a rational and rigorous fashion.

In this context New Brunswick Electric Power Commission and Hydro-Québec, to meet the requirements of the AEBC regulatory document R-90 and as part of the definition of a "plan of action" for the decommissioning of the Point Lepreau Generating Station and Gentilly-2 Generating Station, contracted AECL to provide a cost estimate for the ultimate decommissioning of the facilities. The actual decommissioning operations however are assumed to begin after the postulated plant shutdown date of February 28, 2023.
equipment and structures for ultimate release of the site for unrestricted use.

The U.S. Code of Federal Regulations (10CFR50.82) stipulates that the termination of an operating license be followed by dismantlement and disposal of the facility in a manner which will not be inimical to common defense and security or to the health and safety of the public.

The NRC guidelines for such decommissioning is included in Regulatory Guide 1.86 "Termination of Operating Licenses for Nuclear Reactors" which recommends four options:

a) Mothballing
b) In-place entombment
c) Removal of radioactive components and dismantling
d) Conversion to a new nuclear or a fossil fuel system

The International Atomic Energy Agency (IAEA) has attempted to standardize international decommissioning terminology by defining three broad stages (Technical Reports Series No. 230, 1983) for decommissioning a nuclear facility. These stages or options can be implemented in full, in part, or in some specific combination for a given plant. While there has not been universal acceptance of these options, they were used as a guide in this decommissioning cost estimate for selecting a decommissioning scenario for Point Lepreau and Gentilly-2 stations.

a) Stage 1 Decommissioning (previously called Storage With Surveillance (SWS)). This is similar to mothballing.
b) Stage 2 Decommissioning (previously known as Restricted Site Release (RSR)). This alternative has some similarity with the in-place entombment, although it is somewhat less extensive.
c) Stage 3 Decommissioning (previously referred to as Unrestricted Site Use (USU)). This is similar to removal of radioactive components and dismantling. An extension of Stage 3 is to demolish all buildings and structures and restore the site to a "green grass" condition.

The selection of a decommissioning scenario is highly dependent on many factors. Some of them may appear more predominant than others, depending on the location of the plant and the time when the decision will be made.

The following factors are considered to be of significant importance:

- The plant radioactivity inventory and its characteristics;
- The national policy and regulations;
- The concern for public safety;
- The public opinion;
- The political atmosphere;
- The requirements for the site;
- The economic situation of the plant owner;
- The availability of a commercial low level waste disposal site in reasonable proximity to the plant;
- The availability of a high level national waste disposal site.

Proposed Decommissioning Scenario for Point Lepreau and Gentilly-2

The AECL/CANDU decommissioning scenario which is being given the greatest amount of support these days is a variant of the IAEA Stage 1 (or storage with surveillance) called "static state" followed by a delayed Stage 3 (or unrestricted site release) to the green grass condition. This is the proposed scenario for Point Lepreau and Gentilly-2.

It can be divided into three distinct phases as described below:

- All systems, components and structures which were associated with the nuclear process and either exhibit or contain radioactivity are put into a static state such that with minimal surveillance and maintenance they can be counted on to maintain this containment function for prolonged periods.
- Plant storage under static state with minimal surveillance for a prolonged period. This is around 32 years for Point Lepreau and Gentilly-2 assuming a total decommissioning period of 40 years after shutdown.
- All systems and components which still exhibit some radioactivity following the prolonged storage period are to be removed from the plant and sent to a waste disposal (or storage) site and all building structures are to be decontaminated. This will be sufficient for the unrestricted release of the site.
- In this estimate, all buildings are also expected to be removed and the site is restored to a green grass condition by back filling and landscaping.

In Canada, so far, Gentilly-1, Douglas Point and Nuclear Power Demonstration (NPD) Generating Stations have been put into "static state" based on this decommissioning scenario.

KEY ASSUMPTIONS OF THE COST ESTIMATE

- The decommissioning activities follow a normal operating life cycle of a nuclear plant which has not been subject to any major accident, and is kept relatively clean during operation. This results in an amount of contamination at the final shutdown that is predictable with a reasonable accuracy both in quantity and location.
- The decommissioning activities are to be governed by the existing regulations. Where Canadian regulations are not available, such as in the domain of disposal and waste classification etc., the United States Nuclear Regulatory...
Commissioning (USNRC) and/or the International Atomic Energy Agency (IAEA) rules are applied.

- Only presently available technology and proven engineering methods are credited in undertaking the decommissioning activities.

- Both a national deep geological disposal facility and a national shallow land burial site both located at a National Waste Management and Storage Facility will be available at the time of decommissioning of the Point Lepreau N.G.S. and Gentilly-2 N.G.S. and will be within a reference distance of 2500 km from both stations.

- No abnormal incident such as a seismic event, flood or storm of major proportion is envisaged during the period of decommissioning or dormancy (Static State).

WASTE CLASSIFICATION OF THE POINT LEPREAU AND GENTILLY-2 INVENTORY

During each phase of decommissioning, a decision will have to be made on the disposal of waste represented by the contaminated equipment. To assist in this decision, four categories of waste have been developed, each category includes wastes with a certain degree of contamination. The categories are from zero to four, in increasing order of contamination.

The criteria used to define each system's category are:

Category 3: highly contaminated systems: this includes principally the reactor, PHT and Moderator systems and their auxiliary systems.

Category 2: moderately contaminated systems: this includes all the systems that have a high probability of both internal and external contamination; these generally include secondary systems that process D2O vapor or liquid.

Category 1: lightly contaminated systems: this includes systems for which only some external surface contamination is to be expected.

Category 0: clean systems: for some systems, such as ECC, the assumption of cleanliness is made assuming there is no major plant accident or system malfunction.

Table 1 is a listing, by system identification number of according contamination.

In order to achieve "Static State", it is necessary that all contaminated equipment and material in the systems listed in the table which are located outside the Reactor Building are dismantled and transferred to the Reactor Building (or to the on-site low level waste storage area) for long term storage until final decommissioning.

<table>
<thead>
<tr>
<th>CATEGORY 3</th>
<th>CATEGORY 2</th>
<th>CATEGORY 1</th>
</tr>
</thead>
<tbody>
<tr>
<td>HIGH</td>
<td>MEDIUM</td>
<td>LOW</td>
</tr>
<tr>
<td>32XX</td>
<td>3411</td>
<td>3431</td>
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<tr>
<td>33XX</td>
<td>3451</td>
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<td>3441</td>
<td>3471</td>
<td>713X (in R/B)</td>
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<td>3481</td>
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<td>3498</td>
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<td>7312 (in R/B)</td>
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COST ESTIMATING METHODOLOGY

The cost estimate is prepared using the AECL-DECOM Computer Program (see Appendix 1 for details). A scope of work consistent with the requirements for the selected decommissioning scenario for Point Lepreau and Gentilly-2 is prepared. The scope of the estimate for this report is then divided into four major cost categories to suit the AECL-DECOM program.

These categories are: (1) activity-dependent costs, (2) period-dependent costs, (3) special items costs and (4) dormancy period costs. Appropriate contingencies are then incorporated for each cost category.

Activity-Dependent Costs

Activity-dependent costs are those that are directly related to discrete, generally repetitive activities, (e.g., dismantling, packaging and disposal). They include all labor, materials, equipment and services associated with the activities.

Activity-dependent costs may be calculated by means of unit cost factors (UCF'S). A unit cost factor is the estimated amount of money required to handle one unit of material. It is based on a model which takes into consideration all the typical activities associated with, for example, the task of dismantling piping. The model includes the estimated labor hours, crew size and composition, worker base rate plus fringe benefits, consumable materials, special equipment and subcontractor overhead and profit. In order to obtain site specific costs, labor rates, transportation costs for each station were incorporated in the unit cost factors. With this approach, unit cost factors can be developed for various sizes of piping, valves, pumps, heat exchangers, tanks, and for structural concrete and steel.

UCFS and total costs are computed in four major
categories.
- Dismantling
- Packaging
- Transportation
- Burial (disposal)

Dismantling UCFs are further divided into four groups:

a) Components that are non-radioactive and accessible (workers can work on these without scaffoldings or platforms).

b) Non-radioactive and inaccessible (cost premium for scaffolding).

c) Radioactive and accessible (cost premium for man-rem exposure).

d) Radioactive and inaccessible (cost premiums for scaffolding and man-rem exposure).

Table 2 and Table 3 give a sample of cost factors in 1989 Canadian dollars for Point Lepreau and Gentilly-2. Several UCFs were developed for the decommissioning cost estimates.

### Table 2
**Point Lepreau Decommissioning Cost Estimate**

**Dismantling Unit Cost Factors by WBS (in 1989 Canadian Dollars)**

<table>
<thead>
<tr>
<th>WBS</th>
<th>Component Description</th>
<th>Zero Base</th>
<th>Zero Base + SCA</th>
<th>Zero Base + RAP</th>
<th>Zero Base + SCA + RAP</th>
</tr>
</thead>
<tbody>
<tr>
<td>261</td>
<td>Contaminated Conc H³</td>
<td>-</td>
<td>194.00</td>
<td>233.00</td>
<td>258.00</td>
</tr>
<tr>
<td>301</td>
<td>Pump up to 150 kg.</td>
<td>406.00</td>
<td>488.00</td>
<td>540.00</td>
<td>610.00</td>
</tr>
<tr>
<td>302</td>
<td>Pump 150-2250 kg.</td>
<td>1,788.00</td>
<td>1,413.00</td>
<td>1,567.00</td>
<td>1,767.00</td>
</tr>
<tr>
<td>303</td>
<td>Pump 4500-22500 kg.</td>
<td>2,778.00</td>
<td>3,328.00</td>
<td>3,688.00</td>
<td>4,160.00</td>
</tr>
<tr>
<td>311</td>
<td>Ht. Exch. 450 kg.</td>
<td>341.00</td>
<td>571.00</td>
<td>761.00</td>
<td>857.00</td>
</tr>
<tr>
<td>312</td>
<td>Ht. Exch. 450-1350 kg.</td>
<td>1,605.00</td>
<td>1,426.00</td>
<td>2,135.00</td>
<td>2,408.00</td>
</tr>
<tr>
<td>313</td>
<td>Ht. Exch. 1350-2700 kg.</td>
<td>439.00</td>
<td>715.00</td>
<td>793.00</td>
<td>894.00</td>
</tr>
<tr>
<td>321</td>
<td>Tank 450 kg.</td>
<td>531.00</td>
<td>637.00</td>
<td>795.00</td>
<td>894.00</td>
</tr>
<tr>
<td>322</td>
<td>Tank 450-2250 kg.</td>
<td>596.00</td>
<td>715.00</td>
<td>793.00</td>
<td>894.00</td>
</tr>
<tr>
<td>323</td>
<td>Tank 4500-22500 kg.</td>
<td>1,388.00</td>
<td>1,666.00</td>
<td>1,846.00</td>
<td>2,082.00</td>
</tr>
</tbody>
</table>

**SCA** = Scaffolding  
**RAP** = Radioactive Premium

### Table 3
**Gentilly-2 Decommissioning Cost Estimate**

**Dismantling Unit Cost Factors by WBS (in 1989 Canadian Dollars)**

<table>
<thead>
<tr>
<th>WBS</th>
<th>Component Description</th>
<th>Zero Base</th>
<th>Zero Base + SCA</th>
<th>Zero Base + RAP</th>
<th>Zero Base + SCA + RAP</th>
</tr>
</thead>
<tbody>
<tr>
<td>261</td>
<td>Contaminated Conc H³</td>
<td>-</td>
<td>228.00</td>
<td>274.00</td>
<td>304.00</td>
</tr>
<tr>
<td>301</td>
<td>Pump up to 150 kg.</td>
<td>488.00</td>
<td>586.00</td>
<td>649.00</td>
<td>732.00</td>
</tr>
<tr>
<td>302</td>
<td>Pump 150-2250 kg.</td>
<td>1,406.00</td>
<td>1,687.00</td>
<td>1,870.00</td>
<td>2,109.00</td>
</tr>
<tr>
<td>303</td>
<td>Pump 4500-22500 kg.</td>
<td>3,383.00</td>
<td>4,060.00</td>
<td>4,500.00</td>
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</tr>
<tr>
<td>311</td>
<td>Ht. Exch. 450 kg.</td>
<td>418.00</td>
<td>502.00</td>
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</tr>
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<td>Ht. Exch. 450-1350 kg.</td>
<td>701.00</td>
<td>841.00</td>
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<td>1,051.00</td>
</tr>
<tr>
<td>313</td>
<td>Ht. Exch. 1350-2700 kg.</td>
<td>1,967.00</td>
<td>2,360.00</td>
<td>2,616.00</td>
<td>2,951.00</td>
</tr>
<tr>
<td>321</td>
<td>Tank 450 kg.</td>
<td>536.00</td>
<td>643.00</td>
<td>712.00</td>
<td>803.00</td>
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<td>322</td>
<td>Tank 450-2250 kg.</td>
<td>662.00</td>
<td>795.00</td>
<td>881.00</td>
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<td>323</td>
<td>Tank 2250-4500 kg.</td>
<td>727.00</td>
<td>872.00</td>
<td>966.00</td>
<td>1,090.00</td>
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<tr>
<td>324</td>
<td>Tank 4500-22500 kg.</td>
<td>1,716.00</td>
<td>2,059.00</td>
<td>2,283.00</td>
<td>2,574.00</td>
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</tbody>
</table>

**SCA** = Scaffolding  
**RAP** = Radioactive Premium

11-4 CNS 10th Annual Conference, 1989
The unit cost factors are then applied to the inventory of plant components and structures to derive the dismantling costs. The detailed inventory is obtained from the existing Point Lepreau plant drawings, BN'S, Valve Lists, Line Lists, Equipment Lists. Separate inventory for Gentilly-2 was not developed since both stations are based on the standard 600 NW MIK-1 design. Moreover, all major differences were identified and reflected in the cost estimates. The inventory data sheet used for inputs to the AECL-DECOM Program is presented as Table 4.

### TABLE 4

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<th>FIELD</th>
<th>REFERENCE</th>
<th>BUILDING</th>
<th>ROOM</th>
<th>RADIATION CATEGORY</th>
<th>DEVICE CODE</th>
<th>EQUIPMENT NO.</th>
<th>X-DIM (LENGTH IN FT)</th>
<th>Y-DIM (FT)</th>
<th>2-DIM OR PIPE-DU (FT)</th>
<th>ISSUE/ATS (F, C, P, T)</th>
<th>WEIGHT (LBS)</th>
<th>CC3T CODE</th>
<th>REM STAGE</th>
<th>SHAPE</th>
<th>ZERO BASE</th>
<th>COMMENTS</th>
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The radioactive category for each item has been identified based on the waste categorization described above. Based on the decommissioning Schedule, each item also has been assigned a removal stage to identify the dismantling, packaging, transportation and disposal operations that would occur for that item and the associated costs.

### Period-Dependent Costs

Period-dependent costs have been calculated based on the projected time-phased staffing level of personnel for project management, decommissioning operations, administration, etc. The personnel costs have been estimated from typical yearly salaries for the appropriate job positions for the duration of the period.

### Special Item Costs

Special item costs, as the name implies, are unique or one time specifically identifiable costs associated with a decommissioning project. There are basically, two broad categories of special items costs, "unique" acquisitions and consumables, and large or distinct dismantling or disposal tasks.

The unique category in this estimate includes such items as heavy equipment purchase or leasing, health physics and radiation protection equipment, energy costs, (heating, diesel fuel, gasoline, etc.), office supplies, lease or purchase of site equipment and tools, etc.

Large or distinct special items costs in this report are, for example, for the cost of dismantling the reactor, removal of asbestos, decontamination, etc.

### Dormancy Costs

Based on the selected decommissioning scenario for both Point Lepreau and Gentilly-2, the dormancy period is approximately 32 years. Costs for site security, inspection and maintenance, energy and periodic radiation surveys are included in this category. Further, due to the nature of the selected decommissioning scenario, the unique requirements of the utility and the cooling time needed for the spent fuel the costs for decontamination and removal of the Spent Fuel Bay, Spent Fuel Storage Canister Facility are also considered as part of the dormancy costs. The offsite disposal of spent fuel, which may take place as early as 6 to 11 years after shutdown or at the time of final decommissioning at the latest, however, is not included in this estimate.

### Contingency

Virtually every cost estimate of large construction or decommissioning projects includes some contingency. Contingency is a specific provision for unforeseen elements of cost within the defined project scope, particularly important where previous experience related estimates and the eventual (actual) costs have shown that unforeseen events and the resulting increase in costs are likely to occur. Contingencies are applied on all categories of costs using a consistent basis from prior experience.

### Summation of Cost Data

Once all these individual components are determined, they are combined together to develop the total cost of the decommissioning program.

Following reports for each category of cost have been generated from AECL-DECOM:
- Plant Inventory Report
- Total Decommissioning Cost Report and cost report for each category mentioned above
- Cash Flow
APPLICATION OF THE METHODOLOGY TO THE POINT LEPREAU AND GENTILLY-2 STATIONS

The methodology described in the previous section was applied in a systematic manner to prepare this decommissioning cost estimate. This approach can be applied to any nuclear power plant. Major steps in this approach are as follows:

- Survey of Plant Inventory
- Classification of Plant Inventory into different levels of waste based on radiological characterization
- Development of site specific unit cost factors using current labor rates, material costs
- Preparation of schedule
- Determination of site organization and manpower requirement based on schedule
- Identification and estimation of special items costs
- Determination of Static State or dormancy period and estimation of costs
- Application of AECL-DECOM Code
- Integration of cost and schedule
- Summary of costs

SCHEDULE AND SCOPE AND ITS INTEGRATION WITH COST ESTIMATE

The total duration for decommissioning was assumed to be 40 years after the shutdown. Taking into consideration the requirements from NB Power and Hydro-Québec, the decommissioning schedule is presented in Figure 1.

Based on the selected decommissioning scenario described earlier, the 40 years duration was broken down into three distinct phases. They are:

Phase I consisting of Reactor Shutdown activities and preparation for static state (dormancy period). It is expected to commence on March 1, 2023 after final shutdown and last for 30 months until August 31, 2025.

1) Reactor Shutdown Activities which include:

- Reactor shutdown and inspection
- D2O removal and disposal (does not include costs for detritiation or income from possible salvage).
- Deactivation and drainage of all systems
- Comprehensive radiological survey
- Chemical decontamination of PHT and moderator systems
- Disposal of spent resins and reactor operation wastes
- Engineering and licensing application for station shutdown

2) Preparation for Static State which includes:

- Isolation of power equipment and dismantling of outdoor transformers
- Dismantling and transfer of contaminated material and equipment from all buildings to Reactor Building for long term storage
- Isolation of Reactor Building
- Layout of Static State Control Area
- Final radiation surveys
- Engineering, project management and site support.

The procedures for reactor shutdown and final system operation procedures (FSOP's) and applications for license for shutdown are expected to be initiated by the utilities prior to the station shutdown date of 2023-02-28.

"Static state" or dormancy period in which the site is expected to be maintained under surveillance until the final decommissioning. It is expected to start on September 1, 2025 and last until February 28, 2058. The scope of activities consist of:

1) Maintenance and surveillance which include:

- Site security
- Periodic inspection and maintenance

DECOMMISSIONING SCHEDULE SUMMARY
- Periodic Radiation surveys
- Site upkeep (snow removal and grass cutting)
- Energy to run the required operating systems.

2) Dismantling and Disposal of Spent Fuel Bay and Spent Fuel Storage Canister Site (SFSCS) which include:

- Decontamination
- Dismantling and disposal
- Engineering, project management and site support

The spent fuel removed from the reactor after shutdown is expected to be cooled for 6 years after the defuelling is completed.

At the end of year 7 after shutdown, the Spent Fuel Bay is expected to be drained and decontaminated. The liquid management tank is also expected to be decontaminated and dismantled at the same time. The systems that are left in operation for maintaining the SFB are expected to be deactivated.

The Spent Fuel Bay is expected to be demolished in the 12th and 13th years after shutdown.

Phase 3 consists of the final decommissioning operations to restore the site to a "green grass" condition and release the site for unrestricted use.

Phase 3 is expected to commence on March 1, 2058 and last for 60 months until February 28, 2063.

Final Decommissioning Operations include:

- Removal and disposal of fuel handling systems
- Dismantling and disposal of Reactor components
- Removal and disposal of asbestos
- Dismantling and disposal of material and equipment from all buildings
- Demolition and disposal of all buildings
- Final cleanup and landscaping
- Engineering, project management and site support

COST SUMMARY

Although several alternatives cost studies for different stages of decommissioning could have been developed, only the reference static state/delayed dismantling scenario for decommissioning Point Lepreau and Gentilly-2 was selected for this cost estimate.

This approach is consistent with the specific utility requirements, the present decommissioning philosophy adopted in Canada and past experience in decommissioning the Gentilly-1, Douglas Point and NPD generating stations.

Absolute dollar amounts of costs for major activities in the three phases are not presented here since the site specific data of Point Lepreau or Gentilly-2 will not necessarily have relevance to the cost of decommissioning any other plant.

However, only the total costs for decommissioning each station, arbitrarily defining the costs for Point Lepreau Reactor shutdown activities as one unit, are presented to show a relative comparison for the two stations. It is also expected that the estimate would provide a basis for establishing a benchmark decommissioning cost estimate for a 600MW MK1 plant or a comparable plant. A cost summary based on this method is presented in Table 5. It shows the total costs in nominal 1989 and in terms of discounted dollars (to 1989) assuming that these funds (in nominal 1989 $) are utilized as required at the time of decommissioning.

This approach is presented only to provide a comparison in costs when the time value of money is considered. The utilities, in this case NB Power and Hydro-Québec are expected to conduct their own cost analyses considering such factors and their other Corporate requirements. There is need for utilities to consider the costs of decommissioning earlier on and plan and establish proper mechanism to collect the required funds during the station operation to ensure their availability at the time of actual decommissioning of the facility.

The overall cost breakdown by different cost categories for each station in terms of percentage distribution is given below:

- Reactor shutdown activities including defuelling, D2O removal, resins and operation waste disposal
- Preparation for static state
- Decontamination
- Dismantling & packaging of all other material equipment building and structures
- Dismantling & disposal of reactor reactor and auxiliaries
- Transportation & disposal of all other material
- Removal & disposal of asbestos
- Final clean up and landscaping
- Site services
- Phase 1 & Phase 3 Dormance period
- Project management, Engineering and Construction management

100%

SIMILARITIES & DIFFERENCES BETWEEN POINT LEPREAU & GENTILLY-2

In general, both stations are built on the 600MW MK1 design basis and therefore have identical reactor configurations. Where they differ in terms of the plant layouts is in the auxiliary facilities such as the turbine building, service building, pumphouse, and the solid radioactive waste management facility. These differences have increased slightly the demolition and disposal costs for Gentilly-2 because of greater volume of concrete to be removed due to larger buildings. For the purpose of this cost estimate, we have also assumed that both NB Power and Hydro-Québec will build canister sites for the
TABLE 5
COST SUMMARY FOR POINT LEPREAU & GENTILLY-2 DECOMMISSIONING

<table>
<thead>
<tr>
<th>Activity Description</th>
<th>Point Lepreau 1989</th>
<th>Discounted to 1989</th>
<th>Gentilly-2 1989</th>
<th>Discounted to 1989</th>
</tr>
</thead>
<tbody>
<tr>
<td>1) Reactor Shutdown Activities (11 yrs)</td>
<td>1.0*</td>
<td>0.36</td>
<td>0.92</td>
<td>0.33</td>
</tr>
<tr>
<td>2) Preparation for Static State (2 yrs)</td>
<td>2.21</td>
<td>0.79</td>
<td>2.27</td>
<td>0.81</td>
</tr>
<tr>
<td>3) Static State or Dormancy Period (32 yrs)</td>
<td>2.88</td>
<td>0.77</td>
<td>2.90</td>
<td>0.78</td>
</tr>
<tr>
<td>4) Removal of Spent Fuel Bay &amp; Spent Fuel Storage Canister Site (2 yrs) in Dormancy Period (7 yrs after shutdown)</td>
<td>1.22</td>
<td>0.38</td>
<td>1.23</td>
<td>0.38</td>
</tr>
<tr>
<td>5) Final Decommissioning Operations (5 yrs)</td>
<td>14.6</td>
<td>3.17</td>
<td>15.4</td>
<td>3.34</td>
</tr>
<tr>
<td>TOTAL</td>
<td>21.91</td>
<td>5.47</td>
<td>22.72</td>
<td>5.64</td>
</tr>
</tbody>
</table>

*Point Lepreau Cost for this activity is considered 1 UNIT measure.

The figures are not dollars.

The following assumptions are made.

1. All activities will commence after shutdown.
2. Shutdown for both stations is Feb. 28, 2023.
3. Total duration of decommissioning is 40 years from shutdown.
4. A real discount rate of 5% for all years.

Interim storage of spent fuel, although presently only NB Power has committed to such a facility.

The stations also (though not significantly) differ in terms of the production of operation wastes and spent resins. Gentilly-2 is expected to produce less volume of operations waste and greater amount of spent resins when compared to Point Lepreau in accordance with the available data. It is difficult to explain this difference since both stations are built on the standard 600MW MK1 design concept with an estimated life expectancy of 40 years and will be shutdown approximately at the same time.

The labor rates and distance to the ultimate waste disposal site also differ as site specific data has been used in this cost estimate. In Table 5 a comparison of decommissioning costs for Point Lepreau and Gentilly-2 is shown. The cost variance due to the differences mentioned above is approximately 3.7%, which is not that significant when compared to the cost of decommissioning.

CONCLUSION

In this paper, the methodology and approach developed for the preparation of the cost estimate for decommissioning the Point Lepreau and Gentilly-2 stations is presented. It is based on a proposed decommissioning scenario which consists of:

- Phase 1
  - Reactor shutdown
  - Refuelling
  - D_2O removal
  - Removal of operation wastes and spent resins
  - Preparation for the static state (a variant of IAEA stage-1).

- Static state or dormancy period in which the stations are maintained under surveillance.

- Removal of the spent fuel bay during the dormancy period.

- Phase 3
  - Dismantling and disposal of reactor
  - Dismantling & disposal of material & equipment form all buildings
  - Demolition and disposal of all buildings
  - Final cleanup and landscaping

In view of the scope of activities involved in this decommissioning scenario, the scheduled durations estimated for each phase, and the large
amount of data to be considered, computerization of the estimation process reduced the preparation time as well as significantly improved the quality (less omissions, errors etc.) and accuracy (more than 90% of plant inventory has been included) of the data collected using the available information. It would have been extremely time consuming and tedious if done manually.

It is concluded that a decommissioning cost estimate is essential in planning for the future decommissioning of a nuclear power plant and to establish that proper mechanisms are put in place to assure the future availability of adequate funding.

Further, since these funds are to be collected from the rate payers the decommissioning cost estimate has been prepared in a reliable, credible and defensible manner.

Using a validated computer code in the preparation of the cost estimate certainly enhances the accuracy of the data and reduces the preparation time.

The present study has also provided an opportunity to identify the major cost items involved in decommissioning a CANDU 6KM1 Nuclear Generating Station.

REFERENCES


(7) "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates", Volume 1 and 2, prepared by TLC Engineering Inc. for the National Environmental Studies Project of the Atomic Industrial Forum Inc.,

May 1986.

APPENDIX 1

AECL-DECOM was specifically developed by Atomic Energy of Canada Limited to prepare plant decommissioning cost estimates for Nuclear Power plants.

It utilizes close to 150 models or Unit Cost Factors (an approach rapidly becoming the accepted industry estimation methodology) to describe the various decommissioning activities and, in addition to providing decommissioning cost estimates, it can compute predicted radiation exposure to workers and waste volumes for nearly any decommissioning scenarios.

The AECL-DECOM computer program was initially developed in 1983 for use on an IBM mainframe computer. In 1985, in order to provide more flexibility and ease of operation for the users of the AECL-DECOM computer program, the AECL-DECOM Code was converted so as to be useable on an IBM-PC type microcomputer.

The AECL-DECOM program has been used in the past few years to do decommissioning estimates for both CANDU and PWR type reactors. The estimates have been found to be within the range reported in the OECD (Organization for Economic Cooperation and Development) decommissioning cost surveys. It can be adapted to other types of reactors and may be extended to non-nuclear facilities with suitable modifications of the cost codes.

The AECL-DECOM computer program, based on the widely accepted unit cost factor approach, is a versatile tool for applications in decommissioning studies. Current incorporation of the AECL-DECOM into a format suitable for use on a microcomputer provides even greater flexibility and wider accessibility.

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A DEMONSTRATIVE IMPACT ANALYSIS OF THE CONCRETE INTEGRATED CONTAINER

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ABSTRACT

An impact analysis of an irradiated fuel container to simulate an accidental drop onto an unyielding surface has been illustrated. The container which is composed of a thick concrete tank and enclosed both outside and inside by steel liners is modelled and analyzed using the finite element technique. The interactive engineering software PATRAN is used for pre- and post-processing while the analysis is carried out using the hydrodynamic code DYN3D. Results in terms of energy, momentum, contact force, stresses, strains and deformation during and after impact are demonstrated. It is concluded that the damage to both liners and concrete are localized in the vicinity of contact. Stresses decrease rapidly toward remote areas and become insignificant. An impact limiter composed of effective energy absorption material will eliminate damage to the container itself.

INTRODUCTION

Ontario Hydro is currently investigating the use of a reinforced concrete integrated container (CIC) for the dry storage, transportation and possible final disposal of irradiated fuel. One major consideration in this investigation is the assurance of its structural integrity in the event of an accidental drop of the container during transportation. Such an investigation can be carried out thoroughly using an analytical approach where the historical response of the container during its impact onto the ground can be recorded and studied. This paper demonstrates a bench mark analysis of such a drop and illustrates its response during and after impact. The objective of this exercise is to achieve the following:

1. To provide detailed information to support the design of the container.
2. To explore the capability of using an analytical approach to investigate the container.
3. To serve as a tool to support experimental drop tests such as drop orientation study, parametric study and others.

GEOMETRY OF THE CONTAINER

The configuration of the analyzed container is shown in Figure 1. It is a thick concrete tank with a lid at the top. The tank has an outside diameter of 2.6 meters. The concrete thickness for the cylindrical shell portion is 455 mm and for the bottom slab 555 mm. The outside of the container is enclosed by a 10 mm steel liner and the inside by a 15 mm liner. The lid is an inverted "hat" shaped structural component which is filled with concrete.

FIGURE 1: CONCRETE INTEGRATED CONTAINER (SECTION VIEW)

MATERIAL PROPERTIES OF THE CONTAINER

Concrete

The concrete is a specially designed high density mix. Information regarding the mechanical properties of high density concrete is extremely rare and is almost nonexistent in the published literature. The properties used in this analysis are based on the Ontario Hydro test data (Ref. 1) and are as follows:

- Density: 3500 kg/m³
- Compressive Strength: 40 MPa
- Tensile Strength: 3.6 MPa
- Young's Modulus: 50 GPa
- Poission Ratio: 0.18
- Shear Modulus: 21.2 GPa

Two types of constitutive laws are required as the concrete is to be treated as hydrodynamic material, namely: the stress-strain relation and the equation of state. They are described in the following:

Stress-Strain Relation. The stress-strain relation employed in concrete is shown in Figure 2. Among the various proposed formulas to describe the
stress-strain curve, Figure 2 was based on the CEB-FIP curve (Ref. 2) and the formulation recommended in Ref. 3 with application of the above noted property data. In general, various formulations for the curve only differ slightly and are not expected to change the response of the structure significantly. Figure 2 was later converted into a form of stress-plastic strain relation to fit the required input format in the DYNA3D computer code. It is worthwhile to mention that the increase of strength in concrete due to the increase of confined pressure (such as the Mohr-Coulomb failure law) is not applied to this particular exercise. Concrete is considered to have lost its entire strength after crush as shown in the figure.

When the stress is in tension, the relation is controlled by the Young’s modulus and the tensile strength specified above. There is no tension stiffening nor shear retention considered in the computer code and hence the analytical results would be on the conservative side.

The reinforcement in the concrete is not included in the current analysis. Reinforcement plays a less important role under the current drop orientation. It is therefore decided to keep the analysis simple but conservative.

Equation of State. In general the equation of state for a material relates hydrostatic pressure P to the volume change (or density change) and to the internal energy. In this analysis the pressure is assumed to be only dependent on the former as shown in Figure 3. The figure indicates that for a concrete cube subjected to a pressure, the pressure initially increases proportionally to the decrease of volume (or increase of density) until the concrete collapses (crushing) at point A. The proportionality (i.e., the slope of line OA) is the bulk modulus of concrete. After collapse, the concrete engages in a compaction stage (line AB) with less pressure required for further volume decrease until the full compaction is achieved at point B. Thereafter the concrete regains approximately its original bulk modulus for further compression (line BC). If the pressure is released after concrete collapse such as at point E, elastic unloading should occur with a permanent volume change.

If the concrete is subjected to tension, the required negative pressure to increase the volume is similar to that of compression with the same bulk modulus. This continues until the maximum tensile strength of the concrete as represented by the spall limit \(P_\text{lim}\) is reached and then the concrete starts to crack and pressure is dropped to zero.

In the current analysis this pressure-volume change curve is approximated by three pieces of linear lines where point A and B represent the points of intersection. The values of these parameters used in the analysis are:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bulk Modulus</td>
<td>26 GPa</td>
</tr>
<tr>
<td>Bulk Modulus for compacting</td>
<td>13 GPa</td>
</tr>
<tr>
<td>state</td>
<td></td>
</tr>
<tr>
<td>Pressure at point A</td>
<td>160 MPa</td>
</tr>
<tr>
<td>Pressure at point B</td>
<td>3000 MPa</td>
</tr>
<tr>
<td>Volumetric strain at point A</td>
<td>0.006</td>
</tr>
<tr>
<td>Volumetric strain at point B</td>
<td>0.24</td>
</tr>
</tbody>
</table>

Steel Liner

The material for both outer and inner liners is C40.21M 300WT steel to ensure its energy absorption capability being conserved under cold temperature. Its mechanical properties are as follows:

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Density</td>
<td>7850 kg/m</td>
</tr>
<tr>
<td>Young’s Modulus</td>
<td>200 GPa</td>
</tr>
<tr>
<td>Poisson Ratio</td>
<td>0.3</td>
</tr>
<tr>
<td>Yield Strength</td>
<td>300 MPa</td>
</tr>
</tbody>
</table>

If the concrete is subjected to tension, the relation is elastic perfectly plastic as shown in Figure 4. The minimum guaranteed yield strength in
accordance with the CSA Standard for structural steel is employed. In the analysis a negligible amount of strain hardening is introduced to avoid any possible numerical instability at the perfect plastic stage. Essentially the additional strength due to strain hardening is ignored.

FINITE ELEMENT MODELING OF THE CONTAINER

The container is transformed into a mathematical model through the interactive engineering software PATRAN. As the finite element technique will be applied in the analysis, the concrete is simulated by solid elements and the steel liners by shell elements. PATRAN, functioning as a pre-processor, creates these elements and generates a set of input data for later analysis. Due to its symmetry, only half of the container is modelled.

Figure 5 shows the complete finite element model of the system including the container and the rigid surface. The rigid surface is modelled as solid elements and is restrained from movement. The impact regions are modelled by a refined mesh in order to obtain more accurate results at that area. The refinement is gradually reduced for the region

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further away from impact location for economy of computing cost. The pattern of the mesh generated for concrete is identical to that of the steel liners while the concrete is divided into four layers of solid elements though thickness. The finite element mesh for each component are illustrated in Figures 6, 7, 8, 9 and 10 for the outer liner, inner liner, lid plate, lid in-filled concrete and container concrete respectively.

**ANALYTICAL RESULTS AND DISCUSSIONS**

The container, without an impact limiter for protection, is assumed to undergo a nine (9) meter upside down free drop with its corner edge impacting onto a rigid ground. The formal stress analysis is carried out using the hydrodynamic code DYNA3D, an explicit three dimensional finite element code for analyzing the large deformation dynamic response of inelastic solids and structures. The analytical data is then fed back to PATRAN for post-processing. With its color graphic capability, PATRAN sorts, interprets, summarizes and presents the resulting data as follows:

![Velocity Time History at the Tip of the Container](image1)

**FIGURE 13: VELOCITY TIME HISTORY AT THE TIP OF THE CONTAINER**

![Normalized Momentum Time History](image2)

**FIGURE 11: NORMALIZED MOMENTUM TIME HISTORY**

![Displacement Time History at the Tip of the Container](image3)

**FIGURE 14: DISPLACEMENT TIME HISTORY AT THE TIP OF THE CONTAINER**

![Kinetic Energy Time History](image4)

**FIGURE 12: KINETIC ENERGY TIME HISTORY**

![Contact Force Time History](image5)

**FIGURE 15: CONTACT FORCE TIME HISTORY**
Figure 11 is a normalized momentum time history plot of the container. It indicates that the container reaches a full stop at approximately 0.05 seconds. The same result is shown in Figure 12 for the kinetic energy time history of the container. In fact the container gradually picks up kinetic energy again after 0.05 seconds indicating a rebound. Figure 13 provides the velocity time history at the tip of the container which is the first contact point when the container impacts onto the surface. The velocity fluctuates at the beginning due to numerical noise but quickly stabilized to zero which means that the tip physically remains in contact with the rigid surface. It is noticeable that the velocity starts to increase in the positive direction (i.e. physically upward) after 0.05 seconds which implies a rebound of the tip point. A similar response of the tip point is seen in Figure 14 in terms of displacement. Note that the displacement in the plot is in units of one tenth of a millimeter and is very small. Again it indicates numerical noise at the beginning, stabilizes to zero displacement and later starts to rebound after the full stop. Figure 15 shows the total contact force (i.e. impact force) between the container and the rigid surface through the entire impact. It can be seen that the maximum impact force is reached at approximately 0.027 seconds and lasts until about 0.042 seconds before it starts to decline again (i.e. the container starts to disengage from the rigid surface). The peak force is close to 13 MN while the average is approximately 12 MN. These are correspondingly 38 and 36.5 times of the CIC weight (i.e. 38 g and 36.5g respectively).

Deformation of the container is shown in Figures 16 and 17. Figure 16 demonstrates the deformation of the outer liner while Figure 17 shows the inner liner. It can be seen that the outer liner deformed significantly while the inner liner deformed mildly at the bottom edge only.

Stresses and strains are examined at 0.032 seconds and 0.04 seconds which correspond to the early stage and end of the maximum impact force period. It is found that stresses and strains at 0.032 seconds are slightly less than that at 0.04 seconds and hence only the later are reported and discussed herein. Stresses and strains are reduced after 0.04 seconds due to elastic unloading.

Figures 18 and 19 are the Von Mises stress contour plots at the middle surface and the outer surface of the outer liner respectively. The difference between the two plots is that the former represents the in-plane stress while the latter is the sum due to in-plane and bending. Recalling that the yield strength of the liners is 300 MPa, the portion of the liner in red has stresses at or close to yield. Note that the tip (i.e. the initial impact area) may not be subjected to the maximum stress at this stage since it has already buckled.
As the contact area grows larger, the indented tip which initially yielded has already experienced unloading. Stress has subsequently been redistributed toward the outer circumferential region as shown in the figures. Further away, the stress starts to decrease which means that the remaining portion of the outer liner maintains elastic throughout the impact.

Similarly, Figures 21 and 22 are the Von Mises stress and equivalent plastic strain contour plots for the inner liner. It can be seen that the response of the inner liner is similar to that of the outer liner except that both stress and strain levels are much lower.

Figure 23 is a Von Mises stress contour plot for concrete. The impact corner in white represents the range of concrete being crushed. As the concrete is crushed, it is not capable of carrying shear stress and hence the Von Mises stress becomes zero. The pulverized concrete transmits further stress to the unfailed concrete through direct stresses (i.e. pressure).
CONCLUSIONS

A stress analysis of the concrete integrated container impacting onto a rigid surface has been demonstrated. The analysis enables the response of the container in terms of various parameters at different stages of the impact to be studied in detail. This is vital information for the design of the container.

The analysis is conservative because the concrete reinforcements, the additional strength of the steel liners due to strain hardening and the increase in concrete strength due to confined pressure are all not considered. The extent of damage predicted from this exercise can therefore be viewed as the worst possible situation. It is noted that the damage to the outer liner and the concrete are localized in the vicinity of the impact area. The inner liner experiences less damage. However this analysis does indicate that an impact limiter is a necessity in order to ensure the complete structural integrity of the container. It is judged that an impact limiter composed of effective energy absorption material will eventually eliminate any possible damage in the container itself. An analysis with the impact limiter will be carried out in the near future.

ACKNOWLEDGEMENT

This CIC project is coordinated by P.J. Armstrong of The Nuclear Engineering Department of Ontario Hydro. Comments by P.J. Armstrong and R.N. Sumar of NED are greatly appreciated.

REFERENCES


PUBLIC OPINION ON NUCLEAR FUEL WASTE MANAGEMENT IN CANADA

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ABSTRACT

Atomic Energy of Canada Limited (AECL) has been conducting sociological research in support of the Canadian Nuclear Fuel Waste Management Program (CNFWMP) since 1978. This research, predominantly in the form of public opinion surveys and focus group discussions, is intended to assess public perceptions and attitudes toward issues associated with nuclear fuel waste disposal. This paper will describe the results of our most recent public opinion survey conducted in 1988 December in the provinces of Ontario, Quebec and New Brunswick.

INTRODUCTION

The results of our sociological research have consistently shown that the public is concerned about the question of nuclear fuel waste disposal and feels that it is important to physically demonstrate the safe disposal of nuclear waste as soon as possible. Although a broad public consensus exists on the need to manage and dispose of high-level radioactive waste safely and promptly, location of a disposal site near a respondent's community has not had the same unequivocal support.

To assist the waste management program, a number of studies have been carried out to assess public perceptions and attitudes toward the Canadian Nuclear Fuel Vaste Management Program (CNFWMP) and related issues. Findings from the sociological research studies provide an indication of public reaction to, and concerns about, different aspects of the Nuclear Fuel Waste Management Program, and suggest conditions for public acceptance of the disposal concept and the ultimate implementation of the disposal technology.

Atomic Energy of Canada Limited (AECL) has conducted public opinion surveys about the waste management program since 1978*. Up to 1986, these surveys consisted of personal face-to-face interviews conducted by the Canadian Gallup Poll as part of its Ontario omnibus studies. In 1988 December, we commissioned a study by Angus Reid Associates, who conducted telephone interviews with residents in Ontario, Quebec and New Brunswick.

This paper will describe the results of the 1988 December survey. The survey was designed to evaluate public knowledge of the Nuclear Fuel Waste Management Program, to assess attitudes toward various aspects associated with the program, to identify public issues and concerns associated with the concept of nuclear fuel waste disposal, and to investigate public acceptability of siting a nuclear fuel waste disposal facility.

KNOWLEDGE OF THE PROGRAM

Over time, there has been a gradual increase in public awareness of the Nuclear Fuel Waste Management Program. When the question was first asked in 1979 June, 44% of Ontario residents stated they knew a little or a great deal about the program. Surveys conducted since then show a small but gradual increase in awareness levels so that by 1986 April, the proportion indicating some knowledge of the program had increased to 61%. (1) However, in 1988 December, self-assessed knowledge levels had decreased in Ontario to 50%. It is too early to tell if this decrease in awareness levels is a continuing trend, or merely a result of sample fluctuations. In Quebec, only 38% of respondents indicated some knowledge, while 47% of New Brunswickers felt they knew at least something about the program. (2)

Self-assessed levels of knowledge are generally higher for males than females, and for the first time, those with college and university education have lower awareness levels than those with public or high-school education.

Fifty percent of the respondents feel their knowledge of the waste management program has not changed over the past few years. For the remainder, a larger proportion feel their knowledge has increased rather than decreased. (2)

Successive surveys have indicated that knowledge about specific aspects of the CNFWMP appears to be limited. The public seems to be most knowledgeable about the proposed method of nuclear fuel waste disposal. Awareness of other details, such as the lead agency in the research program, the type of wastes involved, and where the wastes are currently stored, is considerably lower. For example, in the most recent survey, only one in five respondents was aware that nuclear fuel wastes are currently stored at nuclear reactor sites. (2)

ATTITUDES TOWARD THE CONCEPT OF NUCLEAR FUEL WASTE DISPOSAL

One of the objectives of the public opinion surveys is to assess public attitudes toward various aspects of the CNFWMP and to identify conditions for potential public acceptance of the disposal concept.

There is some degree of public confidence in, and potential support for, the technical program to find a solution for the disposal of nuclear fuel waste. Although the public is somewhat divided in their opinion about whether or not Canada has the technical capability now to safely dispose of nuclear fuel...
waste, most believe the technology will be developed within the next 20 years. Ontario residents are more likely to feel the technology currently exists (39%), compared to people in Quebec (32%) and New Brunswick (34%). (2)

The results of the most recent survey show more people are opposed to geodisposal than in favour. Support levels for the concept are similar in Ontario (38%) and New Brunswick (39%), but lower in Quebec (31%). Men are much more supportive of geodisposal (42%) than women (29%), but attitudes were not affected by the level of education. Potential support increases when respondents are presented with the alternative of continued storage in cooling bays at nuclear reactor sites. Over one-third of those polled said they would more likely support geodisposal given this alternative, while one-quarter said they would less likely support it. (2)

In Ontario, support for the nuclear fuel waste management concept has decreased since the last survey. In 1988 December, 38% of Ontario residents stated they are in favour of the proposed disposal method of burial in stable rock formations in the Canadian Shield, compared to 53% in 1986 April. Question placement, the change in survey instrument or sample fluctuations could possibly account for this decrease in support. (2)

Past survey results indicated that many people (70%) felt Canada should be studying other disposal options as part of the research program, even if it meant delaying disposal for many years and substantially increasing government costs for research. (1) The most recent survey reveals, however, that only 29% feel Canada should be studying alternatives to geological disposal regardless of how long it takes and how much it will cost. A little over half of the respondents (52%) agreed that continuing research results from other countries of other disposal methods, instead of spending time and money to do more research. A further 15% believe that geological disposal is the best alternative for Canada and we should get on with it. On a regional breakdown, Quebec residents were the least likely to feel we should be studying other disposal options (22%) and the most likely to believe that we should rely on research results from other countries (59%). (2)

Results of previous studies (1,3) have indicated that public acceptance of the disposal concept may depend in part on a physical demonstration that the concept is safe. Few people feel that scientific analysis alone is sufficient to prove the disposal concept is safe. Most people believe that it will be necessary to build a facility to test the concept before the public will accept the waste can be safely disposed of. A period of 20 years or less is seen as adequate to demonstrate safety of the disposal concept.

ISSUES ASSOCIATED WITH NUCLEAR FUEL WASTE DISPOSAL

One of the objectives of the surveys is to identify public issues and concerns associated with the concept of nuclear fuel waste disposal. One of the earlier surveys indicated there exists a high degree of public concern about the disposal of nuclear waste material, and that most of the public (90%) feels there is an urgent need to demonstrate the safe disposal of nuclear fuel waste. (4)

AECL recognizes that social concerns may be as important as technical concerns in the implementation of a nuclear fuel waste disposal technology. Therefore, the need to identify and resolve social concerns becomes important. Before an evaluation can be made of the potential for public acceptance of the disposal concept and its implementation, it is necessary to understand the opinions held on specific issues and the types of concerns people have about nuclear fuel waste disposal. This type of information will enable AECL to adequately address the issues in sufficient time for the evaluation of the concept under the Federal Environmental Assessment and Review Process.

In order to determine where the issue of nuclear fuel waste disposal stands on the public's agenda compared to other environmental issues, respondents were provided with a list of five environmental issues and were asked to indicate the level of importance placed on each one. The results show that the issues of toxic waste disposal (99%), acid rain (98%), the disposal of nuclear fuel waste (97%), PCBs (95%), and the greenhouse effect (87%) are all considered to be extremely important issues facing Canadians today. (2)

There is a certain degree of public confidence in the use of computer models to predict possible future environmental consequences. Confidence was expressed by 70% of the survey respondents that this type of scientific analysis can determine the environmental consequences of acid rain, by 69% for the consequences of nuclear waste disposal, and by 66% for the consequences of the greenhouse effect. The main reasons cited for lack of confidence in computer modelling to predict the environmental consequences of nuclear waste disposal include: it is impossible to predict the future, a lack of faith in computers, and computer programs are subject to human error. (2)

The question of monitoring has become a very important issue for the public. In a previous survey about eight out of every ten people disagreed with the statement that monitoring should not be necessary because it would put the responsibility on future generations to manage wastes that are being generated today. In addition, very few people (5%) agreed with the argument that the technology will be sufficiently safe that monitoring the wastes after disposal will not be required. Rather, the vast majority (91%) believed that monitoring is required because absolute guarantees of safety cannot be given. (1)

In the 1988 December survey, 80% of the people polled were very or somewhat concerned that once the disposal vault has been closed, it would be hard to monitor the wastes in the vault. When given a choice between sealing the vault permanently so that future generations would not have to look after the wastes, or leaving the disposal vault open and relying on future generations to monitor the wastes, a slight majority chose permanent disposal. New Brunswickers were more likely to prefer permanent disposal (60%) compared to residents in Ontario (51%) and Quebec (47%). (2)

When told that current regulations require nuclear waste be monitored and retrievable during the 40 years it would take to emplace the waste in the disposal vault, opinions were somewhat divided as to whether (41%) or not (48%) this monitoring period would be sufficient to allow a decision to be made about sealing the vault. Of those who thought a further monitoring period would be required, approximately one-
third indicated monitoring would be necessary indefinitely or forever. (2)

There is also a fairly high level of concern (70%) about the difficulty of retrieving the wastes after vault closure. However, people were divided about whether (49%) or not (41%) they agree with the argument that there should be a method for retrieving the wastes even if it means it would create another pathway for the wastes to escape to the outside environment. (2)

Similar to past survey results, a substantial proportion of people (43%) feel absolute guarantees of safety are needed before nuclear fuel waste disposal would be acceptable. About half (51%) of the respondents, however, believe that this kind of guarantee is not possible and the public should trust the experts if they say it's safe. (2)

There is a high level of concern across all regions about the transportation of nuclear fuel wastes, with 76% of the respondents indicating they were somewhat or very concerned about this issue. (2)

Although previous studies have indicated a public preference for remote siting, the most recent results suggest that when transportation risks are taken into consideration, these opinions can change. Just over one-third of the people polled feel a nuclear waste disposal facility should be located in a remote area, despite the increased risks from transporting the wastes greater distances. The majority (55%), however, feel a nuclear waste disposal facility should be located near nuclear power plants, reducing the risks associated with transporting the wastes. (2)

Survey results have consistently shown that specific concerns about nuclear waste management center around health and safety issues. The most frequently cited concerns include the effects on health, the safety of the disposal method, radiation and contamination due to accidents or leaks, and environmental effects. Recommendations for alleviating public concerns emphasize developing and assuring the safety of nuclear fuel waste disposal. Respondents suggested the following actions to be taken to lessen public concerns informing the public, conducting more research, finding a safe method of disposal, proving/demonstrating the safety of the disposal method, remote siting, and increasing government involvement and funding.

ATTITUDES TOWARD SITING

In addition to monitoring public acceptance of the CNFWHP, acceptance for siting a waste disposal facility has also been assessed. Successive surveys indicate that acceptance by the public of locating a waste disposal facility near their community remains low, even given that siting would only occur if the research program showed the concept to be safe.

In 1988 December, only 15% of the respondents indicated they would be willing of having a nuclear waste disposal facility located near their community. Eighty-three percent stated they would be opposed, with the majority of those strongly opposed. Support for siting is greatest in northern Ontario (20%). (2)

Direct comparison of attitudes toward siting over time cannot be made because of variations in question wording. However, a general assessment shows that support for siting has remained low, whereas opposition has increased. This increase in opposition can be accounted for, in part, by those who were initially indifferent toward siting becoming more negative. The proportion wanting more information has been consistently high; however, when this group is forced to make a choice between being in favour of possible siting or being opposed, most are opposed.

There are some strongly held positions on some issues associated with siting. Most people (68%) agree that because of the perceived risks of a nuclear waste disposal facility, communities located nearby should be provided with some form of compensation. Strongest agreement exists in northern Ontario (75%) and New Brunswick (67%). A large majority (87%) feel communities that are potential sites for a waste disposal facility should have the right to decide whether the facility should be located near them. There is substantial agreement (85%) that if a nuclear waste disposal facility is not safe enough to put near a populated area, we have no right to impose it on the people who live in a remote area. (2)

Our most recent surveys have attempted to determine the conditions under which the siting of a nuclear waste disposal facility would be acceptable to the public. In 1988 December, respondents were given a list of incentives and were asked to indicate which condition would be the most important in making siting acceptable and which the least important. The most important condition for siting acceptability is independent monitoring of the facility (39%), followed by community control in the siting decision (27%). When asked which conditions were the least important for siting acceptability, respondents indicated the following: a large number of new jobs (27%), government/community compensation agreements (21%), and substantial payments to the community (21%). Regional differences were found in northern Ontario and New
Brunswick for the most important condition for siting acceptability. Northern Ontario residents gave equal weight to independent monitoring and community control, and greater emphasis to increased employment. New Brunswick residents were also more likely to emphasize increased employment. These results, along with those of a previous survey (1) reveal that some form of local control over the facility is more likely to increase public acceptance of siting than monetary incentives.

Similar to past surveys (1,3), the introduction of incentives has a positive effect on support levels for nearby siting. If the conditions chosen by the respondent are provided, the proportion that would be in favor of siting a nuclear fuel waste disposal facility near their community increases. In 1988 December, support increased from 15% before the introduction of incentives to 21% after they were introduced. Regionally, the highest level of support with incentives is in northern Ontario (24%). Incentives had the most positive effect on support levels in the provinces of Ontario and New Brunswick, where support increased by 8% and 7%, respectively. (2)

CONCLUSION

The public opinion studies conducted for AECL have obtained measures of the public's knowledge of the Nuclear Fuel Waste Management Program, the public's evaluation of different aspects of the program, an indication of public issues and concerns associated with the program, and the types of conditions required for public acceptance of the concept and the ultimate implementation of the disposal technology.

The findings from these studies indicate that awareness of the waste management program is not as high as would be expected after ten years of research and development and public information activities. There is clearly a need to provide additional information on the Nuclear Fuel Waste Management Program to the general public so that they may become more familiar with the program and its objectives. The surveys revealed fairly high levels of uncertainty in opinions about some aspects of the program, as well as a desire to know more about the program before forming definite opinions about it.

The findings from the research show there are strongly held opinions about a number of issues that will likely play a role in public support for the concept and its eventual implementation. It appears that public acceptability may be contingent upon developing mechanisms for assuring safety of nuclear fuel waste disposal. Some effort will be required to address the issues of monitoring and retrievability. The public is not very comfortable with the idea of sealing the vault permanently. In order to gain the confidence needed to make a decision about sealing the vault, the public is asking for an extensive monitoring period with the ability to retrieve the wastes. The development of monitoring and retrievability systems, during both the operations and closure phases, will help demonstrate the commitment to safety.

It is apparent that there is a need for increased information, particularly related to health and safety. In addition, evidence that safety standards and regulations are clearly established and in place, as well as confirmation of the credibility/objectivity of the organizations involved in the program, would help to lessen some of the concerns of the public. Public acceptability of siting an eventual nuclear fuel waste disposal facility will likely be enhanced by commitments to community involvement in siting decisions, and local control over the disposal facility.

Generally speaking, there is some degree of public confidence in, and potential support for, the concept of deep geological disposal of nuclear fuel waste. Although that confidence and support is still weak, its enhancement can be expected by a judgement of its safety following an independent technical and environmental review. However, a substantial portion of the public will not be convinced until disposal is physically demonstrated.

REFERENCES


** Unrestricted unpublished report available from SDDO, Atomic Energy of Canada Limited Research Company, Chalk River, Ontario KOJ 1JO
ABSTRACT

Atomic Energy of Canada Limited (AECL) has conducted a public consultation program as an integral component of the research done for the Canadian Nuclear Fuel Waste Management Program (CNFWMP). The public consultation program was designed to identify social issues, and to provide the public with an opportunity to have input into the CNFWMP. This paper describes the background to the public consultation program, the methodology used in conducting the program, and the issues and concerns associated with nuclear fuel waste disposal as identified by the public interest groups involved in the process.

INTRODUCTION

As the lead agency in the Canadian Nuclear Fuel Waste Management Program (CNFWMP), Atomic Energy of Canada Limited (AECL) initiated and designed the public consultation program to facilitate public participation during the development of the concept of deep geological disposal of nuclear fuel waste. The Concept Assessment Documentation will be submitted in 1991 to a federal Environmental Assessment and Review Process for evaluation. The panel will conduct public hearings as part of its review of the concept.

When the CNFWMP began in 1978, public participation focused on information programs directed primarily at informing the public about the research program, seeking feedback from the public, and obtaining public acceptance of field research activities. These public information programs proved effective in meeting these objectives. As AECL began to realize the importance of social issues in the public acceptance of the disposal technology, mechanisms for obtaining public input that reached beyond the traditional techniques of information dissemination were developed. Since the issues are complex, a program that would allow in-depth discussion and evaluation of the issues with a broad cross section of society was required.

The public consultation program was designed to identify social issues, and to provide the public with an opportunity to have input into the CNFWMP. It consisted of two stages: a series of consultative meetings with individual public interest groups, and an interactive workshop at which representatives of participating groups came together to discuss the issues they had identified related to the disposal of nuclear fuel wastes.

BACKGROUND TO THE PUBLIC CONSULTATION PROGRAM

Early Information Programs

At the onset of the research program, public acceptance for geological field investigation was required. To gain this acceptance, AECL initiated a public interaction program in communities where geological research activities were to take place. This interaction program had the objective of allaying the frequent misunderstanding that research drilling was the first step in locating a nuclear fuel waste disposal facility near those communities.

AECL also launched a general information program to inform the public about its nuclear fuel waste management research. These two programs included the production and dissemination of information materials, the provision of public speakers, displays, briefings for elected officials and the media, attendance at public meetings, school visits, the establishment of public information offices, direct mail and the occasional use of advertising. A toll-free information line was also established.

Early Concerns About Social Issues

As AECL engaged in the task of informing the public about the research program for nuclear fuel waste disposal, the importance of the larger social issues that may eventually influence public acceptance of the disposal technology began to be recognized. Several public and political forums dealt with these issues and made recommendations on how they might be resolved. For example, the Ontario Royal Commission on Electrical Power Planning expressed concern that while the technology for managing nuclear fuel waste was advancing, the social and political aspects of waste management were not being dealt with. The report warned of the negative implications to the scientific program of not dealing with these broader issues, and recommended mechanisms be established to ensure a meaningful dialogue between the critics of nuclear power and the proponents. (1) The Ontario Legislature Select Committee on Ontario Hydro Affairs expressed some doubt regarding public acceptance of any technical solution and recommended integrated program management for the technical and public aspects of nuclear fuel waste disposal. (2)

Dealing with the Social Issues

Sociological research studies conducted by AECL to assess public perceptions and attitudes toward issues associated with the waste management program illustrate the wide range of concerns the public has about the concept of nuclear fuel waste disposal. The various public interaction activities undertaken by

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* The results of these studies are provided in a series of AECL Technical Records, TR-19-1 to TR-19-14, available from SDDO, Atomic Energy of Canada Limited Research Company, Chalk River, Ontario KOJ 1JO

CNS 10th ANNUAL CONFERENCE, 1989, 11-21
AECL as part of the public information programs already discussed have provided further indications of public reaction to, and concerns about, different aspects of the CNFWHP.

While these public information programs provide an avenue for public feedback, they are not a complete means of identifying and attempting to address the social issues that surround the acceptance of the technology and its ultimate implementation. AECL recognized that new techniques for identifying and addressing these issues were required.

The social issues are complex, but they essentially revolve around one question: What combination of social and technical criteria does the public require if it is to support the implementation of the nuclear fuel waste disposal technology? In order to attempt to address this question, AECL felt it was necessary to provide an opportunity for interested publics to become involved in a meaningful dialogue where their input would be encouraged and considered.

Public participation reflects principles basic to Canadian society. Among the many social benefits that result from the public participation process are information exchange and public input into the decision-making process. Direct involvement in decisions affecting the public tends to enhance confidence in both the decision makers and their decisions, and lends legitimacy and credibility to the decision-making process.

The need for, and the importance of, public participation is also recognized by federal and provincial government agencies charged with the responsibility of assessing the environmental and social impacts of proposed projects prior to granting approval to proceed. According to Canadian government guidelines, a proponent who proposes to construct a project that involves federal funds is responsible for identifying and addressing the public concerns associated with it.

In the province of Ontario, the Ministry of the Environment strongly advises pre-submission consultation, suggesting that the "...formal review of environmental assessments and the related decision-making process can benefit if the proponent consults the public, government reviewers, and potentially concerned parties in the planning and discussions that precede a formal submission under the Act". AECL initiated its public consultation program in 1984.

METHODOLOGY

Identification of Public Interest Groups

In 1984 August, AECL retained a consultant to develop a framework for classifying and identifying public interest groups that could be invited to participate in the public consultation program, and to ensure a balanced reflection of public concerns on the issue of nuclear fuel waste disposal.

In 1984 November, AECL formally invited 52 groups identified by the consultant to participate in the public consultation program. In addition to the formal invitations, approximately 10,000 copies of the news release announcing the public consultation program were sent to individuals and groups across Canada. By announcing publicly the commencement of the consultation program AECL opened the process to other organizations interested in participating.

As a result of the natural networking among public interest groups, responses to the news release, and referrals to other organizations, the number of potential groups increased from 52 to 64.

Level of Participation

While AECL attempted to involve a wide range of different interest groups in the consultation meetings and interactive workshop, not all groups did provide input on the issues and concerns they considered important. As such, AECL considered them to have participated to some degree in the public consultation program, and their concerns were recorded.

The levels of participation in the consultation program therefore ranged from general discussions with AECL, to letters or papers stating positions on the waste management and the consultation programs, through to a series of meetings with AECL where issues were discussed at length and ranked according to the degree of concern they raised about the Nuclear Fuel Waste Management Program.

Of the 64 groups invited to participate, 38 provided valuable input in terms of issues and concerns associated with the disposal of nuclear fuel waste.

Format of the Public Consultation Program

The public consultation program consisted of two stages: a series of consultative meetings with individual public interest groups, and an interactive workshop at which representatives of participating groups came together to discuss the issues related to the disposal of nuclear fuel wastes they had identified. The varying organizational structure of public interest groups and the need to foster an open exchange of ideas required a process which was informal and flexible. Therefore, the public consultation program was not structured around a rigid format but rather evolved according to the needs and desires of the groups themselves.

Public Consultation Meetings

Once a public interest group was briefed on the CNFWMP and the public consultation program, and had decided to participate in the program, a number of consultation meetings took place. During these meetings the group was encouraged to identify and discuss the issues and concerns they considered to be important to nuclear fuel waste disposal. AECL offered to pay for the incidental expenses incurred by the participants.

On average, three consultation meetings appeared to be sufficient; however, AECL was willing to work with a group for as long as necessary to ensure that the
The purpose of the notes was to ensure that AECL had accurately interpreted the issues and concerns raised so that responses could be obtained and the issues addressed as fully as possible. The notes also served as an aid for discussion at subsequent meetings.

Interactive Workshop. It was recognized that the groups being consulted might not agree on which issues are the most important, which courses of action are preferable, and which conditions are the right ones for acceptance of the disposal concept. Therefore, a mechanism was required for the groups to interact among themselves to discuss and attempt to resolve any issue differences they might have.

In order to provide an opportunity for interaction, AECL engaged a consultant to conduct an interactive workshop. The workshop was designed to allow the participating groups to interact among themselves to clarify and try to develop a consensus on the outstanding issues that need to be addressed in order for the concept of deep geological disposal to be acceptable.

Participants in the public consultation program have indicated a concern about the lack of a physical demonstration that the concept is safe. The groups have strong beliefs that the pre-closure phase should include continued research, very detailed monitoring, and retrieval mechanisms. While some may view the pre-closure phase as a demonstration phase, for the most part there is skepticism about relying on computerized predictions of the long-term safety of disposal after closure.

The results of the public consultation program show that the effectiveness of the disposal concept appears to be a priority. The concerns are associated with the technical aspects and feasibility of the disposal concept, and the general efficient operation of the disposal method. For example, concerns have been expressed that the stability of the disposal vault might be threatened by such things as changes to the rock formation as a result of excavation, natural disasters and human intrusion or development. The effectiveness of the disposal system is seen to rely on such things as the reliability and durability of the disposal containers, and the effectiveness of the engineered barriers.

Concerns about groundwater seepage, stability of the host rock and the impossibility of a 100% guarantee of safety, has resulted in most public interest groups emphasizing the need for post-closure monitoring. Many participants feel extensive monitoring is required to improve understanding of nuclear fuel waste disposal, as a reporting mechanism to indicate the effectiveness of the disposal system, and to alert people in case problems arise. The groups also feel the disposal technology must facilitate the retrievability of wastes in the event of an accident, or in the event that recycling becomes viable in the future. There are strongly held views that the vault should be left open for a period of time to develop public confidence that the disposal concept is safe, and to enhance credibility of the program in the eyes of the public. Participants feel closure should be considered as an option, rather than an absolute decision. Some feel AECL should proceed with the premise of closure but remain flexible in the event that new knowledge, data or technology arises prior to the time when closure would be implemented.

Issues Related to Health and Safety

The public consultation results show that issues of concern with respect to public health and safety include health risks to people living in the vicinity of a disposal vault, the health risks to personnel during the construction and operational phases, radiological effects as they relate to environmental impacts on the food chain and local water supplies, and the uncertainty of long-term safety.

Many public interest groups feel there is a need for more research on the effects of radiation on human and animal populations, the food chain and the environment. The relationship between radiation and cancer is uppermost in their minds and there is a belief that there is no safe level of radiation. They have also criticized what they believe to be the nuclear industry's underestimation of the health hazards from low-level radiation.

Issues Identified Through the Public Consultation Program

A large number of issues has been identified through the public consultation program and has been fully reported elsewhere. For the purposes of this paper, some specific issues will be discussed including issues related to the development of the disposal technology, health and safety, risk perception/assessment, transportation, security and control, siting, and issues related to the review process.

Issues Related to the Development of the Disposal Technology

Results of the public consultation program show that consideration of alternative disposal options is an important issue. Discussion with the public interest groups revealed that the concern lies with the inability to evaluate the acceptability of geological disposal in comparison to other options, and a feeling that the decision has already been made about how nuclear wastes will be disposed of in Canada because other alternatives are not being investigated. The groups believe AECL must be prepared to let its disposal option stand up to scrutiny, and it must be seen to be superior to alternatives or else be rejected.
Some groups have suggested that epidemiological research should be part of the CNFWMP, and that the medical profession should be represented on the Program's Technical Advisory Committee. They also suggest epidemiological studies be undertaken in the region where a disposal site will be located.

Issues Related to Risk Perception and Risk Assessment

Closely related to health and safety concerns are issues associated with risk. In the public consultation program, concerns have been expressed about the risk assessment approach taken by the nuclear industry and the vast differences in scientific and public perceptions of risk associated with nuclear fuel waste management.

Participants in the consultation program expressed uneasiness about the probabilistic approach taken in determining risks associated with the disposal of nuclear wastes. There is a fair amount of uncertainty regarding the accuracy of predictions, and some skepticism about relying on computer modelling to predict impacts resulting from disposal far into the future. Some of the concern about the accuracy and reliability of computer simulations stems from questions regarding the credibility of the people developing predictive modelling techniques.

Perceptions exist that the risks associated with nuclear fuel waste disposal are exceptional, and that it is inappropriate to compare the risk associated with nuclear waste disposal to other "well-known" risks. The interest groups believe that the public will have difficulty accepting risks over which it has no control, the consequences of which are unknown, and where there is little that they can do to prevent exposure to the risks.

Issues Related to Transportation

The results of the public consultation program provide evidence of the importance of transportation as an issue associated with the CNFWMP. The public consultation groups feel strongly that when considering methods of transporting the wastes, and possible transportation routes, priority should be given to those having the least impact on the population. They also believe that since transportation of nuclear fuel wastes poses a risk, especially with increased distances, a compromise is needed between minimizing transportation risks and sitting in low population density areas.

Concerns expressed about the different modes of transportation relate to containment of the waste in the event of an accident, emergency response capability, environmental impacts, and the selection of routes. A number of questions have been raised concerning proposed transportation routes such as distance, the number of communities that would be exposed to potential hazards, and the environmental impacts in the event of an accident.

While no consensus was reached on the preferred mode of transportation, it appears that road transportation is the least preferred. There was some frustration among the public interest groups concerning the different levels of analysis given to the various modes of transportation in AECL's interim concept assessment document. They feel strongly that in order to evaluate the best mode of transportation, the same level of analysis must be given to rail and barge transport as has been given to truck transport.

Concerns have also been raised about the integrity of the shipping container, and the frequency with which the wastes are handled when selecting container design and the mode of transportation. It is felt that loading and unloading of the wastes should be minimized, and the feasibility of an integrated container, used for both transportation and disposal, should be examined. The importance of the need for emergency response measures, as well as public acceptance for waste transportation, were also emphasized.

Issues Related to Security and Control

The public consultation results indicate that issues associated with securing and protecting the disposal vault and surrounding populations center around such things as the need for trained emergency personnel, concerns about possible sabotage and terrorism, and the need for evacuation plans.

The public interest groups have raised questions concerning security provisions to prevent and deal with sabotage, terrorism, a bomb being brought to the facility, theft of the waste, and non-peaceful uses of the waste by employees or others during transportation, construction and operation of the disposal facility.

A number of recommendations have been made concerning security and control of a waste disposal site. For example, it has been suggested that access to the site be limited and strictly controlled. In addition, the site should remain accessible after closure to permit monitoring and surveillance for security reasons. The site should have a sizable buffer zone to provide room for expansion if needed, to provide site security, and to protect the site from encroachment by incompatible land uses. Finally, the site must be marked or identifiable in perpetuity so that future generations will know and understand what is buried in the disposal vault, as well as developing mechanisms ensure this knowledge and information is passed on.

Issues Related to Siting

Most of the issues raised about siting a nuclear fuel waste disposal facility deal with the potential socio-economic, sociocultural and psychological impacts of siting a nuclear waste disposal facility on affected communities and populations, how these impacts are to be assessed and mitigated, and whether the impacts that remain appear to be generally acceptable. Concerns about compensation and employment opportunities are also frequently raised. Any economic benefits will be evaluated in light of environmental costs and the health and safety of the local population. Environmental impacts due to the siting of a nuclear fuel waste disposal facility are expressed in relation to local water resources and the impact on local food chains.

Equity issues regarding the location of a disposal site have also been raised, and revolve around the distribution of risks, costs and benefits associated with nuclear fuel waste disposal in terms of the location of the disposal site, for example in a northern or southern location, or an urban or rural location.
Many siting issues involve concerns for the development of a fair site selection process that would include consultation and negotiation with residents and other interested parties in the affected area. To provide fairness in the selection of a site, the local population should be an integral part of the siting decisions. It was recommended that AECL indicate in the Concept Assessment Documentation the principles by which siting decisions will be made, and what role communities will have in the siting decision. It is felt that siting criteria should include geological stability, a remote location, and community acceptance. However, it was emphasized that technical and geological criteria cannot be compromised when decisions are made about selecting an appropriate site.

Community input into the siting decisions is viewed as an important component of the site selection process. However, for some public interest groups, community input, due largely to a perceived NIMBY (Not-In-My-Backyard) syndrome, would detract from a process designed to select the most technically suitable site. Some groups have even suggested a political mechanism be put in place to prevent communities with the best technical site from prohibiting siting to take place. While community veto for site selection is considered essential, this privilege should not permit the elimination of all the "best sites".

Issues Associated with the Review Process

A number of concerns were raised throughout the public consultation program regarding the lack of clarity of the review process to be put in place to evaluate the acceptability of the nuclear fuel waste disposal concept. Such questions as who will oversee the hearings, what is to be the structure and mandate of the hearings panel, what is to be the role of the regulatory agencies, and whether or not intervenor funding will be provided, were major concerns. Some groups felt that without a defined review process, they would be wasting their time becoming involved in the public consultation program.

Many of the issues raised in the public consultation program regarding the uncertainty of the review process are addressed by the recent announcement by the federal government of the establishment of a Review Panel to ensure meaningful informed public participation at the public hearings. AECL's publication of the Concept Assessment Documentation and the public acceptance of the technology for disposing of nuclear fuel wastes. AECL's public information programs and sociological research studies provide indications of public reactions to, and concerns about, various aspects of the Nuclear Fuel Waste Management Program. The public consultation program developed and conducted by AECL has provided opportunities for more formal public input, and a more complete means of clarifying social issues.

Social issues have been identified by independent review agencies to be of considerable importance to the public acceptance of the technology for disposing of nuclear fuel wastes. AECL's public information programs and sociological research studies provide indications of public reactions to, and concerns about, various aspects of the Nuclear Fuel Waste Management Program. The public consultation program developed and conducted by AECL has provided opportunities for more formal public input, and a more complete means of clarifying social issues.

The response of the public interest groups to AECL's public consultation program demonstrates a societal need for public participation, and a willingness to work towards developing common solutions to difficult problems. The level of participation acknowledges the appropriateness of the program as a valid participatory technique, and accommodates the desire to have input into the decision-making process. Efforts are continuing to encourage further input from these groups, and to receive input from other groups.

The public consultation program is an important part of AECL's overall effort to identify and address areas of public concern. The interest groups involved in the public consultation program have clarified the issues surrounding nuclear fuel waste management, and have arrived at a general consensus on the outstanding concerns that need to be addressed.

AECL is currently working towards addressing the issues identified through the public consultation program and has made some significant progress. The Concept Assessment Documentation will include a section devoted to a discussion of alternative disposal options. Working parties have been formed to prepare positions on some difficult questions such as monitoring and retrievability. A quality assurance system has been developed to track the issues in the documentation and to ensure that they have been adequately addressed. Addressing the issues is assisting AECL to prepare for the review of the concept, and will ful-

CONCLUSIONS
fill AECL's commitment to the public interest groups to consider their input and take it into account in the preparation of the Concept Assessment Documentation.

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THE PROPOSED SPENT FUEL DRY STORAGE FACILITIES
AT POINT LEPREAU NUCLEAR GENERATING STATION

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ABSTRACT

The New Brunswick Electric Power Commission recently contracted with Atomic Energy of Canada for the supply of the engineering, technical services, tooling and equipment required to implement a spent fuel dry storage program at the Point Lepreau Nuclear Station. Commencing in 1991, spent fuel will be taken into dry storage, over the remaining lifetime of the Station, at an approximate rate of 4,500 bundles per year. The spent fuel bundles will be removed from the Station's Spent Fuel Storage Bay and transferred to dry storage in concrete canisters, located at the Station's Waste Management Facility. Each canister will hold 540 bundles.

This paper provides an overview of the sequential fuel handling operations that are going to take place during the transfer of the spent fuel and describes the tools and significant items of equipment that will be involved. Brief descriptions of the proposed concrete canister site and security features are also provided.

The decision to construct these facilities and to implement the first CANDU 6 spent fuel dry storage program at the Point Lepreau Nuclear Station represents an important achievement in the ongoing commercial development of CANDU technology.

BACKGROUND

As of February 1, 1989, the Point Lepreau Nuclear Generating Station of the New Brunswick Electric Power Commission (NBEP) had completed six years of operation, at a lifetime capacity factor that was amongst the highest achieved in the world. Irradiated fuel has been accumulating in the Station's Spent Fuel Storage Bay at a rate that will see the Bay occupied to its specified capacity by mid 1991.

During September of 1988, NBEP announced that the possible extension of the Station's existing fuel storage bay capacity had been compared in a comprehensive review to the construction of concrete canisters in which fuel would be stored, on-site, in a dry state; and that the dry storage in canisters option (as shown in Figure 1) had been retained.

NBEP contracted with Atomic Energy of Canada in January of 1989 for the supply of the engineering, the technical services and the necessary tools and equipment for the construction and implementation of a spent fuel dry storage program at the Point Lepreau Nuclear Station.

![Diagram of Point Lepreau Canister with 540 Bundles]

NODE OF BAY STORAGE OF THE FUEL BUNDLES

The Point Lepreau fuel bundle is a Zircaloy-clad, natural uranium, 37 element design. The bundle diameter is 102 mm and the bundle length is 495 mm. The weight of a bundle is 23.7 kg, of which 19 kg is the weight of the uranium it contains.

The Point Lepreau Station Spent Fuel Storage Bay is 10.97 m wide, 18.29 m long, and 7.8 m deep (36 x 60 x 25.5 feet). The Bay has space for 102 stacks of 20 fuel trays each. A fuel tray holds 24 fuel bundles; hence the Bay’s total capacity is 48,960 CANDU fuel bundles.
OVERVIEW OF FUEL HANDLING OPERATIONS FOR STORAGE

The fuel handling operation consist generally of transferring the fuel (while still in the Bay) from the trays to stainless steel baskets, removing the baskets from the Bay, drying, sealing and transporting the baskets to the Solid Radioactive Waste Management Facility, and inserting them into thick-walled concrete canisters, for long term retention.

Fuel Basket Loading

Manipulation of the individual fuel bundles takes place underwater, using manual tools and hoists. The empty fuel basket is first placed on the underwater work table. The fuel bundles from a tray are tilted from the horizontal to the vertical position, twelve at a time. The layout of the fuel handling equipment is shown in Figure 2. Using the bundle lifting tool, the bundles are removed from the tray one-by-one, and loaded into the fuel basket.

After the basket has been filled to capacity and the bundles in it noted for IAEA safeguards purposes, the cover is placed over the basket and the basket is ready for removal from the Bay.

Basket Drying and Welding

These operations are performed in a shielded welding station (SWS). The basket is lifted through the shielded loading chute into the SWS and placed on the turntable. It is spray washed, moved into the drying position, and both it and the fuel bundles inside it are dried by means of a hot air blast. The basket is then moved to the welding area in the SWS and the cover seal-welded to the baseplate and the centre post.

Flask Transportation

After the seal-welding has been verified, the fuel basket is hoisted out of the SWS, into the fuel transport flask, which has been in place over the opening in the top of the SWS while the basket has been in the SWS. The loaded flask, with its door closed and locked, is transferred to the trailer. The trailer is pulled out of the building and towed for a short distance to the Point Lepreau Waste Management Area, where the Canister Site is located.

Concrete Canister Loading

At the Canister Site, the flask is brought under the gantry crane, and lifted onto the loading platform using a 30-ton hoist. An auxiliary 3-ton hoist is used to remove the canister plug. As the plug is removed, the flask is positioned over the canister opening to maintain shielding.

The fuel basket inside the flask is then hoisted just enough to take its weight off of the gate. The door is opened, and the basket lowered into the concrete canister, using the flask winch.

The canister plug is put back into place and the flask lowered back onto the trailer, to be returned to the station for the next iteration.

When the canister contains nine baskets, the canister plug liner is seal-welded to the canister liner. An IAEA seal is affixed.

SIGNIFICANT ITEMS OF EQUIPMENT

The significant items of equipment (in order of use) are:
- the fuel baskets,
- the fuel bundle tilter,
- the shielded welding station,
- the transport flask, and
- the concrete canister.

The Fuel Basket

The fuel basket is shown in Figure 3. It is approximately 1050 mm in outside diameter and 550 mm in height. Type 304L stainless steel is used in its construction to assure compatibility with bay water and high resistance to any potential corrosion.
Each basket has a capacity of 60 bundles, arranged in four rings. The bundles are held in position by means of 2 retaining plates. An 102 mm, schedule 80, stainless steel pipe, located along the axis of the basket, serves to uniformly transfer the weight of the 60 bundles (which rest on the base plate) up to the ring at the top into which the grapple engages during the lifting operation.

The Fuel Bundle Tilter

This item of Point Lepreau SFDS Program equipment will reside permanently on the underwater work table, which is at an elevation of 2.75 meters above the bottom of the Spent Fuel Bay. Its function is to reorient the fuel bundles from the horizontal to the vertical position. A grapple is used to transfer the bundles underwater, one-at-a-time, from the trays on which they have been stored to the fuel baskets.

A fuel tray containing 24 bundles, in two, 12-bundle rows is lifted from the bottom of the Bay and placed on the fuel bundle tilter. An undercarriage arrangement "tilts" the inner row of 12 bundles, in unison, so that the axis of the respective bundles is vertical and, hence, amenable to grapple attachment. After these 12 bundles have been removed and transferred to the basket which is positioned alongside, the tray is lifted to just above the fuel bundle tilter, rotated 180°, and lowered back onto it. The remaining row of bundles now rests over the tilter's undercarriage arrangement and these bundles can now be "tilted", removed from the tray, and placed into the basket.

The Point Lepreau program fuel bundle tilter design represents a departure from the design employed in previous CANDU fuel dry storage programs in one significant respect. This tilter is designed to accommodate the standard CANDU 600 fuel tray design, which is a two-row fuel bundle tray arrangement. The previous fuel bundle tilters were designed and fabricated to accommodate a single-row tray arrangement.

The Shielded Welding Station

The shielded welding station (SWS) that has been designed for the Point Lepreau fuel storage program is quite similar to the design employed during several earlier CANDU spent fuel dry storage programs.

The main departure from previous projects is related to the spent fuel bay layout. A 30 ton capacity crane will be added over the SWS and a new building extension constructed to facilitate the flask transfer to the transporter as shown schematically in Figure 4. The station will be constructed of sections assembled into a unit, with its base section anchored to the bay wall concrete in the region of the wall where the transfer canal was to have been located. The walls and ceiling sections of the SWS will be fabricated from 10 mm steel plate on the inside and outside, with 190 mm of lead in between to provide sufficient shielding so that the radiation field at the exterior is restricted to less than 25 microSv per hour, on contact.

There are two rails inside, running along the length of the station, on the base section. The turntable that rests on these rails is moved by a drive mechanism, transporting the basket so that the following activities take place:

- the washing and draining of the basket,
- the air drying of its interior (and the fuel bundles), and
- the semi-automatic seal-welding.

The SWS has two lead-glass windows in the walls for visual observation, and two TV cameras inside for monitoring of the seal-welding operation.

The Fuel Transport Flask

The fuel transfer flask (FTF) that will be used for the Point Lepreau program is different in design from the design used in previous CANDU programs, in three significant areas, i.e.:

- the flask will be circular in cross-sectional configuration (not square), and
- a sliding bottom door will be employed, in lieu of the hinged bottom "shutters" used in the past, and
- its internal diameter can house the larger basket.

The shielding consists of 190 mm of lead between inside and outside steel plate, 9.5 mm in thickness.

As previously, the manual hoist used for lifting and lowering the basket (into and out of the flask) uses chain rather than cable, to prevent basket rotation while these activities are underway.

Radiation fields on contact with the outside surface of the FTF are expected to be less than 25 microSv per hour.
The Canister

The canister is a cylindrical, reinforced concrete structure with an internal liner of 9.5 mm (3/8 inch) standard weight carbon steel pipe. The canister is approximately 3 m in diameter, 6.2 m tall and the liner is 1.12 m in diameter. The canister provides a combined shielding of 0.94 m of concrete and 9.5 mm of steel. The opening at the top is circular and sized to accept the canister plug and adapt to the flask. The carbon steel lined canister plug is semi-welded to the liner at the completion of the loading operation, and the IAEA safeguard seals are installed.

The Point Lepreau concrete canister design is similar to that in use at Douglas Point and is an adaptation of the original Whiteshell Nuclear Research Establishment design. The canisters are designed for deadweight, thermal, wind and tornado driven missiles and seismic loads. They are supported on a common reinforced concrete foundation that rests on a deep bed of crushed stone.

The conceptual design and licensing of these concrete canisters took place in the early 70's and actual demonstration testing in 1975-1976. Quarterly surveillance inspections have been carried out during the ensuing years and no significant deterioration has been observed to date.

The concrete mix is designed to provide a 28-day design compressive strength of 27.6 MPa (4,000 psi). The mix incorporates 5% air entrainment to provide resistance to alternate freezing and thawing cycles.

Two main rebar cages serve to reinforce the canister. One is located alongside the canister liner, the other envelops the canister periphery. Supplementary rebar is also present along the plug/liner interface and between the canister and the base. The rebar at the base serves to anchor the canister to the foundation pad.

The canisters will be sequentially built, at a rate of 10 per year. (About 9 are needed for an 80% capacity factor.) A quantity of canisters will be prebuilt to ensure that the construction area will always be separated from the loaded canisters.

CANISTER SITE DESCRIPTION

The canister site which has been selected is situated adjacent to the Point Lepreau Solid Radioactive Waste Management Facility (SRWHF), on the east side of the existing access road. The canisters will be arranged in two arrays of 150 each (i.e. 30 rows of 5 canisters per row). The two arrays will be separated by a new road, which will serve to provide access to both of the arrays.

Site work will commence with the construction of 15 canisters, in the first array (immediately alongside of the SRWHF). Ten of these canisters will be filled during the first year. During each subsequent year ten canisters will be constructed and ten canisters filled. The loading schedule will be such that there will always be two rows of completed (empty) canisters separating the canisters in the row being loaded from the row that is being constructed. Construction of the canisters in the second array will commence in the fifteenth year.

The canister site will require an area of about 55 metres by 130 metres. The site fence will enclose an area of about 80 metres by 155 metres.

SECURITY FACILITIES

The physical protection of the Point Lepreau N.G.S. site and, in particular, the Spent Fuel Storage Bay is based on a system of access alarms, closed-circuit television and motion detection.

The SFDS Program will require some modification to the fuel bay area, and the introduction of a fenced security area at the Waste Management site.

Prior to the loading of the first basket in the first canister, the canister site will be securely fenced in, with controlled personnel and vehicle access through a lockable gate. Security of the canister site will be by means of surveillance cameras and the associated lighting. The security fence gate will be equipped with intrusion alarms.

Safeguards

As a signatory of the Nuclear Non-Proliferation Treaty, Canada observes and abides by the international safeguards rules which are enforced by the International Atomic Energy Agency (IAEA).

The fundamental objective of safeguards procedures is to ensure that all spent fuel is properly accounted for and under the “constant” surveillance of the IAEA-installed device, or an IAEA inspector. To achieve this, each canister will be equipped with IAEA seals identical to the ones used on previous canisters. These seals are installed across the canister liner and the plug liner. Periodic verification of the inventory by the IAEA inspector provides a further check that diversion of sensitive material has not occurred.

To safeguard against such diversion, the Point Lepreau canister will be equipped with two IAEA reverification tubes. These are 50 mm (2 inch) schedule 40 pipes that run alongside the liner (just outside of the rebar) and exit, after a suitably radius bend, at the top of the liner, beside the plug, as seen in Figure 1.

During the transport of the fuel baskets to the canisters, the IAEA inspector will be present to provide continued surveillance.
CONCLUSION

The first generation of concrete canisters has been successfully used to store the spent fuel from the retired CANDU prototype stations (i.e. WR1, Gentilly-1, Douglas Point and NPD). To date, 463 MgU of fuel has been stored, in 81 canisters.

The second generation canister design, as described in the text, with a 60% higher payload, will be used at the Point Lepreau Station, starting in the spring of 1991.

The decision to implement this spent fuel dry storage program at Point Lepreau represents an important achievement in the ongoing commercial development of the CANDU technology.
Session 12:

Thermalhydraulics II

Chairman:

J.C. Amrouni, ENAQ
The TUF (Two-Unequal-Fluids) code has been developed by Ontario Hydro as a state-of-the-art two-fluid, systems thermal-hydraulics code for the analysis of transient behaviour in Ontario Hydro’s CANDU nuclear generating stations. The primary motivation for developing this code is to provide the capability to adequately represent phase separation effects that can occur during postulated accident transients and, thereby, to enhance Ontario Hydro’s nuclear safety analysis capability. The development of the TUF code has been an integral part of an ongoing generic study of the effectiveness of the Emergency Coolant Injection Systems (ECIS) in Ontario Hydro reactors during a large, critical-break Loss of Coolant Accident (LOCA).

An overview description of the TUF code is presented with particular emphasis on two-fluid thermal-hydraulic modelling considerations. The manner in which the various physical phenomena associated with phase separation, occurring in three-dimensional reactor geometries, are modelled within the constraints of a one-dimensional representation is discussed.

A key element in the development strategy for this code is a careful, systematic approach to verification and validation of the capability of the code to perform it’s required functions. This strategy, which includes benchmark test cases, separate effects tests and integral tests based on experimental loop data, is described in the paper. The capabilities of the code are summarized in the paper.

1.0 INTRODUCTION

Development of the TUF (Two-Unequal-Fluids) code was initiated at Nuclear Safety Department, Ontario Hydro. The TUF code is designed to be a flexible tool for analysis of CANDU reactor transients with the capability of modelling an extended range of physical phenomena relevant to postulated accident conditions. In particular, it provides a means for best-estimate simulations of LOCA transients including assessments of sustained phase separation conditions, ECI effectiveness and natural circulation.

Features incorporated in the TUF code include:

1. a two-fluid model which can be reduced to a one-fluid model,
2. an enhanced fuel channel model,
3. interactions between thermal-hydraulics and piping elasticity/plasticity for waterhammer analysis, and
4. modelling non-condensable gases in thermal-hydraulic analysis.

In the two-fluid model, there are more than three equations to be solved depending on the level of sophistication of the model. The following requirements have been considered during the development of this model:

1. conservation of total mass and energy,
2. reduction from two-fluid to one-fluid model,
3. computing efficiency and numerical stability, and
4. steady state results.

Consequently, the solution strategy of TUF code is different in the implementation of the two-fluid model from other advanced thermal-hydraulic codes [1-5].

The original impetus for developing the TUF code was an enhancement for the SOPHT system thermal-hydraulics code [6,7], which has served as Ontario Hydro’s primary system thermal-hydraulics code for safety analysis. Because of the large expenditure of resources to develop and maintain the SOPHT code and its widespread use within the corporation, a conscious decision was made to ensure that the TUF code would provide a capability for upward compatibility with the SOPHT code. This ensured that the large base of plant-specific modelling performed over the years for Ontario Hydro’s nuclear generating stations could be directly incorporated in the new two-fluid code.

In this paper, an overview of the TUF code is presented with particular emphasis on the two-fluid modelling. The main feature of the TUF code is outlined, and the code capabilities are summarized. Results of the selected simulations performed as part of a code verification program are presented to demonstrate some of the capabilities of the code.

2.0 GENERAL DESCRIPTION OF THE TUF CODE

Specific areas in which modelling development effort was directed to enhance analysis capability are briefly described below. Where appropriate, the enhancements made relative to the SOPHT code are identified.

Two-Fluid Model: An additional set of differential equations used to describe the unequal phase velocity and unequal phase temperature effects is included in the code. The effects of two-fluid model on the overall mixture conservation equations are in the following areas: convection transport terms, the source terms and the relationships among density, energy and pressure. In the two-fluid model, more constitutive equations are required than the one-fluid model. These constitutive equations, which describe wall and interfacial mass, momentum and energy transfers, dependent on the geometry of the flow regime present, and thus, models for the flow regimes are included. The two-fluid model can be simplified down to a one-fluid model, under user selected options.
Channel Model: A simplified regime map approach in the fuel pin modelling is applied in the TUF code. The wall surface is separated into two regions according to the void fraction of the vapour. The lumped wall model of a fuel pin is then applied in each region. The heat transfer between the pressure/calandria tube and the moderator has also been included in the TUF code. An average fuel pin is simulated in the SOPHT channel model.

Metal-Water Reaction: The metal-water reaction becomes a significant heat source when the sheath temperature exceeds 900 degrees Celsius. The metal-water reaction model applied employed in the TUF code is based upon models used in other safety analysis codes. The non-condensable hydrogen gas generated due to metal-water reaction is also simulated in the TUF code.

Phase Separation Capability: Phase separation in both horizontal and vertical piping is predicted through the flow regime map and the liquid carryover velocity, respectively, in the TUF code. Phase separation effects in the channel becomes important since they affect the heat transfer coefficient between coolant and the wall surface. Liquid droplet carryover velocity is calculated in the TUF code to determine whether or not phase separation occurs in vertical piping sections.

Multi-dimensional Effects: By averaging the three-dimensional governing equations over the cross-sectional flow area, the one-dimensional conservation equations are obtained. These area-averaged one-dimensional equations contain some distribution parameters or co-variant terms. These distribution parameters have been included in the TUF conservation equations based on a power law in the velocity profile. This distribution parameter has been included not only in the momentum-flux term but also in the break discharge model.

Flow Regime Map: In the two-fluid model, the distribution of each phase and the shape of the interface are needed to compute the interfacial transfers. Some of this information is obtained from flow regime maps.

Liquid and Bubble Entrainments: The slug, annular and stratified flow regimes may contain liquid droplets or bubbles. The entrainments of liquid droplets or bubble formation are important in the calculations of the interfacial area between two phases. In the TUF code, these effects have been included in those three flow regimes.

Fluid/Structure Interaction: In the analysis of waterhammer, two important factors should be taken into account: compressibility of the fluid and the deformation of the piping wall. When the pressure pulse is low, the assumption of a rigid pipe is applicable. However, the elasticity/plasticity of the piping wall becomes important in the simulation of waterhammer problems. The ability to directly compute fluid/structural interactions associated with transient piping deformation is incorporated in the TUF code.

Non-Condensable Gas: The non-condensable gas is included in the TUF code to simulate hydrogen production due to the metal-water reaction and accumulators.

In addition to the above differences in the code capability, other differences exist between TUF and SOPHT codes in the following areas:

Network Presentation: In the SOPHT code, the node/link concept (or non-central nodalization) of a module is applied. In this representation, solutions are depending on the link direction assigned. In the TUF code, both non-central and central nodalization are available.

Input Structure: The input data structure for the TUF and SOPHT codes are similar, except that the TUF code input data set is free format. A facility exists to semi-automate the conversion of a SOPHT input data file to a TUF input data file.

Steady State: In the SOPHT code, mixture energy equations are applied to the equivalent nodes. Consequently, solutions are only valid for the positive flows. In the TUF code, however, all nodal energy equations are solved, and hence solutions are independent of the link direction assigned. In addition to these mixture equations, the dynamic slip equation is also included in the steady state routine.

Heat Conduction Model: In the TUF code, the finite difference method in one region and the lumped wall models in two regions are applied in piping wall, pressure/calandria tubes, fuel pins and heat exchanger tubes. Heat transfer from feeders to atmosphere and from calandria tubes to the moderator are included in the TUF code. In the SOPHT code, the heat conduction models for the fuel pins and heat exchanger tubes are solved by the finite-difference method for one region.

Pressurizer Model: In the TUF code, the pressurizer is treated as a normal node where the two-fluid model is applied. This is in contrast to the SOPHT code which employs special component models.

Treatment of Water Packing: In the TUF code, when water packing occurs the piping volume is not allowed to expand, except through the expansion of piping elasticity. In the SOPHT code, however, the piping volume is allowed to expand to accommodate the extra mass.

Discharge Model: Both steady-state correlation and sonic discharge models are available in the TUF code whereas only the steady-state correlation is applied in the SOPHT code.
area and time averaged one-dimensional equations contain some space distribution parameters or co-variant terms to include the multi-dimensional contribution. The multi-dimensional contributions are also included in the source terms (transfer between two phases and between coolant and surrounding surface). The distribution parameters are calculated based on a power law in the velocity profile across the cross-sectional area in a potential flow.

The conservation equations for a two-phase mixture are deduced by combining those for the individual phases. The integrated form of mass and energy balance equations for the mixture over a control volume $V$ are given by

$$\frac{dM}{dt} = \sum_{in} W - \sum_{out} W$$

$$\frac{dU}{dt} = \sum_{in} (h_g W_g + h_f W_f + h_a W_a) - \sum_{out} (h_g W_g + h_f W_f + h_a W_a) + Q_w$$

where $h_g$, $h_f$, and $h_a$ denote vapour, liquid, and non-condensable gas respectively. The $M$ and $U$ are total mixture density and internal energy, respectively. $W$ is the mixture flow rate across the boundary of the control volume, $h$ is the specific enthalpy and $Q_w$ is the total heat transfer rate between coolant and wall surface.

By adding two phasic momentum equations, the mixture momentum equation is obtained as follows:

$$\frac{1}{A} \frac{dA}{dt} + C_o \frac{d}{ds} (W \nu + \phi g E_k^2) + \frac{dp}{ds} = E_v + E_f = -\tau_{vw} - \tau_{fw} - \rho g \frac{dz}{ds} + H_{\text{pump}}$$

where $A$ is the flow area, $C_o$ is the drag coefficient, $\nu$ is the mixture velocity, $W$ is the mixture flow rate and $E_k$ is the source term for phase $k$ containing wall shear stress $\tau_{vw}$, interfacial shear stress $\tau_{fw}$, gravity force, momentum transfer due to vapour generation and pump head. The source terms are defined by:

$$E_v = \Delta m v_{v1} + \tau_{v1} - \rho_v g \frac{dz}{ds} + \alpha_v H_{\text{pump}}$$

$$E_f = -\Delta m v_{f1} + \tau_{f1} - \rho_f g \frac{dz}{ds} + \alpha_f H_{\text{pump}}$$

and $\phi$ and $v_\nu$ are defined as,

$$\phi = \phi_g + \phi_a$$

$$v_\nu = v - v_f$$

$$\phi_g = \Lambda x_g x_f \rho$$

$$\phi_a = \Lambda x_a x_f \rho$$

where the subscript $v$ denotes the mixture of vapour and gas phases, and the subscript $i$ denotes interface. $\Delta m$ is the mass transfer rate per unit volume, $Z$ is the elevation, and $\rho$ is the density. The $x_v$ and $x_i$ are the void fraction and quality of phase $k$, respectively.

In the mixture conservation equations, the relationships between two-phase pressures, velocities and temperatures are required. In the homogeneous equilibrium (HEM or one-fluid) model, it is assumed that both phases have equal pressures, velocities and temperatures. Therefore, except for the constitutive equations between coolant and wall surface and the equations of state, no other dynamic equations are required to close the system of equations for the HEM model.

For the other thermal-hydraulic models, either correlations or dynamic equations are required to describe the relationships between two phases. The number of dynamic equations required depends on the level of sophistication of the thermal-hydraulic model applied.

The mass and energy balance equations for the vapour phase over a control volume $V$ are as follows:

$$\frac{dM_g}{dt} = \sum_{in} W_g - \sum_{out} W_g + \Gamma - \Gamma_{ga}$$

$$\frac{dU_g}{dt} + \frac{pg}{\rho} \frac{d\rho}{dt} = \sum_{in} h_g W_g - \sum_{out} h_g W_g$$

$$+ \Gamma h_{gs} + Q_{gp} + Q_{gl} + Q_{gw} + Q_{ga}$$

where $\Gamma$ is the vapour generation rate within the control volume; $\Gamma_{ga}$ is the non-condensable gas production rate from phase $k$; $h_k$ is the phasic specific enthalpy at a boundary, $h_{gs}$ is the saturated specific enthalpy of phase $k$; $Q_k$ is the heat transfer rate of phase $k$ due to pressure transient; $Q_i$ is the heat transfer rate from the interface $i$ to phase $k$; $Q_{gw}$ is the heat transfer rate from the wall to phase $k$ which only changes the phasic average energy level; and $Q_{ga}$ is the heat transfer rate from non-condensable gas to phase $k$.

The conservative equations for the non-condensable gas are written as follows:

$$\frac{dM_a}{dt} = \sum_{in} W_a - \sum_{out} W_a + \Gamma_{ga} + \Gamma_f a$$

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where $Q_w$ is the heat transfer rate from the wall surface to the non-condensable gas.

Since the mass fraction of non-condensable gas considered in the TUF code is generally small and also the relaxation time for mechanical equilibrium between vapour and non-condensable gas is generally shorter than that between vapour and liquid phase, it is assumed that non-condensable gas and vapour have the same velocities and pressures. Therefore, they can be treated as an indistinguishable phase in the momentum equations.

After rearranging the phasic momentum equations, the slip equation for the velocity difference can be written as

$$\frac{dV_g}{dt} + V_p \frac{dV_p}{dt} = \sum_{in} h_a w_a - \sum_{out} h_a w_a - Q_g - Q_{fa} + Q_{aw}$$


Equation (3) is solved for the flow rate.

**Drift-Flux Model (SUf)**

In this model, the velocity difference is obtained from the following equation:

$$V_{gf} = \frac{(C_{o-1}) W}{\rho_f} - V_{gf}$$

where $C_o$ is the distribution parameter, $V_{gf}$ is the slip velocity which is non-zero for a vertical flow, and $(\pm 1)$ sign depend on the flow direction, upward or downward. The constitutive correlations for $\tau_w$, $C_o$, and $V_{gf}$ are required.

**Dynamic Slip Model (UV)**

Equations (3) and (8) are solved for the mixture flow rate and velocity difference, respectively. The constitutive equations for $\tau_w$, $C_o$, $V_{gf}$, and $C_{vm}$ are required.

The effects of thermal non-equilibrium on the thermal-hydraulics are the mass and energy transfers between two phases, and the relationships among mass, energy, void fraction and pressure. The following temperature models have been included in the TUF code:

**Thermal Equilibrium (ET)**

In the thermal equilibrium model, vapour and liquid phases are assumed on the saturation line. Equations (1) and (2) are applied for the mixture. The constitutive equation for $Q_w$ is required.

**Simplified Thermal Non-Equilibrium (SUT)**

In the SUT model, vapour phase is assumed to be on the saturation line. The balance equations applied are Equations (1), (2), and (4). The constitutive equations for $Q_w$, $Q_{in}$, $Q_{in}$, $Q_{in}$, and $Q_{in}$ are required.

**Thermal Non-Equilibrium (UT)**

In the UT model, Equations (1), and (2) are solved for $M$ and $U$, and Equations (4) and (5) for $M_s$ and $U_s$. The constitutive equations for $Q_{in}$, $Q_{in}$, $Q_{in}$, and $Q_{in}$ are required.

Different combinations of the velocity and temperature models are available in the TUF code as user selected options.
Mass and Energy Transfer Modelling

The heat transfer rate between the saturated phases undergoing a pressure transient is modelled by the following equation:

\[ Q_{kp} = \beta_k \left[ M_k \frac{dV_{ks}}{dp} - V_{k} \right] \frac{dp}{dt} \]

where \( \beta_k \) is a boiling parameter which is to be determined empirically and has a value between 0 and 1. This parameter controls the pressure wave propagation velocity associated with the equation set used. The subscript s denotes the saturation line.

The following form of the interfacial heat transfer rate \( Q_k \) based on a conduction controlled heat transfer model is applied.

\[ Q_k = \nu \frac{A_{gf}}{h_k} (T_{ks} - T_k) \]

where \( A_{gf} \) is the interfacial heat transfer area per unit volume and \( h_k \) is the interfacial heat transfer coefficient for phase \( k \). It is noted that \( A_{gf} \) is strongly dependent on the flow regime.

The vapour generation rate \( \Gamma \) consists of three components: pressure transient \( \Gamma_p \), interface \( \Gamma_i \), and wall heat transfer term \( \Gamma_w \), or

\[ \Gamma = \Gamma_p + \Gamma_i + \Gamma_w \]

\[ \Gamma_p = -\left( Q_{g+} + Q_{fp} \right) \left( h_{gs} - h_{fs} \right) \]

\[ \Gamma_i = -\left( Q_{g1} + Q_{f1} \right) \left( h_{gs} - h_{fs} \right) \]

\[ \Gamma_w = \left( Q_{gw} + Q_{fw} \right) \left( h_{gs} - h_{fs} \right) \]

where \( Q_{fw} \) is the heat transfer rate from the wall to phase \( k \) which causes a phase change.

The constitutive correlations for the interphase momentum transfer consist of three parameters: virtual mass coefficient, interfacial velocities and interfacial shear stress. The general form of the interfacial shear stress \( \tau_{kl} \) is written as:

\[ \tau_{kl} = \left( \epsilon \right) \frac{A_{gf}}{\rho^i} \frac{f_s}{8} \left| v_{gf} \right| \left| v_{gf} \right| \]

where the negative sign is for gas (vapour plus air) and positive sign for liquid, \( \rho^i \) is the effective fluid density at interface and \( f_s \) is the interfacial Darcy friction coefficient. All these parameters are flow-regime dependent.

The flow regime maps applied in the TUF code are similar to those applied in the RELAP5 code with some modifications. The regimes are classified according to the mass flux and void fraction. The Kelvin-Helmholtz criterion was used to check the stability of stratification.

TUF has an efficient solution method for the two-fluid model. The method simplifies the resulting equations by substituting a linearized form of the mixture continuity and energy equations, and vapor continuity and energy equations into the mixture momentum equation to produce a set of \( N \) coupled equations that can be solved for the flow rate at each of the \( N \) junctions or links. The slip equation is solved implicitly with the mixture momentum equation.

The transport effects on solutions of the field equations strongly depend on the assigned link properties. The following link properties are required in the two-fluid field equations: phasic densities, qualities, void fractions and specific enthalpies. Usually, the link properties are assigned by using the properties at an upstream donor node with the donor node dynamically assigned depending upon the flow direction.

4.0 CODE ASSESSMENTS AND APPLICATIONS

An integral part of the development of the TUF code is a systematic code verification program. The objective of this program is to systematically verify the adequacy of the code to represent the physical phenomena governing thermal-hydraulic behaviour in Ontario Hydro's nuclear reactors. This program involves the progressive use of benchmark tests and experimental data from separate effect and integrated tests.

4.1 CALANDRIA TUBE INTEGRITY PROGRAM

The full-scale pressure rupture experiments performed as part of the Calandria Tube Integrity Program [8] are utilized to verify features of the TUF code, including the implicit strain feedback model which accounts for both the elastic and plastic deformation of the piping components and the non-condensable gas model.

The experimental test rig consists of a full-scale Bruce Unit 2 fuel channel which is connected via inlet and outlet feeder pipes to a large reservoir capable of maintaining pressure for several seconds following pressure tube failure. A schematic of the experimental loop, including flow and pressure control system is shown in Figure 1.

FIGURE 1: Experimental Loop Schematic
The TUF node-link diagram for the test rig is shown in Figure 2. The pressure tube as well as the calandria are represented by 13 nodes each, with each node corresponding to a fuel bundle in the channel. The neighbouring nodes of the pressure tube and the calandria tube are connected by "break discharge valve" links, as indicated in Figure 2. This allows for an accurate representation of the crack length, area and the propagation speed of the crack by opening the valve links representing the crack at times which match the measured crack propagation. The flow path through both the inboard and outboard bearings clearances are represented by valve links of equivalent flow area and hydraulic diameter. The bearings clearance is an important factor in both the experiment and the model, because if primarily restricts the flow from the annulus to the bellows and thus determines how rapidly the annulus void collapses. This in turn determines the pressurization rate and waterhammer loading on the calandria tube.

A total of four tests have been simulated, of which the results from two (Test #5 and #6) are presented. The TUF code simulations are presented for both a rigid and an elastic calandria tube. Each case is run once with the annulus filled with H₂O vapour and then the vapour is replaced with air, which is actually present in the experiments, to investigate the role of the non-condensable gas. The experimental data for Tests #5 and #6 are given in Table 1. The main difference between these two tests is the calandria tube material and thickness.

The pressure transient in the annulus near the crack (Node 27) and bellows for Test #5 for a rigid and elastic wall assumption are shown in Figure 3. In this simulation the annulus is assumed to be vapour filled. Figure 3 clearly demonstrates the waterhammer pressure and the effect of the calandria tube deformation in both reducing the peak pressure in the annulus from about 17 MPa to about 11.5 MPa and damping out the pressure pulses.

If the assumed vapour in the annulus is replaced with air (Test #5) to account for the non-condensable gas then a significant damping effect on the pressure wave is observed as shown in Figure 4. The damping effect is due to the additional compressibility associated with the gas which prevents complete collapse of vapour in the annulus. Figure 4 shows that the pressure pulse peak is attenuated from 17 MPa to 10.5 MPa due to the non-condensable gas alone, i.e. calandria tube is assumed rigid.

| Table 1: Experimental Data for Selected Tests from Pressure Tube Rupture Program |
|------------------|---|---|
| Test No. | 5 | 6 |
| Pressure [MPa] | 7.5 | 8.5 |
| Temperature [°C] | 255 | 256 |
| Subcooling [°C] | 35 | 35 |
| Calandria Tube Material | Stainless Steel | Zircaloy Zr-2 |
| CT Thickness [mm] | 2.5 | 1.37 |
| Crack Length [m] | 2.7 | 2.7 |
| Crack Velocity [m/s] | 400 | 400 |
| Average hoop strain in the Calandria % | 0.12 | 0.75 |
Test #6 has a calandria tube thickness of 1.37 mm, which is typical of Bruce and Darlington NGS. The pressure transient in the annulus near the crack (Node 27) and bellows, for Test #6, for rigid and elastic wall assumptions are shown in Figure 5. The attenuation of the pressure pulse from 21 MPa to 10 MPa due to the calandria tube deformation is clearly shown in Figure 5.

Replacing the vapour in the annulus with air for Test #6, to model the non-condensable gas present in the experiment, results in a similar trend to what is observed in Test #5, as shown in Figure 6. The pressure pulse peak is attenuated from 21 MPa to 12 MPa due to the non-condensable gas alone. This highlights the importance of the existence of non-condensable gas in mitigating the waterhammer type of loading on any piping component even without deformation occurring. By including the piping elasticity, the pressure in the annulus is further reduced, due to the deformation of the calandria tube which takes place at about 8.5 MPa.

### 4.2 NUCLEAR POWER DEMONSTRATION PRESSURIZER EXPERIMENT

Recently, the pressurizer experiments have been conducted at NPD (Nuclear Power Demonstration GS) in 1985 [9]. These experiments provide excellent cases to examine the separate effect of thermal non-equilibrium on the pressure transient. A schematic drawing of the pressurizer vessel at NPD is shown in Figure 7. The case of an insurge experiment (No. 11) is presented here. The insurge liquid temperature is about 155 °C and the insurge flow rate starts at time 15 seconds and stops at 150 seconds (see Figure 7). This case represents the direct contact condensation of stagnant steam on slowly moving subcooled water. Three different models (adiabatic, equilibrium and non-equilibrium) have been applied. In the thermal non-equilibrium model, the interfacial heat transfer coefficient for the liquid phase for a stagnant condensation is about 0.3 kW/m² °C. Comparison with experimental result for the pressure transient is shown in Figure 8. Agreement is excellent for the thermal non-equilibrium model. The corresponding fluid temperatures are shown in Figure 9.
Three standard problems were selected from Reference [10] as benchmark tests for verification of the TUF code numerical solution scheme. The first problem deals with the instantaneous heat addition to flowing subcooled water at high pressure in a vertical pipe. The second problem involves a step injection of subcooled water into a pipe containing a vertical flow of slightly superheated steam at high pressure. The third problem is essentially the same as problem 2 except that the injection water is highly subcooled and the steam is at a much lower pressure. Benchmark solutions were obtained by MECA, [11] a method of characteristic wave tracing. Results of the simulations show good agreement with the MECA predictions. Particularly, the TUF code predictions exhibit very little numerical diffusion. Detailed comparisons with the MECA predictions are provided in Reference [12].
4.4 RD-14 THERMOSYPHONING AND BLOWDOWN TESTS

Large scale integral system verification, with the TUF code, was also undertaken. A series of tests performed at the RD-14 test facility at AECL-WNRE, provided excellent experimental data for TUF verification. RD-14 is a scaled representation of a typical CANDU reactor with similar nominal operating conditions.

One partial-inventory thermosyphoning and two critical inlet header break experiments were selected for simulation by the TUF code. Full system nodalization including primary, secondary and EC1 systems were modelled. A more detailed description of the experiments and the simulations is given in the companion paper to be presented at this conference [13].

ACKNOWLEDGMENTS

The Calandria Tube Integrity Program and RD-14 Experiments are funded by the CANDU Owners Group (COG).

REFERENCES


ANALYSIS OF LARGE BREAK LOCA EXPERIMENTS IN THE RD-14 TEST FACILITY

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ABSTRACT

Selected large break loss-of-coolant experiments, conducted in the RD-14 test facility, are analysed in this paper. The objective was to obtain a clear physical understanding of the thermalhydraulic mechanisms leading to, and the consequences of, periods of flow stagnation or low flow in the heated sections. The blowdown and refill phenomena, fuel element simulator sheath temperatures, fluid temperatures and void fractions at the inlet and outlet of heated sections, observed during these experiments, are described in detail. It is shown that the flow behaviour in the loop can be better understood using diagrams displaying the pressure distribution around the loop at various times.

INTRODUCTION

Many loss-of-coolant (LOCA) experiments have been performed in the RD-14 test facility (1) to improve the understanding of the CANDU primary heat transport system and safety systems under postulated accidental conditions. In this paper, measured parameters during selected large break LOCA experiments have been analysed. The principal parameters varied in these experiments were: the type of pump operation (pumps rared or tripped), and the operation of low-pressure Emergency Coolant Injection (ECI) system (on or off).

Brief Description of Experimental Facility

The RD-14 thermalhydraulic test facility is a full-elevation representation of a CANDU primary heat transport system (1). A simplified flow diagram of the RD-14 facility is shown on Figure 1. The facility is designed to produce the same fluid mass flux, transit time, pressure and enthalpy distributions in the primary system as those in a typical CANDU reactor. The most important parameters of RD-14 are compared with those of a typical CANDU reactor in Table 1.

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>RD-14</th>
<th>CANDU Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operat. press. (MPa)</td>
<td>10</td>
<td>10</td>
</tr>
<tr>
<td>Loop volume (m³)</td>
<td>0.9514</td>
<td>57.0</td>
</tr>
<tr>
<td>Loop piping I.D. (m)</td>
<td>0.074</td>
<td>varies</td>
</tr>
<tr>
<td>Heated sections:</td>
<td>Indir. heat. 37-rod bndl</td>
<td>Nuclear fuel 37-elem. bndl</td>
</tr>
<tr>
<td>Length (m)</td>
<td>6</td>
<td>12 x 0.5</td>
</tr>
<tr>
<td>Rod diameter (m)</td>
<td>0.0131</td>
<td>0.0131</td>
</tr>
<tr>
<td>Flow tube diam.(m)</td>
<td>0.1034</td>
<td>0.1034</td>
</tr>
<tr>
<td>Power (kW/channel)</td>
<td>5550</td>
<td>5410</td>
</tr>
<tr>
<td>Pumps:</td>
<td>Single stage</td>
<td>Single Stage</td>
</tr>
<tr>
<td>Impeller diam. (m)</td>
<td>0.381</td>
<td>0.813</td>
</tr>
<tr>
<td>Rated flow (kg/s)</td>
<td>24</td>
<td>24</td>
</tr>
<tr>
<td>Rated head (m)</td>
<td>224</td>
<td>215</td>
</tr>
<tr>
<td>Specific speed</td>
<td>565</td>
<td>2000</td>
</tr>
<tr>
<td>Steam generators:</td>
<td>Recirculat. U-tube</td>
<td>Recirculat. U-tube</td>
</tr>
<tr>
<td>Number of tubes</td>
<td>44</td>
<td>37/channel</td>
</tr>
<tr>
<td>Tube diam. I.D.(m)</td>
<td>0.0136</td>
<td>0.01475</td>
</tr>
<tr>
<td>Secondary heat-trans. area (m²)</td>
<td>41</td>
<td>32.9/ch.</td>
</tr>
<tr>
<td>Heated section-to-boiler top elev. difference (m)</td>
<td>21.9</td>
<td>21.9</td>
</tr>
</tbody>
</table>

As shown in Figure 1, RD-14 consists of two full-scale (6-m long), full-power (maximum of 5.5 MW) horizontal channels, representing reactor fuel channels. Each channel contains 37 electrically heated Fuel Element Simulators (FES), which have uniform heat flux distribution and have almost the same heat capacity as reactor fuel. Primary fluid circulation is provided by two high-head, single-stage, bottom-suction centrifugal pumps. Typical pressure distribution in the RD-14 loop during normal operation is shown on Figure 2.

Heat is removed from the primary circuit through two full-height U-tube-type steam generators equipped with internal preheaters, spiral arm steam separators and external downcomers. The heated sections, steam generators, headers and feeders are arranged to give a full-elevation representation of a typical CANDU primary heat transport loop.
The break simulation system consists of a fast-acting 6-inch (nominal) ball valve, which may be connected to an inlet or an outlet header. A restricting orifice is placed immediately upstream of the valve to simulate the desired break size. For the experiments described in this paper, 48-mm and 52-mm orifices were used to simulate a large break of an inlet or outlet header respectively. The ECI system consists of a nitrogen-pressurized tank for high-pressure injection, and a tank at atmospheric pressure and pump for low-pressure injection. Coolant supplied by either ECI system is directed to all four headers.

Test Conditions and Experimental Procedures

Before each experiment, the RD-14 loop was evacuated, filled with distilled water and degassed; final instrument calibrations were then completed. The loop was then brought to conditions of stable single-phase forced flow at typical full-power conditions for a CANDU reactor. Recording of data was initiated, the surge tank was isolated from the heat transport loop, and a fast-acting valve opened to simulate the break. The channel power was decreased to typical decay power levels 2 s after the break opened. A programmable pump speed controller was used in some experiments to simulate exponential pump rundown following a loss-of-class IV power. Emergency coolant was injected into the loop when the primary pressure fell to or below the ECI pressure.

The test matrix for the large break experiments is shown in Table 2, including information about the type of pump and low-pressure ECI system operation, as well as about the time of each pump trip, and the time of power trip, as it was considered to be of major importance for the loop behavior. The initial conditions applied to all experiments in this series are listed in Table 3, while the test procedure used in these experiments is given in Table 4.

TABLE 2
TEST CONDITIONS FOR RD-14
LARGE BREAK EXPERIMENTS

<table>
<thead>
<tr>
<th>TEST NUM.</th>
<th>BREAK LOC.</th>
<th>POWER TRIP</th>
<th>ACC. INJ.</th>
<th>PUMP 1 RAMP</th>
<th>PUMP 1 TRIP</th>
<th>PUMP 2 TRIP</th>
</tr>
</thead>
<tbody>
<tr>
<td>BB607</td>
<td>INLET</td>
<td>230 s</td>
<td>yes</td>
<td>none</td>
<td>linear</td>
<td>no</td>
</tr>
<tr>
<td>BB608</td>
<td>INLET</td>
<td>168 s</td>
<td>no</td>
<td>none</td>
<td>none</td>
<td>33 s</td>
</tr>
<tr>
<td>BB609</td>
<td>INLET</td>
<td>240 s</td>
<td>yes</td>
<td>none</td>
<td>40 s</td>
<td>no</td>
</tr>
<tr>
<td>BB610</td>
<td>INLET</td>
<td>225 s</td>
<td>yes</td>
<td>none</td>
<td>40 s</td>
<td>no</td>
</tr>
<tr>
<td>BB703</td>
<td>INLET</td>
<td>170 s</td>
<td>yes</td>
<td>expon.</td>
<td>no</td>
<td>no</td>
</tr>
<tr>
<td>BB704</td>
<td>INLET</td>
<td>190 s</td>
<td>yes</td>
<td>yes</td>
<td>48 s</td>
<td>20 s</td>
</tr>
<tr>
<td>BB611</td>
<td>OUTL.</td>
<td>20 s</td>
<td>yes</td>
<td>none</td>
<td>linear</td>
<td>20 s</td>
</tr>
<tr>
<td>BB716</td>
<td>OUTL.</td>
<td>no</td>
<td>yes</td>
<td>expon.</td>
<td>no</td>
<td>16 s</td>
</tr>
</tbody>
</table>

TABLE 3
NOMINAL INITIAL CONDITIONS

| Primary System: | outlet header pressure - 10 MPa(g) |
| Secondary System: | input power - 5.0 MW per heated sec. flow rate - 27 L/s |
| Fluid: | steam drum pressure - 4.5 MPa(g) feedwater temperature - 187°C |
| Cold Water: | high pressure - 5.5 MPa(g) low pressure - 1.5 MPa(g) |
| Break: | inlet header - 48 mm outlet header - 52 mm |
| Pump Ramp: | 100 s (linear) |
| Down Time: | 180 s (exponential) |

TABLE 4
TEST PROCEDURE

| t = 0 s: | Start data gathering |
| t = 6 s: | Isolate surge tank |
| t = 10 s: | Open break valve |
| t = 12 s: | Step power to decay level, start pump speed ramp |
| P = 5.5 MPa: | Start high pressure ECI |
| Accumulator Empty: | Stop high-pressure ECI |
| Tank Empty: | Start low-pressure ECI |
| Conditions stabilized or process trip: | Stop gathering data |

ANALYSIS OF RESULTS

Inlet-Header Break Experiments

A series of six inlet-header, large break experiments was conducted in RD-14, as shown in Table 2. The break was located at header 4. Typical pressure distribution (based on measured gauge pressures) in the loop at selected times during an inlet-header break experiment is shown in Figure 3.

Loop Pressure Behaviour. After the break valve opened at header 4, vigorous depressurization of the loop began, along with rapid coolant discharging and flashing at the break. The pressure difference between headers 4 and 1 started to decrease, as pressure at header 4 was decreasing more rapidly than in header 1.
Because of pressure reduction in the loop, the negative pressure difference between header 4 and header 1 also decreased with time. In addition, the ECI tank was depleted at approximately 85 s in the tests. All this caused significant reduction of the reverse flow rate through the downstream heated section after 90 s in the tests, resulting in refilling of heated section 1 from outlet to inlet, regardless of low-pressure ECI being active or not.

During the early period of the experiments, as long as the pump 1 head was not degraded, the pressure difference across the heated section upstream of the break (heated section 2) was not significantly reduced, indicating forward flow, as seen from Figure 3. The broken header 4 was the higher pressurized header, located close to the pump 2 outlet, causing rapid degradation of pump 2 head, and reversal into a high negative head. Pump 1 head was degraded because of voiding (usually by 35 s in the test), or, because of reduced speed by the imposed ramp down. Thus, the combination of low pump 1 head and high negative pump 2 head caused flow stagnation or periods of low flow in the heated section 2, especially after 50 s in the tests. Because of this, except for brief periods of refilling, heated section 2 was voided after 50 s in the experiments.

Differential Pressures Across Heated Sections.

Figures 4, 5 and 6 show pressure difference across both heated sections for three experiments: B8607, B8703 and B8704. As shown in Table 2, the selected experiments were carried out with linear, exponential, and no pump ramp down, respectively. Generally, comparing Figures 4, 5 and 6, little difference is observed in differential pressure across both heated sections. Therefore, it can be readily concluded that in case of inlet-header break experiments, pump operation had little influence on the heated section pressure difference.
Pump trip was closely associated with pump ramp. In all large inlet-header break experiments, pump trip because of voiding and overvoltage of the pump motor, never occurred when pumps were ramped down, as their speed and torque significantly reduced before the voiding occurred. On the other hand, pump 1 (located far from the break) tripped between 30 and 40 s in the test whenever no pump ramp was imposed.

Regardless of being ramped or not, pump 2 never tripped, except in the experiment B8704. The flow through pump 2 was always in forward direction, as the break was located at the pump delivery side. It acted as a flow restriction in the loop, obstructing coolant discharge from heated section 2 and steam generator 2 towards the break. Therefore, high negative head was established at pump 2, regardless of mode of pump 2 operation (ramped or tripped).

Comparing Figures 4, 5 and 6, it can be concluded that the negative pressure difference across heated section 1 was higher in the test with linear pump ramp (B8607), than in the tests with exponential ramp (B8703) and no ramp (B8704), owing to higher pump 2 head.

By 20 s in the tests, the pressure difference across heated section 2 was significantly reduced. After 50 s in the tests, regardless of pump operation, the pressure difference across heated section 2 became negligibly low, indicating flow stagnation or very low flow condition.

Heated Section Temperatures. Figures 7 and 8 show the inlet top, centre and bottom FES sheath temperatures for experiments B8607 and B8703. Note that, sheath temperatures at the inlet of heated section 1 were higher than at the outlet. A typical sharp increase of inlet sheath temperatures up to 500°C during early flow stagnation (between 10 and 25 s) was clearly observed. It was caused by flashing and voiding of heated section 1 at this time.

After heated section 1 was refilled, and FES were quenched, the inlet sheath temperatures were reduced to saturation temperature. Later, in the period between 60 and 100 s, sheath temperatures were significantly below saturation temperature.

As soon as the ECI tank for high-pressure injection was depleted (at 85 s), and heated section 1 revoided, the inlet sheath temperatures increased to saturation. When low-pressure ECI was not active, like in the test B8607, the inlet top and centre sheath temperatures continued to increase significantly above saturation temperature after 120 s. In the test B8703 the low-pressure ECI system was initiated as soon as the ECI tank for high-pressure ECI was depleted. Therefore, the heated section 1 inlet sheath temperatures were close to saturation temperature after 100 s in the test.

From Figures 7 and 8, it is interesting to note that during early flow stagnation the bottom FES sheath was much shorter and less superheated than the centre and top sheath, indicating void stratification at the heated section 1 inlet. Also, after 120 s in test B8607, the inlet bottom FES sheath temperature followed saturation temperature, while the inlet centre and top FES sheath temperatures started to rise, indicating void stratification at the inlet.

Figure 9 shows the inlet and outlet top FES sheath temperatures at heated section 2 (upstream of the break). Owing to the prolonged period of low-flow condition after 50 s in the test (low-pressure difference shown on Figure 5), i.e., insufficient cooling, both inlet and outlet top FES sheath temperatures...
sharply increased above saturation temperature, regardless of high- and low-pressure ECI systems being active, or the mode of pump operation. Therefore, in all large inlet-header break tests, the tests were terminated when sheath temperature in heated section 2 reached the set value of 600 or 700°C.

Outlet-Header Break Experiments

Two outlet-header large break experiments are analysed in this paper: experiment B8611 (break size of 52 mm and decay power level of 385 kW per pass) and experiment B8716 (break size of 50 mm and decay power level of 200 kW per pass). The downstream pump (pump 2) behaved differently in these two experiments, resulting in completely different flow situation in the downstream heated section (heated section 2).

Loop Pressure Behaviour. Pressure distributions (based on measured gauge pressures) in the RD-1A loop at selected times during the outlet-header large break experiments B8611 and B8716 are shown on Figures 10 and 11. It is important to note that in case of outlet-header break experiments (break located at header 3), unlike in the inlet-header break experiments, reverse flow was established through pump 2.

As seen from Figure 10, in case of an outlet-header break, as soon as the break valve opened, pressure at header 3 started to decrease, thus increasing the pressure difference across the heated section upstream of the break. At approximately 15 s in the experiment B8611, the combination of high pump 1 head and reduced pressure at header 3 resulted in increased pressure drop across heated section 2, as seen from Figure 12, indicating accelerated coolant discharge flow toward the break through most of the test. Therefore, after the high-pressure ECI was initiated into header 2, heated section 2 quickly refilled, and stayed filled with liquid coolant through most of the test.
FIGURE 9: HEATED SECTION 2 TOP SHEATH TEMPERATURES FOR EXPERIMENT B8703

FIGURE 10: PRESSURE DISTRIBUTION IN THE RD-14 LOOP FOR OUTLET-HEADER BREAK EXPERIMENT B8611

It is evident from Figure 10 that at the same time as pump 2 downstream of the break did not trip in the test B8611, it kept operating with high positive head, thus obstructing the reverse flow toward the break. Therefore, as a combination of decreased pressure at broken header 3 and high positive head of pump 2, the pressure difference across the downstream heated section decreased to a negligibly low value, and stayed

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low throughout experiment B8611, indicating prolonged flow stagnation or low flow.

Qualitatively different flow situation in the downstream heated section was observed when pump 2 downstream of the break tripped due to voiding, as occurred in experiment B8716. In that case, pump 2 head degraded quickly to a very low positive value, as soon as the pump tripped (at 16 s). Note that in tripped condition, the differential pressure across pump 2 was very high in forward flow (Figure 3), whereas it was negligibly low in reverse flow (Figure 11), resulting from different flow rate through pump 2, and perhaps, from different pump 2 flow resistance in the two cases.

Comparing Figure 10 with Figure 11 at 25 s in the test, it is evident that significant negative pressure difference was established across heated section 1 in the test B8716, indicating that it was refilled in reverse flow. At the same time, the pressure difference across the upstream heated section was not changed. It may be concluded that in the case of outlet-header break tests, pump 2 trip improved the cooling of the heated section 1.

Differential Pressures Across Heated Sections. Figures 12 and 13 show pressure drops across both heated sections in the analysed outlet-header break experiments B8611 and B8716, respectively.

When the downstream pump 2 tripped, the pressure difference across the downstream heated section 1 was negative through most of the test, indicating that it was refilled and well cooled in reverse flow until the high-pressure ECI tank was depleted. Very low pressure difference across heated section 1 in the test B8611 (pump 2 not tripped), indicates a condition of low flow rate and insufficient cooling.

No significant change in the pressure difference across upstream heated section 2 was observed between the two tests. It significantly increased after the break opened. After 50 s, it decreased to approximately 1/3 of the nominal value, and remained until the tank for high-pressure ECI was depleted.

Heated Section Temperatures. Figures 14 and 15 show selected heated section 1 sheath temperatures for the outlet-header large break experiments B8611 and B8716, respectively. Note that in the experiment B8611 power was tripped at 20 s, as soon as heated section 1 outlet temperature reached 350°C. Nevertheless, the top FES sheath temperatures in the downstream heated section 1 peaked early in the test (17 s) up to 490°C, and then started to slowly decrease (a few seconds before the power was set off), but remained significantly above saturation temperature (especially at the outlet) until the end of the test, owing to very low flow and voiding of heated section 1 outlet.

The sheath temperatures at the upstream heated section were close to saturation temperature until 50 s in both outlet-header experiments. After that, they were significantly below saturation temperature until the end of experiments. Hence, it may be concluded that the upstream heated section was well cooled in both experiments.

Figure 15 shows that the downstream heated section 1 top FES sheath temperatures also peaked during early flow stagnation at approximately 450°C. As soon as the pump 2 tripped and reversed flow was established through heated section 1, however, the top FES were quenched, and top sheath temperatures decreased to saturation temperature. The outlet top FES sheath temperature remained close to saturation temperature until the end of the test B8716, whereas the inlet top FES sheath temperature increased above saturation temperature after 100 s, after the high-pressure ECI tank was depleted. However, by 200 s it levelled at approximately 320°C, close to the nominal value.
SUMMARY AND CONCLUSIONS

The RD-14 thermalhydraulic test facility and large break LOCA simulation tests have been described briefly. Selected experimental results for six inlet-header and two outlet-header large break experiments were analysed. The general flow behaviour and the key mechanisms governing the loop flow behaviour were described. The analysis was based on measured parameters such as pressure drops across loop components, void fractions and fluid temperatures at various points, as well as sheath temperatures at various fuel element simulators.

In the inlet header break experiments, the brief flow stagnation in the heated section downstream of the break was caused by high negative head of the pump located close to the break, and by loss of head at the pump far from the break. Prolonged periods of liquid coolant deficiency were observed in this pass, and the ECI system could not quench the heated section upstream of the break before FES sheath temperatures of 600 or 700°C were reached, terminating the experiments.

The mode of pump operation (ramped or not, or, tripped or not) in the inlet-header break experiments had very little effect on the flow situation in both heated sections.

In the outlet-header break experiments, flow stagnation in the heated section downstream of the break was caused by decreased pressure at the broken outlet header and high positive head of the downstream pump (when it did not trip). If this pump tripped, however, the pressure drop across this pump degraded, and reverse flow was established. In this case the ECI system readily refilled the downstream heated section, and it was well cooled throughout the rest of the experiments.

In the outlet-header break experiments, increased pressure difference was observed across the heated section upstream of the break, maintaining sufficient forward flow and good cooling, owing to decreased pressure at the broken outlet header 3.

As expected, the location of the break in the loop (i.e., the location of the most rapid depressurization) was found to be very significant. The break location determined the flow direction in the loop after the break valve opened. It also influenced the rate of loop depressurization and the pump operation.

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REFERENCES

METHODOLOGY FOR ASSESSING FIGURE OF EIGHT TYPE FLOW OSCILLATIONS IN CANDU PRIMARY HEAT TRANSPORT SYSTEMS

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Ontario Hydro, 700 University Avenue, Toronto, Ontario M5G 1X6

ABSTRACT

This work provides a methodology to assess the effects of flow oscillations during forced convective cooling. Oscillations are initiated when power is increased, or heat transport pressure is decreased until significant void develops. The model described here is applied to a loss of heat transport pressure control, for predicting the condition when oscillations begin and for predicting the maximum oscillation amplitude.

The different regions of interest are shown schematically in Figure 1. The loop is assumed to consist of two symmetrical passes. Each pass is divided into eight regions representing the inlet feeders, reactor channels, outlet feeders and the steam generators. The divisions also account for the horizontal or vertical orientations of the outlet feeder section and the presence of single or two phase coolant in the sections.

In each of these sections, a transmission matrix is formulated to characterize how disturbances in void fraction, pressure and enthalpy at the section inlet are propagated through the section. A combination of these individual section matrices give the system characteristic matrix. The stability of the system can then be determined analytically by deriving the roots of the determinant of this characteristic matrix, or graphically from the the NYQUIST plot of the determinant.

1.0 INTRODUCTION

The phenomenon of thermally-induced, two phase flow instabilities is of interest for the design and operation of industrial systems and equipment such as steam generators, thermosyphon reboilers, and in particular the primary heat transport systems of CANDU nuclear power plants. In the latter, oscillations of flow and system pressure may be undesirable as they can cause mechanical vibration, problem in system control and in extreme circumstances can lead to reactor trip by the automatic shutdown systems. The heat transport oscillations directly affect the time at which the reactor is tripped following a transient, and the fuel cooling conditions before reactor trip occurs.

A linear model for two-phase flow instabilities in a CANDU type of primary heat transport system (PHTS) has been developed and verified for the natural circulation cooling mode(1,2). The drift flux form of conservation equations(3) and the concept of transmission matrices(4) were used to predict the stability characteristics of two phase natural circulation flow. In this study, the model is modified to apply to cases where the heat transport pumps provide forced convective cooling and an increase in reactor power or a decrease in system pressure initiates the flow oscillations. The model is used to determine the thermal hydraulic conditions at the onset of flow oscillations. However, the model cannot predict the oscillation amplitude per se because of the non-linear nature of the limit cycle analysis. A separate model was then employed using a void collapse criterion and the appropriate phase shift between the different thermal hydraulic parameters, to determine the maximum oscillation amplitude.

2.0 MODEL DESCRIPTION

This section describes the linear model used to analyze CANDU heat transport system flow stability characteristics and to predict the necessary conditions for flow oscillations under forced cooling conditions. The linear model is formulated using the drift flux form of the conservation equations and the concept of transmission matrices.

Figure 1 : Schematic diagram of the figure of eight loop
2.1 Governing Equations

The governing equations used in the formulation are the one-dimensional drift flux form of the mixture mass, momentum, energy conservation equations, the vapour phase continuity equation (void propagation equation) and the equation of state. These general equations are further reduced to make analytical solutions feasible. It is assumed that the co-variance terms (arising from volume averaging) in all conservation equations are negligible. Momentum and enthalpy diffusion due to the relative motion between phases are neglected. It is further assumed that the effect of the rate of change of liquid phase and vapour phase density is small compared to the rate of change in void fraction, i.e., \( \frac{\partial \rho_g}{\partial t}, \frac{\partial \rho_f}{\partial t} \ll \frac{\partial a}{\partial t} \). (For the conditions at the onset of flow oscillation, the system pressure can be assumed to oscillate with small amplitude, thus the saturated density at that pressure does not change significantly).

With these assumptions, the one-dimensional governing equations are written as follows:

(a) Mixture continuity equation (1)
\[
\frac{\partial \rho}{\partial t} + \frac{\partial j}{\partial z} = 0
\]

(b) Vapour phase continuity equation (2)
\[
\frac{\partial \rho_v}{\partial t} + \frac{\partial j_v}{\partial z} = \frac{r_g}{\rho_g}
\]

(c) Mixture momentum equation (3)
\[
\frac{\partial j}{\partial t} + \frac{\partial}{\partial z} \left( \frac{j^2}{\rho_m} \right) = \frac{-\rho_m g \sin \varphi - f_m \frac{G^2}{4 \rho_m}}{\rho_m}
\]

(d) Mixture enthalpy equation (4)
\[
\frac{\partial h_m}{\partial t} + \frac{\partial}{\partial z} \left( \frac{h_m j}{\rho_m} \right) = g \frac{\xi}{A}
\]

These are used along with:

(a) Definition of vapour phase velocity (5)
\[
v_g = \frac{C_0 G/\rho_f + V_g}{\rho_f [1 - \delta \rho C_0 a_g/\rho_f]}
\]

(b) Definition of drift flux velocity
\[
v_{gj} = v_g - a_G
\]

where \( j \) is the volumetric flux. For the case of one dimensional flow of a two-phase mixture with a change of phase, the gradient of \( j \) is expressed as

\[
\frac{\partial j}{\partial t} = \frac{\partial \rho_g}{\partial t} - \frac{\partial \rho_f}{\partial t} \left( a_G \frac{\partial \rho_f}{\partial t} + \frac{\partial \rho_g}{\partial t} \right)
\]

where the first term on the right hand side of (7) denotes the vapour generation/suppression effect and the second term denotes the effect of compressibility of the vapour phase.

Equations (1) to (7) are transformed into dimensionless form with the following dimensionless variables:

\[
u_g = \frac{V_g}{V_{ref}}; C_0 = \frac{v_g}{V_{ref}} C_0; \quad \delta = \frac{t/t_{ref}}{L}; \quad \xi = \frac{z/L}{P} = \frac{V_{ref}/t_{ref}}{L}
\]

\[
\alpha = \alpha_g C_0 \frac{\partial \rho_f}{\partial t}; \quad \omega = \frac{G/\rho_f}{V_{ref}}
\]

\[
\rho = \frac{\rho_m/\rho_f}{\rho_g + (1-\alpha_g) \rho_f/\rho_f}
\]

and the following dimensionless groupings:

Phase change number
\[
Q_2 = \frac{T_g - \Delta P L}{C_0 \rho_f}
\]

Heater inlet subcooled number
\[
J_a = \frac{(h_f - h_{fg})}{\rho_f V_{fg} h_{fg}}
\]

Heat flux number
\[
Q_1 = 2 \frac{q_{in}}{Lc/R h_f}\rho_f V_{ref}
\]

Froude number
\[
F_r = \frac{2 \rho L}{D_T}
\]

The dimensionless governing equations are

(a) Mixture continuity equation (9)
\[
\frac{\partial \alpha}{\partial \theta} - \frac{\partial \omega}{\partial \xi} = 0
\]

(b) Void propagation equation (10)
\[
\frac{\partial \alpha}{\partial \theta} + C_0 \frac{\partial \alpha}{\partial \xi} = Q_2 (1-\alpha) + \frac{\alpha^2 \rho_f}{\rho_g} \frac{\partial \rho_g}{\partial \xi} + C_0 \frac{\partial \rho_g}{\partial \xi}
\]

where the first term on the right hand side of (10) denotes the vapour generation/suppression effect and the second term denotes the effect of compressibility of the vapour phase.
(c) Mixture momentum equation (11)
\[ \frac{\partial \mathbf{w}}{\partial t} + \nabla \cdot (\mathbf{w}\rho) = -\nabla \cdot (\mathbf{p} \mathbf{v}) - \rho \mathbf{g} + \mathbf{f} \wedge \mathbf{w} \]

(d) Mixture enthalpy equation (12)
\[ \frac{\partial w_h}{\partial t} + \frac{\partial w_h}{\partial z} = Q_1 \]

For closure, several constitutive equations are also needed. For the heat transfer coefficient, the Dittus-Boelter correlation is used for the single phase region, and the Chen correlation is used for the two phase region. There are two options for the distribution parameter \( C_0 \) and the drift velocity \( V_d \). One option is to use the Ishii correlation; however, this requires the determination of a flow regime map. Another option two is the full range correlation by Chexal where the flow regime map is not required. For the wall friction factor Blasius’ correlation or Colebrook’s correlation is used. For the two-phase friction multiplier, three correlations, Chisholm, Owens, and McAdam are available.

The vapour generation/suppression rate \( f_g \) entering via \( Q_2 \) in equation (10) is shown to be (2).
\[ f_g = \frac{2\bar{a}_g}{R_h \rho_g} + \frac{\rho_g}{h_f} \left[ x \frac{d\bar{a}_g}{d\rho_g} - \frac{1}{\rho_g} \frac{\partial \rho_g}{\partial x} \right] \]

Flashing

2.2 Thermal Hydraulic Formulation

In the formulation described above, the four independent parameters are the channel power, inlet subcooling, and outlet header pressure. For any combinations of the above parameters, the system thermal hydraulic conditions and the system stability characteristics can be determined. To obtain the system thermal hydraulic operating conditions, the equation of (1) to (4) are solved in the steady state with the following boundary conditions:
(a) Closed loop \( f_P = \Delta P_{\text{pump}} \)
(b) Net energy balance \( \sum_{i=1}^{N} Q_i = 0 \)
(c) Continuous void fraction and density loop around the loop

By combining the mass and momentum equations (9) and (11), the steady state mass flow rate can be expressed as:
\[ \dot{m} = \left( \Delta P_{\text{pump}} - \rho_m g \sin \psi \frac{dz}{dr} \right)^{1/2} \]

where the mixture density \( \rho_m \), friction factor and fraction multiplier must be determined from the void propagation Equation (10) and energy Equation (12). To simplify the coupling between Equations (17), (10) and (12) a transformation from the axial coordinate (\( \tau \)) to the residence time coordinate (\( t \)) through the transformation
\[ \tau = \frac{1}{u_g} \frac{dz}{dt} \]

is needed. As a result, the expression for mixture density is simply expressed as
\[ \rho = \bar{a}_g \rho_j + (1-\bar{a}_g) \rho_f \]

where
\[ \bar{a}_g = \frac{\rho_f}{\rho_j} \left[ 1 - (1-\bar{a}_g) e^{-Q_2 \frac{\tau}{C_0}} \right] \]

Flashing

2.3 Transmission Matrix Formulation

To formulate the transmission matrices, general perturbations of mass flow rate, void fraction, enthalpy, and pressure (i.e., liquid and vapour density) of the form are introduced in Equations (9) to (12) for each appropriate region, neglecting second and higher order term. The resulting equations are then Laplace
transformed to reduce the partial D.E to ordinary D.E in the spatial coordinate. This gives a general set of four coupled differential equations, the solutions of which describe the distribution of the disturbances along the section. From these distributions individual transmission matrices can be obtained by expressing the disturbances at the outlet of the region in terms of the inlet disturbances. Several assumptions can be applied to each individual section to simplify the general form of these matrices.

\[ \dot{w} = \bar{w} + \delta \omega^* (\theta, \zeta); \quad \alpha = \bar{\alpha} + \delta \alpha^* (\theta, \zeta) \]
\[ p = \bar{p} + \delta p^* (\theta, \zeta); \quad h = \bar{h} + \delta h^* (\theta, \zeta) \tag{22} \]

For the two-phase section, the following assumptions are applied:

(a) The change in mixture density due to void formation is more dominant than the change in individual vapour and liquid density.

(b) The liquid phase is incompressible, the vapour phase is compressible.

(c) Liquid and vapour phases are at the same pressure.

(d) The vapour density is related to the mixture pressure and enthalpy through the equation of state of the form 

\[ \rho_v = \rho (h_g, P_g); \text{ thus} \]

\[ \delta \rho_v = \left( \frac{\partial \rho_v}{\partial h_g} \right) \delta h_g + \left( \frac{\partial \rho_v}{\partial P_g} \right) \delta P_g \tag{23} \]

Because the liquid is assumed to be incompressible, the equations for the liquid region are decoupled and can be easily solved. However, for the two-phase region, the formulation results in a system of four heavily coupled differential equations that describes the axial distribution of the disturbances in flow, pressure, void fraction and enthalpy given an inlet perturbation. The equations are written as follows:

(a) Perturbed void propagation equation (25)

\[ \frac{d^2 \delta \omega}{d\tau^2} + \frac{s Q_2}{C_0} \delta \omega = s^2 \frac{S + 2Q_2}{C_0} (1 + uq_1) e^{Q_2 \gamma/C_0} \delta \alpha + \frac{s^2 G_{ref}}{P_0 \Delta P/C_0} (1 + uq_1) e^{Q_2 \gamma/C_0} \delta \alpha \tag{25} \]

(b) Perturbed momentum equation (26)

\[ \frac{d \delta \rho}{d\tau} = - \left( s + \frac{\delta \rho}{\rho} \right) \frac{1 + uq_1}{1 - \alpha_1} e^{Q_2 \gamma/C_0} \delta \alpha \]

\[ - \frac{\delta \rho}{\rho^2} \left( \frac{1 + uq_1}{1 - \alpha_1} e^{Q_2 \gamma/C_0} \delta \alpha \right) \tag{26} \]

For the liquid sections, the following assumptions are applicable:

(a) Liquid density is directly proportional to temperature through Boussinesq's approximation.

(b) Liquid phase is incompressible.

(c) No heat loss in the inlet feeder piping.

(d) Wall temperature is azimuthally uniform.

Figure 2: Solution methodology for the model.
The stability of the system is determined by examining the roots of the determinant

$$|H| = |I - G|.$$  

The system is unstable if roots with a positive real part are found. For roots with a positive real part, the imaginary part of the roots indicates whether the instability is excursive (Leideneck instabilities - zero root) or oscillatory in nature (non-zero root). The significance of the contribution of each inlet disturbance to the outlet disturbances (i.e., how sensitive are outlet disturbances in flow, pressure and enthalpy to the inlet disturbances) are determined from the argument of each coefficient in the transmission matrix. An examination of these coefficients has indicated that flow and pressure have the most significant effect and the effect of inlet enthalpy disturbances is negligible.

Another method of examining the stability of the physical system without having to completely solve for all the roots of the determinant, is with a polar plot and application of the Nyquist criterion of stability. This is a way of mapping points along a Nyquist contour in the S-plane (frequency domain) into a point in the characteristic function plane (f-plane). The closed loop system is unstable if $|H(s)| = |I - G(s)|$ has any root in the right-half of the S-plane (including the imaginary axis). By assuming that there is no pole present, then any clockwise encircling of the origin point (-1 + 0j) indicates an unstable system.

3.0 LIMIT CYCLE

The linear model described above is used to determine the conditions under which oscillations occur and the nature of the oscillations. However, at large amplitude when conditions deviate sufficiently from the steady state operating conditions, the model always indicates an unbounded oscillation. In reality, no oscillation will be unbounded because of the limitation of the physical system, and the oscillation will approach a limit cycle. For example, in the figure-of-eight geometry, an excursive increase of void in one pass and decrease of void in the other pass cannot be sustained once the void in one side has collapsed. When this occurs the liquid regions are joined and the oscillation is bounded by a limit cycle amplitude.

The following describes a method used to estimate the limit cycle oscillation amplitude at different outlet header pressures. The method is derived from a lumped but non-linear model representing the simplified single channel symmetrical figure of eight circuit shown in Figure 3. Four lumped volumes are considered in the formulation, representing two single phase (liquid) regions and two two-phase regions. Because of the lumped volumes selected, the formulation is a moving boundary problem. In each of the four regions, there are four governing equations: conservation of mass, momentum, energy and the equation of state. The
solutions of these equations gives the density, mass flux, enthalpy and pressure transient during the oscillation cycle. In carrying out the formulation, the following assumptions are used:

(a) After the onset of the flow instability, the instability is oscillatory and the flow oscillations in the two halves of the circuit are completely out of phase (180° out of phase oscillation). i.e., when the flow in one pass increases by dw, the flow in the other pass decreases by dw.

(b) The two-phase mixture is homogeneously distributed and in thermodynamic equilibrium. Liquid and vapour phase velocities are the same and equal to the mixture velocity.

(c) The effect of heat addition/removal is the most dominant when considering the energy equations.

(d) The density and void fraction are continuously distributed in the circuit.

The lumped governing equations for each region are derived by integrating the conservation equations along the length of each region. Leibnitz’s rule of integration is used to account for the effect of the moving boundary (i.e., the limit of integration is a function of time). The one-dimensional co-ordinate is taken along the axial direction of the circuit with the origin at the inlet of the heated length in one pass. Subsequent boundaries of the four regions are x1, x2, x3 and x4.

By introducing the perturbation in flow in the three conservation equations and assuming that only the change in the primary variables (flow, density, enthalpy and pressure) are significant, the governing lumped equations, and equations governing the region boundaries are obtained as a system of 13 coupled differential equations. The solution to these equations gives the thermalhydraulic response during the oscillations.

From Appendix A, the resulting Equations (A.1) to (A.3) show that there is no phase shift between the single phase density, pressure and flow.

For the two-phase region, Equation (A.4) shows two competing effects in the void collapsing and formation process. The first effect is the reduction of void in the two-phase region due to increase of flow. The second effect is the increase in void because of the moving boiling and condensing boundaries. In general, the effect of the first term is the most significant and the second term can be neglected. Since Equation (A.6) shows an integrated flow effect in the voiding process, the void fraction decreases as flow increases but the void reduction process takes the flow by one integration. In the case of a periodic oscillation, this implies that the change in void lags the change in flow by one quarter of a cycle (90° lag).

With the effect of incoming flow being dominant, the average pressure for the two-phase region L1 increases with increasing flow but the change in pressure lags the flow by 90°. By comparing the phase lags of void with respect to flow, and pressure with respect to flow, it can be deduced that the change in void and pressure oppose one another in a single pass. Also, the void in the two-phase region is smallest when the two-phase region average pressure is highest (Figure 4).

The void collapse condition is used to provide a conservative estimate of the magnitude of the flow, and pressure oscillations in the figure-of-eight circuit. Based on the lumped formulation, it is assumed that void collapse occurs at the maximum pressure at an effective average flow. Since the void collapsing process starts when the channel flow is a maximum and completes when the channel flow is a at the average, the effective flow required to collapse the void is obtained as:

\[
\bar{\dot{w}} = \frac{1}{T} \int_0^T \dot{w}(t) \, dt
\]
where $T$ is the oscillation period. For a constant inlet header pressure, this effective flow is a function of the average outlet header pressure and its maximum oscillation amplitude. A unique solution for the latter is obtained from a heat balance given by:

$$\dot{\bar{u}} = \frac{0}{P_{\text{outlet}} \left( P_{\text{max}} \right) - P_{\text{inlet}}}$$

In practice the maximum and average outlet header pressure are obtained iteratively.

4.0 RESULTS

Figures 5 and 6 shows the NYQUIST plot at outlet header pressure of 7.25 MPa and 7.1 MPa respectively for a typical Pickering type CANDU reactor. In Figure 5 the mapped contour does not encircle the origin (-1.0j) point; therefore, the system is stable at 7.25 MPa outlet header pressure. At and below 7.1 MPa, the contour encircles the origin. This indicates that the system is unstable. Examination of the roots shows a pair of complex conjugate roots with a non-zero imaginary part. This indicates an instability of oscillatory nature.

Figure 6: NYQUIST plot at outlet header pressure of 7.1 MPa.

Figure 7: Predicted limit cycle amplitude using the void collapse criterion and methodology described above. It is noted that outlet header pressure oscillation amplitude increases as the average pressure decreases.

5.0 CONCLUSIONS

A rigorous model based on the drift-flux conservation equations, the concept of transmission matrices and the void collapse criterion has been developed to assess the figure-of-eight type flow oscillations in CANDU
primary heat transport systems. The model can be generically applied to predict the operating thermal hydraulic conditions, the stability characteristics of the system and, in the event of flow oscillations, the maximum oscillation amplitude boundary conditions.

REFERENCES

1. TRAN, F.B.P., GARLAND, W.J., "Modelling of Two Phase Instabilities in a Figure of Eight Loop Under Natural Circulation Conditions", 11th Simulation Symposium, April 1986.

2. TRAN, F.B.P., "Two Phase Instabilities in CANDU Primary Heat Transport System Geometry".


APPENDIX A

Non Linear Lumped Formulation

Using the assumptions stated in Section 3, the resulting governing equations for the single and two phase region in the increasing flow pass are the following.

(a) Single Phase Region Lo:

Continuity Equation: (A.1)

\[ \frac{dP_0}{dt} = \frac{A_H}{Q} (h_f - h_l) \left(1 - \frac{1}{\xi} (\rho_f - \rho_o) \right) \frac{d\delta w}{dt} \]

Energy Equation: (A.2)

\[ \frac{dP_0}{dt} = \frac{A_H}{Q} (h_f - h_o) \left(1 - \frac{1}{\xi} \right) \frac{d\delta w}{dt} \]

Equation of State: (A.3)

\[ \frac{dP_0}{dt} = \frac{A_H}{Q} (h_f - h_l) \left(1 - \frac{1}{\xi} \right) \frac{d\delta w}{dt} + \frac{\delta h \rho_f}{\delta P} \frac{d\delta w}{dt} \]

(b) Two Phase Region L1:

Continuity Equation: (A.4)

\[ \frac{d\delta w}{dt} = -2\delta w + \frac{A_H}{Q} (h_f - h_l) \left(1 + \frac{1}{\xi} \right) \frac{d\delta w}{dt} \]

Energy Equation: (A.5)

\[ \frac{d\delta w}{dt} = -2(h_f - h_l) \delta w \]

Equation of State: (A.6)

\[ \frac{d\delta w}{dt} = \frac{A_H}{Q} (h_f - h_l) \left(1 + \frac{1}{\xi} \right) \frac{d\delta w}{dt} \]

\[ \frac{d\delta w}{dt} = \frac{2\delta w}{L1} \left( \frac{\delta h \rho_f}{\delta P} - \frac{\delta h \rho_f}{\delta \rho} \right) \left( h_l - h_f \right) \]

\[ \frac{d\delta w}{dt} = \frac{A_H}{Q} (h_f - h_l) \left(1 + \frac{1}{\xi} \right) \frac{d\delta w}{dt} \]

\[ + \frac{\delta h \rho_f}{\delta P} \frac{d\delta w}{dt} \]

\[ + \frac{\delta h \rho_f}{\delta \rho} \frac{d\delta w}{dt} \]
Nomenclature:

\( A \) : cross sectional area
\( C_p \) : heat capacity
\( D' \) : pipe diameter
\( D_h \) : hydraulic diameter
\( E_c \) : Eckert number
\( Fr \) : Froude number
\( G \) : mass flux
\( G \) : half loop transmission matrix
\( H \) : dimensionless heat transfer coefficient
\( J_a \) : Jacob number (subcooling) = \((h_f-h_g) \frac{P_f \cdot V_{fg}}{h_f}\)
\( K \) : heat conductivity
\( L_c \) : characteristics length
\( L_y \) : length of section i-j
\( P_m \) : mixture pressure
\( Q_1 \) : dimensionless heat flux = \(2q''\)
\( Q_2 \) : phase change number
\( R \) : radius of pipe
\( T \) : temperature
\( U \) : heat transfer coefficient
\( V_g \) : vapour velocity
\( V_gj \) : drift velocity
\( d_w \) : wall thickness of pipes
\( f_m \) : mixture friction factor
\( f \) : dimensionless friction factor
\( g \) : gravitational acceleration constant
\( h_m \) : mixture enthalpy
\( h \) : dimensionless mixture enthalpy
\( h_w \) : dimensionless wall enthalpy
\( h_f \) : saturated liquid enthalpy
\( h_g \) : saturated vapour enthalpy
\( h_{fg} \) : liquid enthalpy at heater inlet
\( h_{fg} \) : = \( h_f - h_g \)
\( j \) : volumetric flux
\( p \) : dimensionless pressure
\( q'' \) : wall heat flux
\( s \) : Laplace variable
\( t \) : time
\( t_{ref} \) : reference time
\( u_g \) : dimensionless vapour velocity
\( u_{gj} \) : dimensionless drift velocity
\( V_g \) : = \(1/\rho_g\), liquid specific volume
\( V_{gf} \) : = \(1/\rho_f\), vapour specific volume
\( V_{ref} \) : reference velocity
\( w \) : dimensionless mass flux
\( z \) : axial coordinate
\( x \) : thermodynamic quality

Greek:

\( \alpha_g \) : dimensionless void fraction
\( \alpha \) : void fraction
\( \xi \) : heated perimeter
\( P_g \) : equilibrium vapour generation
\( P_{g,ne} \) : non-equilibrium vapour generation
\( \lambda_{eq} \) : equilibrium boiling boundary
\( \lambda_{ne} \) : non-equilibrium boiling boundary
\( \beta \) : dimensionless mixture density
\( \rho_g \) : vapour density
\( \rho_f \) : liquid density
\( \rho_m \) : mixture density
\( \theta \) : dimensionless time
\( \tau \) : dimensionless residence time
\( \zeta \) : dimensionless axial coordinate
\( \sin \alpha \) : pipe inclination
\( \Gamma \) : vapour generation rate

Subscript:

\( f \) : liquid phase
\( g \) : vapour phase
\( i \) : inlet
\( o \) : outlet
\( w \) : wall
\( \text{ref} \) : reference
USE OF A THREE-BEAM GAMMA DENSITOMETER TO DETERMINE AVERAGE VOID FRACTION AND FLOW REGIME

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ABSTRACT

Multi-beam gamma densitometers are commonly used to determine the average void fraction and phase distribution in two-phase flow experiments. A simple, reliable and efficient algorithm is developed to determine the void fraction and flow regime from the experimental data.

The algorithm was first tested on idealized and realistic flow situations, where the chordal and average void fractions, and phase distributions were known exactly. Results were found to be in excellent agreement with the exact flow cases. Then, a comparison with a previous technique was made using experimentally obtained input. The present scheme required considerably less computational time and yielded comparable results.

INTRODUCTION

For an adequate understanding of many physical situations involving two-phase flow, e.g., a mixture of gas and liquid flowing through a pipe, the cross-sectional average void fraction and distribution of the phases should be known. In particular, an understanding of the results of a transient thermohydraulics experiment often depends on measurements of the void fraction and flow regime at various locations. Multi-beam gamma densitometers are commonly used for this purpose. The geometrical structure of the three-beam gamma densitometer developed and in use at Whiteshell Nuclear Research Establishment (WRRE) is shown in Figure 1. Three coplanar gamma rays originating at a single source pass through the mixture in the pipe. The intensity of the rays at the source is known and the attenuated intensity of each ray is measured by a detecting device.

The first step, to determine the void fraction and flow regime from the measured data is to obtain the chordal void fractions along the ray paths, which may be calculated from the attenuation caused by the mixture. Heldrick et al. (1) isolated the effect of the mixture by calculating the attenuation caused by the pipe walls. This calculation can be avoided by making measurements on the empty pipe. Also, the attenuation coefficient of the liquid can be eliminated by measuring the attenuation produced by a pipe filled with liquid of known density. This procedure to determine the chordal void fractions is described in the present paper.

Several attempts have been made in the past to obtain the average void fraction and the distribution of phases from the knowledge of the chordal void fractions (1,2). The procedures of References 1 and 2 are very similar, and provided the basis for the code named "FAST" that was in use at WRRE. The algorithm used in FAST is based on a three-parameter model of the flow, which is assumed to be displaced annular, superposed with a homogeneous mixture of gas and liquid. Several values of the chordal void fractions \( \alpha_j \) and cross-sectional average void fraction \( \bar{\alpha} \), are computed and tabulated. A "best approximation" value for the void fraction is then found in the table and the corresponding values of the parameters are assumed to describe the flow. This approach requires computation of a large number of values not directly related to the experimental situation and hence is very time-consuming. In the present paper, we develop an algorithm that uses the values of \( \alpha_j \) directly to compute \( \bar{\alpha} \) and phase distributions.

![FIGURE 1: BEAM GEOMETRY OF THE WRRE THREE-BEAM GAMMA DENSITOMETER.](image-url)
A very time-efficient numerical scheme.

The present algorithm was first tested on known, idealized flow situations, namely, homogeneous, radially symmetric annular and stratified, as well as their linear combinations. As expected, the scheme yielded void fraction and flow regime exactly. As a second step, it was tested on a more realistic and known flow situation, namely, the displaced annular. Results were found to be in close agreement with the exact ones. The next step was to use the present technique to calculate the void fraction where those calculated by the previous method were available. Finally, a computer program was developed to calculate the time-varying void fractions and densities for a complete set of experimental data, and to plot the results.

**Representations of Void Fractions**

Let \( (r, \phi) \) denote a point inside the pipe in the plane of beams in a polar coordinate system with origin at the source of \( \gamma \)-rays, as shown in Figure 2. Because the plane of rays and the length of pipe intersect at a right angle, \( (r, \phi) \) varies over a circular region. As \( (r, \phi) \) varies over this region, \( \phi \) varies from \( \phi_0 \) to \( \phi_1 \), the angular coordinates of two tangents to the circle issuing from the origin. For a fixed \( \phi \), \( r \) varies from \( r_0(\phi) \) to \( r_1(\phi) \), the radial coordinates of the points of intersection of the line \( L(\phi) \) with angular coordinate \( \phi \) and the circumference of the circle. The density of the mixture at \( (r, \phi) \) will be denoted by \( \rho(r, \phi) \).

**Figure 2: Polar Coordinate System Geometry**

**Chordal Void Fraction**

The chordal void fraction \( a(\phi) \) along \( L(\phi) \) is given by

\[
a(\phi) = \frac{\rho_L - \rho(\phi)}{\rho_L - \rho_g}
\]

where \( \rho_L \) and \( \rho_g \) denote the densities of the liquid and gas phases, respectively, and

\[
\rho(\phi) = \frac{1}{d(\phi)} \int_{r_0(\phi)}^{r_1(\phi)} dr \rho(r, \phi)
\]

with \( d(\phi) = (r_1(\phi) - r_0(\phi)) \). Let \( s(\phi) \) and \( \tilde{s}(\phi) \) denote the sets in \([r_0(\phi), r_1(\phi)]\) such that, for \( r \) in \( s(\phi) \), \( \rho(r, \phi) = \rho_L \), and for \( r \) in \( \tilde{s}(\phi) \), \( \rho(r, \phi) = \rho_g \). It follows that

\[
\rho(\phi) = \frac{1}{d(\phi)} \int_{s(\phi)} \rho_L \, dr + \frac{1}{d(\phi)} \int_{\tilde{s}(\phi)} \rho_g \, dr
\]

Because \( \rho_L = \frac{\rho}{d(\phi)} \int_{s(\phi)} dr \), \( \rho(\phi) = \frac{r_1(\phi)}{d(\phi)} \int_{s(\phi)} \Gamma(s(\phi), r) \).

The average void fraction, \( \bar{a} \), is given by

\[
\bar{a} = \frac{\rho_L - \rho_g}{\rho_L - \rho_g}
\]

where

\[
\rho = \frac{\frac{1}{R} \int_{0}^{\frac{1}{R}} \int_{0}^{\phi_0} \rho(r, \phi) \, d\phi \, dr}{\rho_0} = \frac{1}{\pi R^2} \int_{0}^{\phi_0} \rho(r, \phi) \, d\phi
\]

Because \( \rho_L = \frac{\phi_1}{\pi R^2} \int_{0}^{\phi_1} \rho(r, \phi) \, d\phi \), \( \rho(\phi) = \frac{\phi_1}{\pi R^2} \int_{0}^{\phi_1} \rho(r, \phi) \, d\phi \).

It follows that

\[
\bar{a} = \frac{1}{\pi R^2} \int_{0}^{\phi_0} d\phi \int_{0}^{\phi_0} \Gamma(\phi, r) \, d\phi
\]

The function \( \Gamma(\phi, r) \) appearing in Equations (2) and (4) depends on the distribution of phases. Consequently, the value of

\[
\Gamma(\phi, r) = \int_{r_0(\phi)}^{r_1(\phi)} dr \Gamma(s(\phi), r)
\]

depends on \( s(\phi) \), a function defining the distribution, and some other functions of \( \phi \). Consider a class of distributions, for example, annular, such that a beam at angle \( \phi \) passes through liquid to gas to liquid. Then a straightforward calculation yields

\[
\Gamma(\phi) = (r_1^2 - 2) \, a + (1 - 2\beta \, (r_1 - r_0)^2 (1 - a) a
\]
where \( \beta = \beta(\phi) \) is the fraction of liquid encountered by the beam during the last segment of its passage. When \( \beta = 0 \) or 1, this class of distributions also includes cases of beams passing through liquid to gas or gas to liquid, respectively. Also \( \beta = 1/2 \) results in a passage through a homogeneous mixture. Thus the example illustrated here applies to homogeneous, annular and stratified flows.

**METHOD**

The procedure to determine the average void fraction and flow regime from the measured values of attenuation may be conveniently described in two steps:

1. Determination of chordal void fractions from the measured values of incident and transmitted intensities.
2. Determination of average void fraction and flow regime from the chordal void fractions.

**Chordal Void Fraction**

Referring to Figure 1, the strength of the transmitted (attenuated) beam, \( N \), after traversing the pipe wall, the gas and liquid phases, is given by:

\[
N = N_s \cdot \exp(-2 \mu_d \omega) \cdot \exp \left[ -\mu_d (\rho_f - (\rho_f - \rho_g) a') \right].
\]

(6)

where:
- \( N \) = attenuated beam strength,
- \( N_s \) = incident beam strength,
- \( \mu_d \) = attenuation coefficient of the pipe wall material,
- \( \mu_{\omega} \) = attenuation coefficient of H2O,
- \( d_d \) = thickness of the pipe wall,
- \( d_c \) = chordal length inside the pipe traversed by the beam,
- \( d_0 \) = chordal length occupied by the gas-phase traversed by the beam,
- \( a^* = d_d/d_c = \) chordal void fraction.

From Equation (6), for an empty pipe \( N = N_s \) is given by

\[
N_s = N_s \cdot \exp(-2 \mu_d \omega) .
\]

(7)

Similarly, for a pipe filled with water at a standard temperature and pressure, with \( \rho_f = \rho_{\text{water}} \) and \( a^* = 0 \), the beam strength \( N_t \) is given by

\[
N_t = N_s \cdot \exp(-2 \mu_d \omega) \cdot \exp(-\mu_d \rho_{\text{water}} t) .
\]

(8)

With Equation (7), Equations (6) and (8) are reduced to

\[
N = N_s \cdot \exp(-\mu_d \omega \rho_{\text{water}} t) \cdot \exp(-\mu_d \rho_{\text{water}} t a^*)
\]

(9)

\[
N_t = N_s \cdot \exp(-\mu_d \rho_{\text{water}} t)
\]

(10)

which yield

\[
a^* = \frac{\rho_f}{(\rho_f - \rho_g)} \left[ 1 - \frac{\rho_{\text{water}} \ln N_s - \ln N}{\rho_f \ln N_s - \ln N_t} \right] .
\]

(11)

By substituting for \( N_t, N_s \) and \( N_f \) the values measured for jth chord in Equation (11), one obtains \( a_j = a^* \).

**Average Void Fraction and Flow Regime**

The function \( f(\phi, r, \tau) \) determines \( a \) and the phase distribution completely. Thus the purpose here is to approximate \( f(\phi, r, \tau) \) using the measured values of \( a(\phi, \tau) = a_j \).

Let \( \{\phi_j(\tau)\} \) be a set of approximating functions, i.e., there are constants \( \lambda_j \) such that \( \sum \lambda_j \phi_j(\tau) \) is close to \( f(\phi, r, \tau) \) in some sense. From Equation (7), values of \( \lambda_j \) may be determined by solving the following set of algebraic equations:

\[
a_j = \frac{1}{\pi r_j} \sum_{j=1}^{n} \lambda_j \int_0^{2\pi} \phi_j(\phi, r) \, d\phi .
\]

(12)

An approximate value of \( a_0, a_n \), is then given by

\[
a_j = \frac{1}{\pi r_j} \sum_{j=1}^{n} \lambda_j \int_0^{2\pi} \phi_j(\phi, r) \, d\phi .
\]

(13)

For the three-beam densitometer, under consideration, \( n = 3 \). A small value of \( n \) makes it necessary to insert a set \( \{\phi_j\} \) so that a physical \( f \) is satisfactorily approximated. In view of the realistic distributions encountered, it is expected that each of them would be closely approximated by a linear combination of the distributions corresponding to (1) homogeneous, (2) radially symmetric annular, and (3) stratified flow patterns. Consequently, \( \eta_1 \) and \( \xi_1 \) are the chordal and average void fractions, respectively, corresponding to \( j \)th flow regime. The elements \( \eta_j, \xi_j \) may be computed using trigonometric manipulations in terms of \( \xi_j \). Here we restrict ourselves to the geometry of Figure 1. Let \( \gamma = \gamma/\kappa \), where \( \gamma \) is the vertical coordinate of the level of liquid flowing in a stratified pattern, in the \( x-y \) coordinate system with its origin at the centre of the pipe. Thus, \(-1 \leq \gamma \leq 1 \). We have

\[
\eta_1 = \eta_2 = \eta_3 = \xi_1 .
\]

(14i)

\[
\eta_1 = \eta_3 = \max \left( \frac{\xi_2 - \sin^2 \gamma}{\cos^2 \gamma} \right)
\]

(14ii)

\[
\eta_2 = \frac{\xi_2}{\xi_1}
\]

(14iii)

\[
\eta_3 = \min(1, \max(0, \frac{1 - 1 - \gamma}{2 \cos^2 \gamma})
\]

(14iv)

\[
\eta_3 = \min(1, \max(0, \frac{1 - (1 - 1 - \gamma)}{2 \cos^2 \gamma})
\]

(14v)

\[
\eta_3 = \min(1, \max(0, \frac{1 - 1 - \gamma}{2 \cos^2 \gamma})
\]

(14vi)

\[
\eta_3 = \min(1, \max(0, \frac{1 \gamma}{2 \sin^2 \gamma})
\]

(14vii)

\[
\xi_1 = \frac{1}{y} \left[ \sin \gamma + \gamma \left(1 - \gamma \right) \right]
\]

(14viii)
According to the above definitions, $\eta_{j}$ and $\beta_{j}(\phi, t)$ are parameter-dependent, introducing in addition to $\eta_{j}$, three parameters $\xi_{j}$. When we set $\xi_{j} = a_{j}$, then from Equation (13), $\sum_{j} \xi_{j} = 1$. Physically, this means that $\lambda_{j}$ is the fraction of liquid and gas arranged in the $j$th flow pattern determined by $a_{j}$.

The problem now is reduced to solving the following set of equations:

$$
\begin{align*}
\lambda_{1}\bar{a} + \lambda_{2} \max(0, \frac{\bar{a} - \sin^{2} \theta}{\cos \theta}) &= a_{1}, \\
\lambda_{3} \min(1, \frac{1 - \frac{\sqrt{a}}{2}}{\cos \theta}) &= a_{2}, \\
\lambda_{4}\bar{a} + \lambda_{5} \max(0, \frac{\bar{a} - \sin^{2} \theta}{\cos \theta}) &= a_{3}, \\
\lambda_{6} \min[1, \max(0, \frac{1 - \frac{\sqrt{a}}{2}}{\sin \theta})] &= a_{4}.
\end{align*}
$$

(15i - 15v)

Since there is a distinction between exact and approximate values due to the convergence sequence, we omitted the subscript from $\xi_{j}$. From Equations (13i) and (15iii), we have

$$
D(\theta) = \min(1, \frac{1 - \frac{\sqrt{a}}{2}}{\cos \theta}).
$$

(16i)

If $D(\theta) = 0$, the determinant encountered in solving Equations (15i) to (15iii) can be shown to be equal to zero. Hence, from the Fredholm alternative, a solution exists if and only if $a_{1} = a_{6}$, and the existence of one solution implies the existence of a family of solutions. If we assume that $a_{1} = a_{6}$, an additional condition is needed to determine the solution uniquely. The case $a_{1} = a_{6}$ corresponds, for symmetry reasons, to the absence of stratified flow pattern, i.e., $\lambda_{j} = 0$. Therefore, we set $\lambda_{j} = 0$ if $D(\theta) = 0$; otherwise, $\lambda_{j} = (a_{1} - a_{6})/D(\theta)$.

Let

$$
\xi(\theta) = \sqrt{\bar{a}(\theta) - \max(0, \frac{\bar{a}(\theta) - \sin^{2} \theta}{\cos \theta})},
$$

(16ii)

with $\bar{a}(\theta)$ defined by Equation (15iv). As above, we conclude that, if $\xi(\theta) = 0$, then $\lambda_{2} = 0$; otherwise, from Equations (15i) and (15ii),

$$
\lambda_{2} = \frac{1}{\xi(\theta)} (a_{2} - a_{1}) - \lambda_{1}\min(1, \max(0, \frac{1 - \frac{\sqrt{a}}{2}}{\cos \theta}))
$$

(17)

The value of $\lambda_{1}$ is determined by Equation (15v), i.e.,

$$
\lambda_{1} = 1 - \lambda_{2} - \lambda_{3}.
$$

From Equation (15ii),

$$
h(\theta) = a_{2} - \lambda_{1} a_{1} - \lambda_{2} \int_{0}^{1} a(\theta) \cos \phi d\phi.
$$

(18)

Let $\tilde{\theta}$ be the solution of Equation (18). Then an approximate value of the average void fraction is given by $\bar{a}(\tilde{\theta})$ as defined by Equation (15iv), and $\lambda(\tilde{\theta})$ determines the phase distribution. For illustration, the values of $\lambda$ for ideal flow patterns are as follows: for homogeneous flow patterns, $\lambda_{1} = 1, \lambda_{2} = \lambda_{3} = 0$; for radially symmetric annular flow patterns, $\lambda_{1} = \lambda_{3} = 0, \lambda_{2} = 1$; for stratified flow patterns, $\lambda_{1} = \lambda_{2} = 0, \lambda_{3} = 1$.

**Solution of $h(\theta) = 0$**

Although there are numerous methods available to solve Equation (18), we found a combination of the secant and bisection methods to be well suited for the present purpose. These methods are described in many textbooks, such as Hornbeck. (3)

Since the function $h(\theta)$ is defined over the range $-1 \leq \theta \leq 1$, for convenience we always used $\theta_{0} = 1$ and $\theta_{1} = 1$ for the two initial guesses. Next step was to check if either of these was the solution and, if not, that at least one solution existed. Further approximations to the solution were obtained by the secant method. At each iteration, the value of $h(\theta)$ was checked to see if the solution had been found (within tolerance). If the solution had not been found and the rate of convergence was too slow, the bisection method was used for one iteration.

**ALGORITHM**

In this section, we show the steps that have to be followed to determine the average void fraction and the phase distribution from the measured values of the attenuated intensities $N_{j}$, $N_{j}^{(t)}$, and $N_{j}$ corresponding to the empty pipe, pipe filled with fluid of density $\rho_{e}$ and pipe filled with liquid-gas mixture, respectively. The subscript $j$ refers to the $j$th chord and $\rho_{l}$ and $\rho_{g}$ are liquid and gas densities.

**STEP 1**: Determine the chordal void fractions $a_{j}$ using Equation (11).

**STEP 2**: Define function $h(\theta)$ for $-1 \leq \theta \leq 1$, as follows:

- Calculate $D(\theta)$ using Equation (16i). If $D(\theta) = 0$, then $\lambda_{1}(\theta) = 0$; otherwise, $\lambda_{1}(\theta) = (a_{1} - a_{6})/D(\theta)$.
Calculate \( \hat{f}(\tilde{r}) \) using Equation (15iv) and \( \hat{\zeta}(\tilde{r}) \) using Equation (16ii). If \( \hat{\zeta}(\tilde{r}) = 0 \), then \( \lambda_2(\tilde{r}) = 0 \); otherwise, calculate \( \lambda_2(\tilde{r}) \) using Equation (17).

Set \( \lambda_1(\tilde{r}) = 1 - \lambda_2(\tilde{r}) - \lambda_3(\tilde{r}) \).

Calculate \( h(\tilde{r}) \) using Equation (18).

**STEP 1** Obtain \( \tilde{r} \) as described above ("Solution of \( h(\tilde{r}) \)").

**STEP 4** Calculate \( \hat{a}(\tilde{r}) \) and \( \lambda_1(\tilde{r}) \) as in Step 2.

**VERIFICATION AND APPLICATIONS**

For verification, the algorithm described above was used to calculate \( \hat{a} \) and \( \lambda_j \) from the values of \( a_j \) for \( j = 1, 2, 3 \); corresponding to known flow patterns and their superpositions, with void fraction \( a_k \). For homogeneous, radially symmetric annular, stratified and their linear combinations, \( \hat{a} = a_k \), and the flow pattern was identified correctly from the values of \( \lambda_j \), as expected.

Displaced annular flow is commonly found in a horizontal flow where gravity distorts the radial symmetry. Figure 3(a) is an example of such a flow with \( a_k = 0.4 \) and with centre of the gas displaced by 0.3 of the pipe radius. Since this is not one of the ideal flow patterns used to construct the basis functions, it provides a nontrivial, realistic example for which the exact value \( a_k \) is known. As a second phase of the verification, the chordal void fractions were first calculated for a given \( a_k \) and displacement \( R_a \) (ratio between the centre-to-centre distance and the pipe radius), which were then used to compute \( \hat{a} \) and \( \lambda_j \) by the present method. A comparison between the values of \( \hat{a} \) and \( a_k \) is shown in Figure 4. Agreement in this case is not perfect. However, the maximum error in \( \hat{a} \) is less than 10\%, and the approximate phase distribution has the basic features of the exact one: the maximum density occurs at the bottom of the pipe, the density decreases towards the middle, and then increases to a lesser value near the top.

For calculations described above, Step 1 of the algorithm was not used as the values of \( a_j \) were obtained from that of \( a_k \) and the flow pattern.

The algorithm was further tested using experimental data collected from a blowdown experiment conducted at WNNR. (4) The flow conditions in this experiment are typical of two-phase flow phenomena covering a wide range, from a subcooled water flow to a two-phase flow. The raw data includes time, pressure and beam strengths. Average density \( \rho \) determined by FAST program at the inlet and outlet of test section 2 was compared with that obtained by the present scheme. The results are shown in Figure 5. In Figure 5(a), which shows the results for the test section outlet, the density values are all two-phase densities except for points marked A, B, C and D, which are subcooled densities. Figure 5(b) shows the results for the test-section inlet where the points marked A, B are two-phase densities, the rest being mostly, the subcooled water densities. Although there is some discrepancy, on the whole, the agreement between the results obtained by the two methods is excellent.

**FIGURE 3: REALISTIC FLOW PATTERN AS A SUPERPOSITION OF IDEALIZED FLOW PATTERNS**

**FIGURE 4: COMPARISON OF CALCULATED AND KNOWN VOID FRACTIONS FOR DISPLACED ANNULAR FLOW PATTERN**

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A FORTRAN program implementing the algorithm described above was written to calculate the time-varying average void fraction and mixture density for a complete set of experimental data, and plot the results.

In implementing this algorithm, we employed several novel steps. First, the measured data were rounded to a given experimental tolerance prior to any calculations. Second, in the cases where the flow was homogeneous, including subcooled liquid and superheated vapour regimes, the average void fraction was simply the arithmetic mean of the three chordal void fractions and there was no need to solve Equation (18). Finally, if any two chordal void fractions differed by less than the experimental tolerance, they were both made equal to their average. The algorithm described above was then employed to determine the average void fraction, and the mixture density.

The results of this computation for a blowdown and refill experiment are shown in Figure 6, and are quite reasonable. At the beginning of the experiment, the pipe was filled with subcooled liquid, as indicated by the high density and negative void fraction. At about t = 10 s, a break occurred, and the pipe emptied, as indicated by the low density and unity void fraction. At about t = 30 s, the pipe began to refill. At t = 60 s, the pipe was filled with liquid, as indicated by the high density, and slightly negative void fraction.

**CONCLUSIONS**

The algorithm developed to compute the average void fraction and to obtain information on the phase distribution using a three-beam \( \gamma \)-densimeter is straightforward and very efficient. In spite of limitations imposed by the small number of beams used, the results are expected to be quite accurate for realistic flow regimes encountered in most two-phase flow situations. If a flow regime is expressible as a superposition of three basic ones, the results are exact. These characteristics are confirmed by numerical implementations of the scheme.
Although the algorithm was developed for the special case of the Mark IV version of the three-beam $\gamma$-densitometer in use at WNE, the method is applicable to more general situations. If a $\gamma$-densitometer differs from the present one in number of beams and/or geometry, a suitable algorithm would be straightforward to devise using the present analysis.

REFERENCES


TWO-PHASE NATURAL CIRCULATION EXPERIMENTS IN THE RD-14M MULTIPLE-CHANNEL CANDU THERMALHYDRAULICS TEST FACILITY

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Atomic Energy of Canada Limited
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ABSTRACT

A series of experiments was carried out in the RD-14M multiple-channel test facility, located at the Whiteshell Nuclear Research Establishment, to gain a better understanding of two-phase natural circulation in a CANDU*-typical heat transport system.

In these tests, bidirectional flow was observed in the channels connected in parallel; the flow in some channels continued in the normal direction, in others it reversed.

This paper describes the RD-14M facility, the experimental test matrix and the procedures used. Selected results from experiments carried out at a secondary side pressure of 4.6 MPa are discussed.

A theory is proposed where the onset of bidirectional flow is attributed to a reduction in loop two-phase mass flow, resulting in phase separation in the outlet headers.

The experiments described in this paper were funded by the CANDU Owners Group.

INTRODUCTION

The RD-14 Thermalhydraulics Test Facility at Whiteshell Nuclear Research Establishment (UNRE) was built to provide information on the consequences of postulated loss-of-coolant accident (LOCA) scenarios occurring in a CANDU-typical primary heat transport system. Attention was focussed on large and small breaks occurring at a primary heat transport system header (1-3), secondary system steam side breaks (4), and the effects of decreasing primary heat transport system inventory in a two-phase natural circulation situation (5-7).

Following completion of this experimental program, the RD-14 single-channel-per-pass loop was modified to include 5 channels per pass (8). The multiple-channel loop has been designated RD-14M.

Single- and two-phase natural circulation experiments are being conducted in the new RD-14M facility to examine the interactive effects of multiple channels in a full-elevation model of a CANDU-typical primary heat transport system (9-11).

This paper describes the RD-14M test facility. The two-phase natural circulation test series is described in detail. The overall results are discussed in general terms. Selected results from a test carried out at a secondary side pressure of 4.6 MPa are presented.

FACILITY DESCRIPTION

RD-14M - Multiple-Heated-Channel-Per-Pass Facility

The RD-14M Thermalhydraulic Test Facility (Figure 1) is a full-elevation, full-scale model of a CANDU primary heat transport system. The facility is designed to produce similar mass transit times and pressure gradients as would occur in a CANDU reactor during both forced and natural circulation conditions.

Power for the 10 heated sections is provided by four 2.75-MW DC power supplies. Power distribution in each pass is such that two heated sections together operate at 46% of total power per pass and three heated sections operate together at 54% of total power per pass.

Each of the five horizontal heated sections per pass contain seven 6-m-long fuel element simulators (FES) that represent the seven central pins in a 37-element CANDU fuel bundle and operate at an equivalent heat flux. Each FES is divided into 12...
segments representing the 12 fuel bundles in a fuel channel. Thermocouples in the FES provide data and high-temperature protection. Each heated section is designed with end fitting simulators. Elevation differences between the heated section locations cover the full range of fuel channel elevations found in a CANDU reactor.

Two variable-speed-drive, high-head centrifugal pumps provide flow control in the primary heat transport system. Coolant flows from each pump to an inlet header, where it is distributed to individual feeder lines, providing fluid to each heated section. From the heated sections coolant flows into an outlet header and then into a vertical U-tube steam generator. The steam generators contain a steam separator, preheater and external downcomer. The coolant then flows to the next pass to complete the CANDU figure-of-eight.

A heater in the surge tank provides pressure control in the primary system.

Temperature control in the primary system is accomplished by regulating the pressure in the secondary system. Steam generated by the primary system in the steam generator is condensed by a spray of cold water in the secondary jet condenser. Spray flow as well as boiler feedwater is provided by pumping the water from the jet condenser through a heat exchanger in a closed circuit.

Emergency coolant injection (ECI) is simulated by pressurizing a water-filled accumulator tank with high-pressure nitrogen, or by pumping into the loop from a reservoir tank at atmospheric pressure. Injection is normally to all headers.

For blowdown experiments, a six-inch fast-acting ball valve is connected to either an inlet or outlet header. An orifice plate installed upstream of the valve is used to regulate the rate of loop depressurization.

In partial inventory thermosiphoning experiments, to reduce the fluid inventory in the primary system, draining is performed from outlet header 7. The drained fluid is collected in a calibrated inventory tank so that exact amounts of inventory removal are known. The drain times and drain rates can be varied to determine their effects on the tests.

All instrument signals are fed into an analog to digital converter in the RD-14M control room and recorded by computer. Instrumentation includes turbine flow meters, differential and gauge pressure transducers, resistance temperature detectors and thermocouples. Accurate heated section power measurements are made using a voltmeter/computer system. Fluid density is measured using gamma densitometers.

The ECI system, header interconnect and blowdown lines shown in Figure 1 were not used in the experiments described in this paper.

EXPERIMENTS

Procedure

Before each experiment, the loop is evacuated, filled with distilled water and degassed. The loop is then pressurized to 2 MPa and the differential pressure transducers "zero" checked. The loop is started at low power and pump speed, and brought to the desired operating conditions. The trace heating to the inlet and outlet feeders is then switched on to minimize heat losses. Primary pumps are then shut down and the power supplies set to experimental power levels. The loop is then allowed to stabilize at natural circulation conditions for two hours.

Pre-experiment checks on heated section powers and differential pressure transducer malfunction are then run using the data acquisition system. A final check is performed of all the instruments being scanned in the experiment.

Computer data gathering is then started, and after 60 s, draining from header 7 commences. Five initial drains, each removing 2% of loop inventory, are followed by two drains, each removing 5%. Each subsequent drain removes 10% of loop inventory. The drains continue until two FES in any heated section reach 600°C and terminate the experiment.

Partial Inventory Natural Circulation Test Matrix

The test matrix for natural circulation experiments in RD-14M is shown in Table 1. The matrix covers three power levels and three secondary pressures. The power levels of 160, 100 and 60 kW/pass represent decay power corresponding to, respectively, 3%, 2% and 1% of full power levels in a CANDU reactor.

<p>| TABLE 1: RD-14M NATURAL CIRCULATION TEST MATRIX |</p>
<table>
<thead>
<tr>
<th>Test Number</th>
<th>Power kW/pass</th>
<th>Secondary Pressure MPa(a)</th>
<th>Initial Primary Pressure MPa(a)</th>
<th>Pressurizer</th>
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<tr>
<td>T8805</td>
<td>160</td>
<td>4.6</td>
<td>7.0</td>
<td>out</td>
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<td>T8810</td>
<td>160</td>
<td>1.0</td>
<td>5.0</td>
<td>out</td>
</tr>
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<td>T8804</td>
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<td>0.2</td>
<td>5.0</td>
<td>out</td>
</tr>
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<td>4.6</td>
<td>7.0</td>
<td>out</td>
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<td>out</td>
</tr>
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<td>5.0</td>
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</table>

1 No trace heating
2 Reduced drain rates

A secondary pressure of 4.6 MPa represents conditions immediately following a loss-of-class IV power. A pressure of 0.2 MPa could be the result of a secondary steam line rupture. Because RD-14 single-channel experimental results at these extremes varied considerably, an intermediate value of 1.0 MPa was chosen to help explain the differences. At 1.0 MPa the change in fluid properties also becomes significant.
Most of these tests were performed with the surge tank isolated from the primary system.

RESULTS

In this section the overall results are discussed in general terms. Selected results from an experiment carried out at a secondary side pressure of 4.6 MPa are presented and discussed.

GENERAL BEHAVIOUR

In natural circulation tests carried out in RD-14M, the individual channel flows in a specific pass were unidirectional in the early part of the experiment. Then, depending on the specific test conditions, some channels continued to flow in this direction and some in the opposite direction. In this paper, this type of behaviour is called bidirectional flow.

The transition from unidirectional to bidirectional flow appears to be coincident with flow stagnation above the headers. In tests carried out at low power and low secondary side pressures, bidirectional flow behaviour occurred at high primary fluid inventories, typically 94 to 90% of the primary fluid inventory. In high-power, high-secondary-pressure experiments, bidirectional flow behaviour occurred at reduced primary inventory levels, typically 75 to 80% of the primary fluid inventory.

Figure 2 shows the effectiveness of natural circulation cooling in terms of primary inventory removed before a power trip terminated the experiment.

Between 100 and 90% of loop inventory, overall loop flows increased due to increased buoyancy head. Between 90 and 85% of loop inventory, an oscillatory flow was observed. At 80% of loop inventory, flow above the headers stagnated and remained at this condition until the end of the test. Flow stagnation above the headers was most likely caused by an accumulation of void in the cold leg side of the steam generator tubes.

Channel Flows

Figure 4 shows the volumetric flow rate at the inlet to heated sections 5, 7, 8 and 9 at various primary inventories.
The flow behaviour is very similar to the overall above-header flow, shown in Figure 3, until a primary inventory of 85% is reached. Between 85 and 80% of loop inventory, flow in heated sections 5 and 7 reverses; however, flow in heated sections 8 and 9 continues in the normal positive direction. At \( t \approx 4500 \text{ s} \), flow in heated section 9 stagnates. Stagnation also occurs in heated section 8 at \( t \approx 4900 \text{ s} \). Figure 4 shows that flows in RD-14H change at 80% of primary loop inventory from unidirectional to bidirectional flow in channels connected in parallel. Some channels even stagnate for extended periods of time.

**Fuel Element Simulator Temperatures**

The FES temperatures closely followed the channel flow variations, as expected. Figure 5 shows a top FES temperature close to the midpoint of heated section 9. The FES remained well cooled until the flow in heated section 9 stagnated and the FES was uncovered. An FES temperature excursion occurred, which exceeded 600°C, and the experiment was terminated.

**Void Fraction Histories**

Figures 6 and 7 show the void fraction histories at the inlet and outlet of heated section 9, respectively.

A comparison of Figures 5 and 6 shows that, at times greater than \( t \approx 4500 \text{ s} \), heated section 9 has steam or two-phase in both the inlet and outlet feeders.

**DISCUSSION OF RESULTS**

In the experiments described in this paper, it seems likely that above-header flows in RD-14H decrease and stagnate because the steam generator boiler tubes become vapour-locked.

The reduction in overall loop flows under two-phase conditions results in separation of the phases in the outlet headers. Header stratification causes the thermosiphon above the headers to be broken. However, a thermosiphon develops in channels connected in parallel. Channels with a low buoyant head tend to reverse and bidirectional flow in channels connected in parallel is established.

Bidirectional flow results in steam or two-phase being transported to the previously cool inlet header. Since the overall loop flow above the header is stagnant, the inlet header stratifies and functions as a separator. Vapour rises to the top of the header and vents into the pump discharge lines and the liquid is returned to the heated channels in the normal direction.

A typical RD-14H header is shown in Figure 8. It should be noted that approximately 90% of all partial inventory natural circulation experiments carried out in RD-14H were terminated by temperature trips in channels connected horizontally to the RD-14H headers. This suggests that separation in the headers does occur and results in uncovering of the feeders located in the horizontal positions. Feeder uncovering leads to heated section stagnation and venting of steam or two-phase from both the inlet and outlet of the heated section. The top FES dry-out and a temperature excursion takes place, terminating the experiment.

**SUMMARY AND CONCLUSIONS**

The RD-14H facility has been described in detail. The experimental test procedure and test matrix for partial inventory natural circulation experiments have been described.

Results from these tests show that natural circulation is an effective means of heat removal. The FES remained well cooled until a very significant fraction, in some cases 67% of the primary inventory was removed.

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All partial inventory natural circulation tests in RD-14M exhibited bidirectional flow behaviour in channels connected in parallel.

A theory has been proposed that attributes bidirectional flow behaviour in RD-14M to a reduction in overall loop mass flow resulting in phase separation in the outlet headers.

ACKNOWLEDGEMENTS

Technical assistance in conducting these experiments was provided by J.W. Findlay, B.E. Hood, R.G. Thomson and J.M. Wedgewood. Helpful discussions with G.R. McGee and J. Pascoe are acknowledged. The experiments were funded by the CANDU Owners Group (COG).

REFERENCES


METHODOLOGY FOR ASSESSING CANDU FUEL CHANNEL COOLING FOR SUBCOOLED STAGNANT INITIAL CONDITIONS


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ABSTRACT

A methodology has been developed to assess fuel channel cooling during stratified channel flow for the standing start phenomenon. This phenomenon refers to a subcooled stagnant initial channel condition. This condition may occur in a CANDU fuel channel for events in which coolant density heads are balanced by header-to-header pressure differences and other, such as break, forces. Following the standing start condition, the channel coolant boils and stratifies. Consequently, the upper fuel elements and parts of the pressure tube heat up. The steam produced in the channel flows to the end fittings where it condenses. Eventually, the end fittings heat up sufficiently to allow the steam to reach the vertical feeder section nearest one of the end fittings and to vent. This venting induces a gravity head and a flow which refills the channel, restoring, in general, subcooled channel conditions.

In references 1 and 2, the standing start phenomenon and experiments conducted in the Cold Water Injection Test (CWIT) facility at Stern Laboratories Inc. to study the phenomenon are discussed in detail. The models THERMOSS (Reference 1) and THERMOSS-II (Reference 2) were developed to explain and predict the important physical mechanisms governing the standing start phenomenon. In Section 2 of this paper, the model THERMOSS-II is generalized to THERMOSS-III to account for other effects described in section 2. Section 2 also presents the methodology used to carry out a parametric study of fuel channel heatup over a range of standing start conditions relevant to the accident scenarios of concern in the reactor. The fuel channel codes HOTSPOT-CO (Reference 3) (which from here on will be referred to as HOTSPOT) and ELOCA.Mk4 (Reference 4) used in the parametric study are also described in Section 2. Section 3 presents the results of the parametric study. Section 4 presents a summary of the results and conclusions.

1. INTRODUCTION

This paper presents a methodology for assessing channel heatup during a period of stratified channel coolant following subcooled, stagnant initial channel condition. This condition is referred to as the standing start condition. The standing start condition may arise if forced circulation is lost during small or large loss of coolant accident following the primary circuit refill with the emergency core coolant. The condition may also occur during shutdown cooling mode of operation or for other events in which coolant density heads are balanced by header-to-header pressure differences and by other, such as break, forces. Forced circulation may be lost due to either a loss of class IV power or a deliberate pump trip.

2. METHODOLOGY FOR ASSESSING FUEL CHANNEL HEATUP FOR STANDING START CONDITIONS

This section describes the methodology used to carry out the parametric study mentioned in Section 1 using the model THERMOSS-III and the computer codes HOTSPOT and ELOCA.Mk4.
2.1 THERMOS-III

Following standing start channel conditions, the coolant eventually boils and stratifies with the steam in the upper and water at the lower parts of the channel. The steam flows to both end fittings where it condenses. As the condensation proceeds, the end fitting metal and the water within the end fittings heat up and the steam region expands farther into the end fittings.

THERMOS-III computes the time evolution of the collapsed water level in the channel and end fittings and, hence, the expansion of the steam region, the steam flow and temperature transients and the heatup time using the one dimensional two-fluid conservation equations. The heatup time is defined as the time interval between the onset of bulk boiling in the channel coolant and the arrival of steam at the base of the nearest vertical feeder section. THERMOS-III (for THERmohydraulic Model Of Standing Start) uses most of the assumptions and approximations in the model THERMOS-II (Reference 2) and generalizes THERMOS-II to account for the effects of steam superheating and end fitting heatup by the hot water displaced from the channel.

THERMOS-III uses the following assumptions and approximations.

i) Following channel coolant boiling and stratification, the steam phase partitions in the central channel axial plane and flows equally to both end fittings. The water phase remains stagnant. Thus, the central channel axial plane is a stagnation plane and the channel inlet-to-outlet pressure difference is zero. This assumption is conservative because, in reality, existing channel inlet-to-outlet thermal asymmetry and, hence, non-zero pressure difference across the channel causes a small channel flow and, hence, shorter heatup time. CWIT standing start tests have shown (Reference 1 and 2) that channel inlet-to-outlet thermal asymmetry shortens the heatup time significantly.

(ii) THERMOS-III assumes a horizontal water level in the channel and end fittings. At each instant in time, the level is computed by equating the frictional pressure drop in the steam phase to the hydrostatic head change in the water phase. In this computation, either a uniform or cosine axial power shape is used.

(iii) At a given instant in time, THERMOS-III assumes that the part of the end fitting that is exposed to steam is at the saturation temperature and the part of the end fitting that is submerged in water is subcooled.

(iv) Figure 1 shows the water level at two neighboring instants of time. In a small time interval, the level slightly falls and extends further into the end fittings as the steam from the channel condenses on the end fitting surfaces and heats up instantaneously the shaded parts of the end fittings to the saturation temperature. This mode of end fitting heatup assumes an infinite heat transfer coefficient from steam to the end fitting body and liner tube. This approximation is reasonable in view of the relatively small thickness of the body and tube.

(v) The steam superheats as it flows over the uncovered fuel elements towards the end fittings. This superheating results in a larger latent heat of condensation and, hence, faster end fitting heatup. To compute the steam temperature, the model AMTRACT (Reference 5) is used. This model solves the energy equation for the steam phase and has been validated in References 5 and 6 against channel heatup tests (Reference 6) conducted at Whiteshell Nuclear Research Establishment.

(vi) As the steam region in the channel expands, hot water from the channel is displaced into the end fittings. This water raises the temperature of the end fitting metal mass and the water within the annulus between the end fitting body and liner tube. For given channel coolant conditions in power and pressure, the channel water level drops relatively rapidly following boiling since the channel frictional pressure drop in the steam phase is high. To simplify the computation of end fitting heatup by the hot water displaced from the channel, it is assumed that, following boiling, the channel water level drops instantaneously to its final position, i.e., its position just prior to steam venting through the vertical feeder section. The corresponding hot water displaced into each end fitting displaces an equal amount of cold water from the end fitting and heats up the end fitting mass and the water within.

Mass balance on the steam region and the energy transfer mode between steam and the end fittings described in item (iv) are used to obtain an equation for the rate of change of the steam volume assuming bulk boiling in the channel. This equation coupled with the calculation of the water level in item (ii) is integrated to obtain the volume of the steam region as a function of time, channel power and pressure, end fitting subcooling and channel-end fitting geometrical dimensions. The heatup time is
then equated to the time at which the steam region extends to the end fitting feeder connection port. This time also determines the final channel water level.

Many tests were conducted in a Modified CWIT (MCWIT) facility to study the standing start phenomenon in a channel/end fitting assembly geometry which resembles approximately that in the reactor. This facility and the test results are described in Reference 2. A comparison between THERMOSS-III predictions and the test results is presented in Figures 2 and 3.

![Figure 2 Comparison of final fractional channel water level predicted by THERMOSS-III and observed in standing start tests in CWIT facility with Pickering NGS channel-end fitting assembly](image)

Figure 2 compares the final channel water level predicted by THERMOSS-III for the conditions in the MCWIT tests with that observed in the MCWIT tests. Each bar indicates the uncertainty in the location of the level due to the limited number of thermocouples on the heater element sheaths. Figure 2 shows that, within the experimental uncertainty, THERMOSS-III predicts the level reasonably well. Figure 3 shows the percentage difference between the heatup time \(T\) predicted by THERMOSS-III for the MCWIT test conditions and that observed in the MCWIT tests which exhibited most symmetric conditions, i.e., nearly equal degree of heatup of the two end fittings. For the most symmetric of the symmetric tests (as described in Reference 2) in Figure 3 THERMOSS-III overpredicts the duration by at most 10%. This agreement between the predicted and experimental results is reasonably good.

### 2.2 HOTSPOT Code

HOTSPOT (Reference 3) simulates a 1/12 slice of the 37-element bundle and pressure tube. The code computes the temperature of the fuel elements and the pressure tube in the radial and azimuthal directions using a finite difference technique. The computation accounts for radiative heat exchange among the elements and the pressure tube and convective and conductive heat transfer between the elements and pressure tube and the coolant. A constant pellet-to-sheath heat transfer coefficient is used in the code. The code models pressure tube uniform transverse strain. Fuel sheath strain is, however, not modelled.

### 2.3 ELOCA.Mk4 Code

ELOCA.Mk4 (Reference 4) models the thermal-mechanical behaviour of a single fuel element with axi-symmetric properties. The fuel models in ELOCA.Mk4 are more detailed than in other codes such as HOTSPOT. Specifically, ELOCA.Mk4 models the dynamics of heat transfer between the pellet and sheath. The rate of pellet and sheath heatup depends on, among others, the values of the pellet-to-sheath and sheath-to-coolant heat transfer coefficients. The effective pellet-to-sheath heat transfer coefficient for conductive, convective and radiative heat transfer depends on the pellet-sheath gap size, amount and composition of the filling gas and fission gas, etc. As the pellet and sheath heat up, the sheath undergoes stress and, hence, strain under the fission gas pressure. This strain causes the sheath to lift off the pellet reducing the conductive/convective part of the pellet-to-sheath heat transfer coefficient and tending to increase the pellet temperature and reduce the sheath temperature. The strain also reduces the gap gas pressure tending to reduce sheath stress and, hence, further strain. At higher temperature, radiation from the pellet to the sheath and, hence, this component of the pellet-to-sheath heat transfer coefficient as well as
radiation from the sheath to the surroundings becomes significant. Depending on the reactor power, fuel burnup and coolant pressure, the above effects may be self-limiting. Between about 600°C and 1000°C temperature, the zircaloy sheath undergoes a phase transition from α to β state tending to reduce the sheath temperature. Before any terminal temperature and strain are reached, the fuel may fail by, among other mechanisms, uniform plastic hoop sheath strain in excess of 5% and by stress-assisted thermal diffusion of beryllium into sheath grain boundaries at brazed appendages of spacers and bearing pads causing intergranular cracking at temperatures above 750°C.

The ELOCA.MkA predicted sheath strain, internal element gas pressure, fuel centre line temperature, sheath microstructure and thickness of zirconia and oxygen-stabilized α-Zr layers on the sheath have been compared well with experimental results (Reference 7). ELOCA.MkA models: diametral sheath strain, a dynamic effective fuel-to-sheath heat transfer coefficient (dynamic gap conductance) which accounts for both conduction and radiation between the pellet and sheath in contrast to a constant pellet-to-sheath heat transfer coefficient used in HOTSPOT, the effects of the metal-water reaction on heat generation, zircaloy property change due to α-to-β phase transition and fuel failure criteria. However, radiation heat transfer from the sheath surface is not modelled in ELOCA.MkA.

2.4 Parametric Study of Channel Heatup

The models described in Subsections 2.1 to 2.3 were used to carry out a parametric study of fuel channel heatup over a range of standing start conditions in channel power (from 0.5 to 5 percent of nominal), pressure (from 0.2 to 5 MPa) and subcooling (20 to 120°C) relevant to the reactor accident scenarios of concern.

In the parametric study, THERMOSS-III was used to compute fuel channel heatup time. To obtain an upper bound to pellet, sheath and pressure tube temperatures, the channel with the highest bundle power (viz., 933 kW) and lowest possible channel power (which turned out to be 7.3 MW) was selected. For a given standing start condition, the predicted heatup time determined the maximum fuel pellet, sheath and pressure tube temperatures and strains predicted by HOTSPOT and ELOCA.MkA.

For given initial thermohydraulic channel coolant conditions (i.e., bundle power, coolant temperature and sheath-to-coolant heat transfer coefficient), HOTSPOT was used to obtain the initial steady state temperature throughout the bundle at the time of channel coolant stagnation. A constant bundle power of 933 kW at 100% full power corresponding to maximum linear element power of 59.4 kW/m was used. The coolant temperature was chosen to be the temperature at the time of onset of bulk boiling in the channel, i.e., the saturation temperature. To obtain the highest temperatures for a given standing start condition, the fuel temperature was computed in the steam stagnation plane. In this plane, convective and conductive heat transfer are negligibly small and, hence, this heat transfer from sheath to coolant was ignored. Thus, in the HOTSPOT runs, only radiative heat transfer was modelled. HOTSPOT was run for each given set of conditions to obtain the fuel pellet sheath and pressure tube temperature and pressure tube uniform transverse creep strain transients.

For a given channel power and pressure, ELOCA.MkA was used to compute the heatup of a maximum powered fuel element and assess sheath integrity. The code ELESIM.MOD10 (Reference 8) was used to calculate the fission gas and, hence, pellet-sheath gap gas inventory during normal operation. In each ELOCA.MkA simulation, radiation from sheath to the surrounding was accounted for at each instant in time by using the effective heat transfer coefficient computed by HOTSPOT for that simulation. This treatment of radiation is reasonably good for the temperature range of interest. A linear power of 55.4 kW/m corresponding to a fuel burnup of 140 MW·h/kgU was used. This burnup is known to correspond to the highest possible internal gas pressure and linear power.

3. RESULTS OF PARAMETRIC STUDY

3.1 THERMOSS-III Predictions

Figure 4 shows the THERMOSS-III predicted final fractional channel water level (i.e., at the time just before steam venting) as a function of channel power at various channel pressures. In THERMOSS-III, the water level is determined by momentum balance and, hence, it is independent of subcooling. For a given

![Figure 4](image-url)
pressure, the level decreases with power due to increased steam generation rate and flow rate and, hence, frictional pressure drop. For a given power, the level increases with pressure as the steam specific volume and, hence, velocity and frictional pressure drop decreases.

Figure 5 shows the channel heatup time as a function of channel power at various initial fluid subcooling and at 0.2 MPa channel pressure. Similar results were obtained at higher pressures. Figure 5 shows that for a given pressure and subcooling, the heatup time decreases with increasing power as the steam generation rate and, hence, end fitting heatup rate increases. For a given pressure and power, the heatup time increases with increasing subcooling as more steam needs to be produced to heat up the end fitting mass to the saturation temperature. Other results shown that for a given channel power and subcooling, the heatup time decreases with increasing pressure as the channel water level and, hence, steam generation rate is higher at higher pressure.

Figure 7 shows the pressure tube temperature as a function of initial fluid subcooling at various channel power and pressure. At the pressures (up to 5 MPa) and pressure tube temperatures considered, no significant pressure tube strain is predicted.

3.2 HOTSPOT Predictions

Figure 6 shows the HOTSPOT predicted outer and middle element sheath temperatures as functions of initial fluid subcooling at various channel power and at 0.2 MPa channel pressure. Similar results were obtained at higher pressures. The trend in temperature with power, pressure and subcooling is similar to that of the channel heatup time described in Section 3.1. At higher temperatures, the middle element heats up to higher temperature than the higher powered outer element because the outer element radiates to the colder pressure tube.
3.3 ELOCA.Mk4 Predictions

Figure 8 shows the ELOCA.Mk4 predicted sheath failure map as a function of initial channel fluid subcooling and channel power and pressure. Each curve in Figure 8 represents threshold channel thermohydraulic conditions for sheath failure by the mechanism indicated. The dashed part of each curve represents unrealistically high subcooling for the pressure indicated and is drawn only to show trend in the failure curve. For a given channel pressure, the sheath is predicted to fail for values of subcooling and power in the region above the curve and to remain intact for the values of the parameters below the curve. Specifically, at 0.2 MPa, for a given power and at sufficiently high subcooling, sheath is predicted to fail first by uniform strain in excess of 5%. At 2 MPa, for a given power and at sufficiently high subcooling, sheath fails first by beryllium diffusion assisted cracking. At 5 MPa, no sheath failure by any mechanism is predicted for any of the conditions of interest.

4. CONCLUDING REMARKS

A parametric study of channel heatup and fuel behaviour over a range of standing start conditions in channel power, pressure and initial fluid subcooling was carried out using THERMOSS-III, HOTSPOT and ELOCA.Mk4. THERMOSS-III modifies previous models of standing start to account for steam superheating and end fitting heatup by the hot water displaced from the channel.

In the parametric study, THERMOSS-III was used to predict the (final) channel water level at the time of steam venting and channel heatup time. HOTSPOT was used to compute the fuel pellet, sheath and pressure tube temperatures and pressure tube uniform transverse creep strain at the end of channel heatup time. ELOCA.Mk4 was used to assess sheath integrity.

For all standing start conditions relevant to the reactor, the reactor power is below 4% and the subcooling is less than 100°C. High subcooling (above 80–90°C) is associated with low pressure (less than 0.5 MPa) and low power (less than 3%). For all these conditions, the sheath and pressure tube temperatures are predicted to remain within acceptable limits and no sheath failure is predicted.
REFERENCES


Session 13:

Fusion and Physics

Chairman:

D.P. Jackson, AECL-CRNL
ABSTRACT

An international design team comprised of members from Canada, Europe, Japan, the Soviet Union, and the United States of America, are designing an experimental fusion test reactor. The engineering and testing objectives of this International Thermonuclear Experimental Reactor (ITER) are to validate the design and to demonstrate controlled ignition, extended burn of a deuterium and tritium plasma, and achieve steady state using technology expected to be available by 1990. The concept maximizes flexibility while allowing for a variety of plasma configurations and operating scenarios. During physics phase operation, the machine produces a 22 MA plasma current. In the technology phase, the machine can be reconfigured with a thicker shield and a breeding blanket to operate with an 18 MA plasma current at a major radius of 5.5 meters. Canada's involvement in the areas of safety, facility design, reactor configuration and maintenance builds on our internationally recognized design and operational expertise in developing tritium processes and CANDU related technologies.

INTRODUCTION

The genesis of the International Thermonuclear Experimental Reactor project (ITER) was a Soviet proposal for international cooperation in the development of nuclear fusion tabled at the 1985 Geneva talks between the USA and the USSR. The following year in Reykjavik, the leaders of these two nations took up the theme and included the European Community and Japan in the task of designing a thermonuclear experimental reactor. ITER, latin for "the way", was created under the aegis of the IAEA. Canada was formally admitted in July 1988 under the sponsorship of the European Community and has a permanent chair on the EC team. The joint design team consists of 40 members, 10 from each member nation, in addition other resources in the "home team" of each partner country contribute to the design effort through homework tasks and workshops. A management committee consisting of four directors, one from each of the major partners, runs the operations at the design centre in Garching West Germany, and reports quarterly to the ITER council in Vienna.

The Quadripartite Initiative Committee met in March 1987 and agreed on the project guide lines. In October 1987 the objectives were established and in April 1988 the ITER Council held it's first meeting. From May through September 1988 the first work session was held at the Max Planck Institut Für Plasmaphysik in Garching. The most recent joint work session took place in Garching during February and March 1989. Homework tasks performed by the major partners were reviewed and critical design issues, to be addressed through the balance of 1989, were identified.

There have been a few major changes in the organization of the team since the ITER 1988 summer session. Most of the original team members attended the 1989 winter session. However, each of the partners have added individuals to the list of part-time contributors who will be on-site at some point in the next summer session. As a result, besides the approximately 40 full-time team members, there will be an additional 8 to 12 part-timers from each of the four partners. This does not include contributions from home-team participants which have already been committed.

It should be noted that R. Stasko, the Canadian participant, has been moved from the Basic Device Engineering project group to the Nuclear Engineering Project group. This move, agreed to by R. Toschi, manager of the EC participants, reflects the fact that most of the Canadian R&D and design efforts contribute to the activities of this group.

The project schedule outlines an ambitious program with a 31 month concept and design phase ending in November 1990. Procurement and construction phase commence December 1990 with the start of commissioning in 2002. Operation is divided into two separate phases, physics, from 2002 to 2006 and a technology phase starting in 2008 (after some modification to the machine) and running until 2019.

The terms of reference and objectives for the project are:

1. Define technical characteristics for ITER and do the design work necessary to establish a conceptual design.
2. Define research and development needs, resources and scheduling requirements to realize the design of such a device.
3. Define site requirements for ITER and perform a safety and environmental analysis.
4. Perform specific validating research and development work in support of design activities.

The ITER mission is to demonstrate the scientific and technological feasibility of fusion power. To accomplish this mission, ITER must demonstrate controlled ignition and an extended burn of a deuterium and tritium plasma, ultimately in steady
state, as well as demonstrating essential technologies such as superconducting magnets. To be considered successful, the testing phase must also support the design and its extrapolation to commercial power fusion machines. (1)

To put this mission into context with reality the plasma must reach break-even conditions where $Q$, the ratio of fusion power to plasma heating power, equals unity. The Lawson diagram (figure 1) shows current experience. There is a long way to go to reach the ITER objective of demonstrating commercial fusion power.

![Lawson Diagram](image)

**FIGURE 1: LAWSON DIAGRAM**

**BASIC DEVICE DESIGN CONCEPT**

**Configuration.** ITER is a magnetic confinement fusion device of the Tokamak type. The plasma is contained in a "magnetic bottle" within a vacuum chamber such that the plasma under normal operation will not contact the sides of the chamber. A set of toroidal and poloidal superconducting magnets, (figure 2) keep the plasma suspended in the vacuum chamber and the space between the plasma and the chamber wall acts as a thermal insulator. About 80% of the plasma's energy is carried away by high energy neutrons produced in the fusion reaction. This neutron energy is captured in a shielding blanket in the form of heat. The plasma chamber (19 m diameter by 10 m high) consisting of several segments welded together, (figure 3) is supported inside a set of 16 toroidal superconducting magnets. A set of six superconducting poloidal coils 23 m in diameter is supported outside the TF coils giving rise to an assembly, including support structure, of about 25 m in diameter by 20 m high (figure 4). The centre solenoid portion of the P.F. coil set is self supporting and sits in the hub or hole in the centre of the toroid formed by the inner legs of the T.F. coils.

![Magnet System Assembly](image)

**FIGURE 2: MAGNET SYSTEM ASSEMBLY**

Current designs call for the tokamak to be housed in a cryostat containing helium for magnet cooling. The cryostat also acts as a secondary containment boundary for the plasma. This entire assembly is set in a concrete cylinder 25 m in diameter and 27 m deep. The cylinder provides structural support as well as biological shielding for external equipment, however this concept is under review and it may be set back about five meters from the machine to improve access for installation and remote maintenance, thus alleviating afterheat problems as well as structural difficulties. (2)

![Vacuum Chamber Segments](image)

**FIGURE 3: VACUUM CHAMBER SEGMENTS**

The region between each of the TF coils provides space for 16 access ports, one per sector, in the equatorial plane. Port height is optimized with careful location of the PF coils. Each port is assigned to one or more of the following: test modules, current drive, plasma heating, fuelling, diagnostics, and maintenance equipment.
The size of ITER is a significant increase over existing machines in order to provide reasonable plasma physics certainty but this represents not only a major undertaking but a large technological uncertainty (e.g., superconducting magnets). A comparison of some of the performance parameters is presented in figure 5.

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Maintenance. The internals of the plasma chamber consist of test modules, shielding for the magnets, first wall or heat protective surfaces and heat extraction structures. These components are to be maintained by robotic devices which have the capability of in-situ maintenance or removal of components from the Tokamak to a hot cell for repair or disposal. From the point of view of assembly and maintenance, components fall into 3 categories; semi-permanent, medium lived, and short lived. The semi-permanent components will be replaced when they fail if they inhibit the safe operation of the machine. Failure of semi-permanent components implies a major shut down of one to two years. Medium life components such as blankets may be routinely removed to accommodate changes in the machine geometry. The short life in-vessel components such as divertors, first wall tiles, and guard limiters which face the plasma could see frequent damage and need to be "easily" removed to minimize reactor down time. The first wall reference concept specifies a carbon composite armour tile, mechanically attached or bonded to the water cooled heat removal system. The operating load on the first wall is approximately 1.0 MW/m\(^2\) and the ITER engineering test mission requires the equivalent of one full reactor year of operation at this fluence level to prove suitability of tile material. ITER maintenance and design requirements, as listed below, are very ambitious from an engineering standpoint.

- ITER should be fully remotely maintainable but with provision for hands on maintenance everywhere possible.
- ITER is to operate in staged operation phases with variable blanket and divertor geometry possible.
- Maintenance of short life and high failure rate components should be possible without moving other components or in any way disturbing the reactor's internal and external environment.

These requirements have a profound influence on both the design of components and their integration into an operable facility. To ensure the robotics are up to the task, representative parts of the machine will be assembled using remote handling devices in a benign environment. A typical magnet installation is shown in figure 6.

Machine Operation. There are two operational phases required to achieve ITER's mission. The physics phase has three stages: zero, low, and high activation. During the first and second stages general physics studies of H/D plasmas will investigate energy confinement, particle exhaust, and disruption characteristics and control. The third stage will focus on ignition, controlled burn, and driven operation which will lead into the technology phase. Development of ignited plasma conditions without disruptions, and the characterization of divertor performance for technology phase operation, will be key issues for the five-year 15000-shot physics phase. The physics phase configuration will employ a thin shield blanket allowing for a large plasma cross section.
and having sufficient volt-seconds to achieve 22 MA current by full inductive operation. With the use of external current drive devices, larger plasma currents are envisioned. At the end of the physics phase, the internals of the machine will be replaced with a configuration compatible with the technology phase mission.

The technology phase will consist of several stages. The first, about three years in duration, will be devoted to concept verification tests of power reactor blanket options, first wall and divertor performance. Assuming the first stage of testing is successful, the technology phase would be extended for about seven years to perform long term blanket testing and proof-of-concept, as well as providing materials damage information. Safety related transient tests would be conducted near the end of the operating period.

SYSTEMS

The myriad of systems required to operate a Tokamak must be integrated so as to minimize complexity and cost, and optimize ease of operation and maintainability. Some of the more controversial systems are considered below.

Fueling. Fueling for ITER will be gas puffing in the region of the upper divertor. Pellet injection for density, ramp-up and profile control, which are more advanced concepts, might well be considered for later stages of the program. The injector velocity requirements are in the order of 1 to 2 km/s with a repetition rate of 1 to 3 Hz. High speed injectors > 2 km/s may be used if required with the proviso that the technology can be developed. For injection deep into the plasma, higher speeds are being investigated in the USA using compact toroids.

Torus exhaust pumping speeds of 1000 m³/s or higher at 10⁻⁶ Pa will be required. The reference pumping system calls for compound cryopumps or turbomolecular pumps.

Current Drive and Heating. The current drive and heating systems must satisfy the physics requirements for bulk plasma heating to ignition and steady-state non-inductive operation. In addition, they should provide plasma start-up assistance, profile control and current ramp-up. Given that the technology can be developed, any of the systems proposed could heat ITER to ignition. However the goal is that the same system provide both heating and current drive. Four systems have been proposed: EC (electron cyclotron), IC (ion cyclotron), LH (lower hybrid), and NB (neutral beam injection). Three options were proposed to meet these requirements. In each case, LH drives current in the outer region of the plasma and EC initiates plasma and provide disruption control. Current drive in the centre of the plasma would be provided by either NB, EC, or IC. The preferred option using only LH, and EC is the simplest, the most credible option employs neutral beams as well as LH and EC. These systems are power consumers; for the selected option, up to 280 MW for 100 MW injected into the plasma. A main feature of the most credible option (NB) is its size - much larger than that of the tokamak. The neutralizer, 8 m in length is to be located with unrealistic accuracy in a 40 m long duct. Twelve ion sources per beam, each 4 m in diameter are housed in a concrete bunker 30 m wide, 35 m long and 26 m high. There are 3 such beam lines for ITER each having similar vacuum pumping requirements as the entire torus.

Blankets. The tritium-breeding blanket concepts have been narrowed to three options; aqueous lithium salt, solid ceramic, and lead lithium eutectic. All concepts are to be low-temperature low-pressure systems using 316 L austenitic stainless steel for the structural and coolant piping material. For a viable fusion power reactor there is a need to investigate reactor-relevant breeding blanket concepts as well as high-grade heat extraction capabilities of blankets. There is an intent to design full sector blankets with the cross-section and shape of a 14 m long banana. Moving such a component into place and maintaining it remotely will be a challenge.

ITER TECHNOLOGY ISSUES

In the aftermath of the most recent work session (spring of 1989, at Garching) the level of overall design-related activity is expected to increase, as each of the four major participants addresses the significant technological impediments and knowledge shortfalls that will have to be overcome before detailed design commences. These are some of the major issues which will dominate the design homework performed by national home teams, with preliminary results to be reported during the 1989 summer session.
Divertor Technology. At present, there is no known divertor concept which will survive the heat load associated with the technology phase of ITER operation. The plasma facing surface will have to survive average heat loads of 5.0 MW/m² with a peaking factor of two or greater. No design based on the existing materials database for fusion can assure that divertors will survive more than a few days in this environment. Since this has been recognized as a critical technology issue for design of the machine, ITER project management has charged the designers to explore alternate concepts that might still be incorporated into the ITER baseline design. Several options are being developed for presentation and evaluation during the 1989 summer session.

Magnet Reliability. This is an important design issue, as the reliability and availability of the large TF and PP coils required for ITER cannot be predicted with sufficient confidence. If any of these superconducting coils requires replacement during the facility operating life, a downtime of several years is anticipated. This could be reduced to several months if replacements are incorporated into the design configuration (ie: in-situ spares) or if they are at least on-site. Establishing detailed failure modes and frequencies can have a large impact on machine cost. At time of writing, no superconducting coils of the required size and current capacities are in operation, and smaller coils have limited experience, or have not performed with the desired reliability. However, the operating experience base at Tora Supra (Cadarache, France) and T-15 (Kurchatov Institute, USSR) is expected to provide design-relevant guidance over the next few years as these superconducting machines begin their experimental programs.

Disruption Forces. Forces on in-vessel components resulting from the disruption of a 25 MA plasma current will be in the range of 10-100 MN over a time span of a few milliseconds. At present, designing fastening mechanisms for blanket modules which would survive such forces, yet which would enable blanket replacement, appear problematic. Present concepts involve the use of hydraulic locking wedges or inflatable bladders which will fix the modules firmly in place after remote installation. The vulnerability of such systems to failures, or degradation due to neutron bombardment must be assessed before a reference concept is selected. Designers from all the home teams are addressing this issue.

Tritium Inventory. In order to limit offsite doses to 10 rem or less (at a 1 km exclusion radius), the ITER safety group has specified a design target that no system failure mode can result in releases of more than 200 g of tritium (as HTO) from the facility. This is not an insurmountable goal for system designers. For example, the whole tritium systems building is not expected to contain an inventory of more than 350 g and possibly less. Graphite armour tile material can absorb several kilograms of tritium over the operating life of the facility and is presently envisioned as part of the first wall design. Mechanisms will have to be found to limit this uptake, to release it under controlled conditions, or to ensure that failure modes which release part or all of this inventory will leave the containment intact.

Helium Transport. Although there is a large margin of uncertainty, physicists have updated calculations on the rate of helium transport from the plasma centre to plasma edge, which indicate that this transport could be slower than originally envisaged. In order to ensure that helium does not poison a self-sustaining fusion ignition, it may be necessary, for example, to increase the exhaust pumping by as much as a factor of ten. Experiments with helium at existing tokamaks may help in clearing up this uncertainty.

CANADA'S DESIGN/R&D CONTRIBUTIONS TO ITER

Design contributions for Canadian home-team participants are still evolving, but are expected to concentrate in the areas of safety system engineering, blanket design, tritium systems, remote maintenance and facility layout. Approximately 8 man years of effort have been identified by the EC as the Canadian component. There has already been significant contribution to the design of the tritium purification systems; to assembly and maintenance of the torus configuration; and to the layout and equipment location of torus systems in the reactor hall.

We have been involved in the design and development of ceramic and aqueous salt driver blankets for ITER. Radiolysis and other chemistry issues for the aqueous salt, and irradiation of a ceramic blanket test module, are our major R&D contributions to ITER.

Design tasks in the tritium technologies include design of structures and air handling systems to maintain tritium leak-tightness. In addition Canada is involved in the preparation of design documents for an aqueous salt blanket tritium extraction system. Other Canadian activities include design of a tritium Isotopic separation system, and design and integration of tritium systems in general.

Canada's expertise in tritium technology and fission plant design will be employed in the development of atmospheric clean-up and processing systems, as well as the handling and disposal of tritiated wastes. The influence of systems containing tritium on plant arrangement and facility design will be another opportunity area.

By hosting atmospheric release studies with other nations, Canada has taken a lead role in the investigation of the environmental effects of tritium. These activities and our tritium dosimetry capability positions us to make useful contributions to the safety studies associated with ITER.
Our CANDU experience in design and operation will be drawn on in areas of engineering assessment, reliability, safety and costing studies.

The need for remote maintenance skills on ITER assures Canada a role both in robotics design and influencing the configuration of the Tokamak. Canada has placed engineers from Spar in key remote handling design roles in both JET and NET. In addition Wardrop Engineering has an engineer stationed at NET developing remote maintenance designs. We are involved in the design concept of an in-vessel vehicle which will perform in-situ repairs, as well as remove components from the Tokamak for ex-vessel repair.

In the critical area of first wall and divertor materials, Canada is providing key data on graphite and modified graphite properties on exposure to plasmas, including tritium retention and outgassing. This work is being conducted at the University of Toronto.

CONCLUSIONS

It should be noted that although Canadian contributions have stemmed from areas of R&D and engineering activities associated with the CFFTP fusion program, contributions from other research and engineering agencies in Canada are anticipated later in the design phase. This process will be furthered by a continuation of the broadening of Canada's technical interests, as well as recognition by the International community that Canada's technical contributions are worthwhile; all the more so as design effort expands during the latter phase of the project.

The ITER project was conceived as a study with the design phase scheduled for completion in 1990. It remains to be seen whether the four parties will proceed with joint construction of ITER beyond 1990. The course of events will hinge on both technical merit and the international political climate. If ITER becomes a real facility, Canada will most assuredly be on the team and make a notable contribution.

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ABSTRACT

The fusion-blanket program at Chalk River Nuclear Laboratories concentrates on fabrication development and irradiation testing. Canadian efforts to develop a solid-breeder blanket are focused on the sphere-pac concept. The spheres, fabricated by an extrusion-based process, are strong, survive rapid heating cycles, and have an average grain size of about 1 micrometre. In the recently completed CRITIC-I vented-capsule irradiation of Li20, a total lithium burnup of 1% was achieved, and 2100 curies of tritium collected, over the 21-month irradiation. We are also participating in the development of an aqueous lithium salt blanket concept. Gamma-cell radiolysis of lithium salt solutions has been completed in support of the concept, and lithium hydroxide has been chosen for further in-reactor studies. Canada is participating in the BEATRIX-II project, which will test Canadian-fabricated lithium zirconate spheres in FFTF, Hanford, with Japan and the US as the other partners. We are also members of the JAERI (Japan) Tritium Project, to improve blanket neutronics data.

INTRODUCTION

Canada is making significant contributions to the worldwide effort to develop fusion as an energy source. The lead agency in the Canadian program is Atomic Energy of Canada Limited (AECL) through the National Fusion Program, and major components are the Tokamak at Varennes, Quebec and the Canadian Fusion Fuels Technology Project (CFFTP) in Toronto, Ontario.

A program cofunded by AECL and CFFTP, and focusing on fusion-blanket technology, began at Chalk River Nuclear Laboratories (CRNL) in 1983. The program was based on AECL's generic expertise in ceramics, irradiation testing and tritium technology, all key aspects in developing a breeder blanket. This paper updates the status of the program (1). The major components discussed are lithium-ceramic fabrication development, in-reactor testing, and work on an alternate concept, the aqueous lithium salt blanket (ALSB). Particular emphasis is given to the recently completed CRITIC-I experiment (CRITIC - Chalk River In-reactor Tritium Instrumented Capsule), now in the post-irradiation examination phase. Additionally, progress in international aspects of the program is outlined.

FABRICATION DEVELOPMENT

Canadian efforts to develop solid-breeder blankets focus on sphere-pac. AECL has substantial generic expertise through its experience with the concept in fission-fuel development (2). Sphere-pac blankets for fusion application require large numbers of tritium-breeding lithium-ceramic spheres. These are randomly packed into dense beds within the blanket modules incorporated into a fusion reactor. Because of their relatively small size, typically about 1 mm diameter, large numbers are required to fill a blanket bed. For example, a small-scale irradiation test on a 1-litre sphere bed requires in excess of 1 000 000, 1 mm diameter spheres. A full-size sphere-pac breeder blanket would be about 300 000 litres in volume, requiring more than 3x1011, 1 mm diameter ceramic spheres.

Work is under way at CRNL to develop technology for high-speed production of lithium aluminate and lithium zirconate spheres. Lithium aluminate is a well-established candidate for blanket application. Lithium zirconate has recently shown attractive low-temperature tritium-release properties during in-reactor tests (3). Emphasis in the fabrication program is on achieving the high production rates required to produce the large numbers of spheres necessary for a blanket module. Pilot-scale equipment capable of producing 1 mm spheres at the rate of 250 000 spheres per minute is currently being installed and commissioned. To date, more than 1 000 000 prototype lithium aluminate spheres have been produced at CRNL by the production process, based on extrusion. The spheres are mechanically strong, survive rapid heating cycles and have an average grain size of about 1 micrometre. In particular, the small surface-to-centre temperature differential (AT, less than 20°C), inherent in the sphere concept, provides one of its major advantages (4). Long-term irradiation testing (5) has identified cracking due to large ΔTs as a potential problem for blanket designs based on pellets or monolithic assemblies.

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In addition to the development of a production process, good progress has been made in lithium-based powder synthesis. While lithium aluminate is available in industrial quantities via custom order, only small amounts of lithium zirconate have been produced worldwide. At CRNL, the precipitation technique has shown itself to be not feasible for quantity production. Current efforts focus on the solid-state reaction between zirconia and lithium carbonate.

The thermal, mechanical and tritium-release performance of spheres prepared by the extrusion-based process will be characterized in detail. An in-reactor CREATE (Chalk River Experiment to Assess Tritium Emission) test has just been completed on lithium aluminate spheres. In addition, high-flux irradiation tests are planned in the CRITIC facility of the NRU reactor at CRNL, and in the Fast Flux Test Facility (FFTF) at Hanford, Washington, as part of the BEATRIX-II program (International Breeder Exchange Matrix).

IRRADIATION TESTING

Two types of ceramic-blanket irradiation testing are performed at CRNL: CREATE tests, in which tritium-release information is obtained via post-irradiation annealing, and CRITIC tests, which permit on-line monitoring of tritium release as the irradiation progresses. Nine CREATE tests have been performed on lithium oxide and lithium aluminate, examining the effects of temperature, microstructure, sweep-gas composition and capsule material (oxygen potential) on the amount and form of tritium release. Full details of these tests are given elsewhere (6-8).

The recently-completed CRITIC-I test in the NRU reactor at CRNL has produced a substantial amount of reactor-relevant data. Anular Li2O pellets (30 mm I.D. x 40 mm O.D.) were fabricated for CRITIC-I at Argonne National Laboratories under the BEATRIX-I program coordinated by the International Energy Agency (IEA). The fabrication techniques and characterization results for CRITIC-I are described in detail elsewhere (9-12). The total weight of Li2O irradiated was 103 g, density 91.5% of theoretical, original isotopic content was 1.53 wt% 6Li and average grain size was 50 µm. The temperature of the ceramic was controlled by varying the composition of a He-Ar insulating gas layer (gap gas). Twelve thermocouples were located on the Li2O stack. The capsule provided approximately uniform ceramic temperatures (±50°C) in the range 370-850°C. Full details of the capsule design are given elsewhere (9-12). A sweep gas flowed over the inner surface of the annular pellets to an analysis train which provided on-line monitoring of the tritium release. The first ionization chamber measured sweep-gas composition and capsule material (oxygen potential) on the amount and form of tritium release. Full details of these tests are given elsewhere (6-8).

The irradiation test plan was divided into the following phases:

Phase 1
Conditioning the Li2O to 650°C - ceramic heated slowly to 650°C over 2 weeks.

Phase 2
Testing in the range 400-620°C - temperature step-change tests carried out, He sweep gas for 6 weeks and He-0.01% H2 for 10 days.

Phase 3
Conditioning the Li2O to 850°C - ceramic temperature increased to 850°C and held for 2 days.

Phase 4
Testing in the range 400-850°C - temperature step-change tests with He, He-0.01% H2, He-0.1% H2, and He-1% H2 sweep gases.

Phase 5
Testing with H2O in the sweep gas, as described below.

Results from these phases, which include 95 temperature-transient tests, have been reported in detail (9-12). Figure 1 shows the results of one typical temperature transient, with tritium release shown as a function of temperature. This transient is a Phase 5 test, with moisture in the He-0.01% H2 sweep gas. Under these conditions most of the tritium was released as HTO. A final burnup of 1% total lithium (58% 6Li) was achieved during the 21-month irradiation. The tritium generation rate was initially 5 Ci/day, dropping to 3.5 Ci/day after 461 full-power days of irradiation. Total tritium recovered was 2100 Ci.

Tritium release was found to be controlled by a surface-desorption release mechanism, and the influence of impurities (H2O or CO2) in producing tritium as HTO during the initial irradiation phases was clear (10). The amount of tritium recovered in the reduced form (HT) increased from an initial value of approximately 50% with pure He sweep gas, to 99% with He-1% H2. The increasing H2O concentration in the sweep gas also reduced the time constants for tritium release (tritium residence time in the Li2O). For the last 5 months of the irradiation, a small defect in the reactor insert resulted in ~300 ppm moisture in the sweep gas. Post-irradiation examination showed the Li2O to be hard and durable, with only modest cracking. X-ray diffraction identified Li2O, with traces of LiOH. The post-irradiation tritium inventory was found to be less than 1 Ci, about 6 hours' tritium production. Figure 2 shows a scanning electron micrograph of a fracture-surface of the irradiated Li2O. Of note is the increased porosity (20%) compared with 10% in the as-fabricated material (10), confirmed by mercury porosimetry. Figure 3 shows a higher-magnification micrograph; the bubble-like features are tentatively attributed to helium generated by the interaction of neutrons with lithium.
The CRITIC-I data show that tritium release is a complex process. Increasing the H₂ concentration in the sweep gas reduced the time constant for the tritium-release peaks, indicating a surface desorption-controlled release process. A desorption activation energy of 125 to 140 kJ/mol was obtained, in good agreement with previous results from post-irradiation studies. However, the initial part of the tritium-release peaks for temperature-increase tests were not consistent with simple first-order desorption kinetics. Kopasz et al. suggest (13,14) that the desorption activation energy changes with fractional coverage of the surface with tritium, and is in agreement with some of the CRITIC-I data. This and other models are still being investigated.

Preparations are currently under way for CRITIC-II. This will be an irradiation in the NRU reactor of CRNL-fabricated lithium zirconate spheres. The test will feature on-line tritium analysis, as for CRITIC-I. The scheduled insertion date is 1989 December.

AQUEOUS LITHIUM SALT BLANKET (ALSB)

An alternative to the solid ceramic breeder blanket is an aqueous solution containing approximately 2 mol dm⁻³ 6Li⁺. The ALSB is a simple, low-temperature, low-pressure concept (1), which is a candidate for the ITER/NET driver blanket. Solubility restrictions limit the choice to three lithium salts: the hydroxide, the nitrate and the sulphate. The ALSB has an advantage in that it draws on established technologies used in fission reactors and in tritium removal from aqueous solutions. However, it is recognized that radiolysis effects could form potentially-explosive mixtures of oxygen and hydrogen, as well as degrading the counterion. At CRNL, we are studying the radiolysis of lithium salt solutions, to select the most appropriate lithium salt for the ALSB concept and also to evaluate the use of an initial excess of hydrogen to suppress the radiolytic production of hydrogen and oxygen.
FIGURE 2 SCANNING ELECTRON MICROGRAPH OF IRRADIATED LITHIUM OXIDE FROM CRITIC-I. NOTE AREAS OF POROSITY.

FIGURE 3 SCANNING ELECTRON MICROGRAPH OF IRRADIATED LITHIUM OXIDE FROM CRITIC-I, SHOWING FEATURES TENTATIVELY IDENTIFIED AS He BUBBLES.
When degassed solutions containing either 3.6 mol dm\(^{-3}\) LiOH or 1 mol dm\(^{-3}\) Li\(_2\)SO\(_4\) were gamma irradiated (dose rate 1.4 Gy s\(^{-1}\)) with \(>3.5\times10^5\) Gy, no oxygen was detected and only a small, steady-state concentration of hydrogen (1-2.5\times10^{20}\) molecules kg\(^{-1}\)) was observed. In this case, trace impurities reacted with the hydroxyl radicals formed to leave an excess of hydrogen, which then suppressed further radiolysis of the water. On the other hand, when 5 mol dm\(^{-3}\) LiNO\(_3\) was gamma irradiated (<5x10^4 Gy) steady-state conditions were not observed: i.e., the formation of products was linearly dependent on dose over the range investigated. The product yields observed were: \(G(\text{H}_2)\) of 0.026; \(G(\text{O}_2)\) of 0.43; and \(G(\text{NO}_2^-)\) of 1.1. (G-values are defined as the number of species formed per 100 eV of energy absorbed.) The large nitrite ion yield indicates that the nitrate ion is being decomposed by direct action of the radiation in these solutions.

On the basis of the gamma radiolysis experiments (Linear Energy Transfer (LET) \(-0.2\) eV nm\(^{-1}\)), 4.7 mol dm\(^{-3}\) LiOH solutions (natural abundance lithium) were chosen for in-reactor radiolysis experiments, where the radiation energy is essentially all derived from the recoil ions resulting from the reaction of thermal neutrons with \(^{6}\text{Li}^*\) (LET \(-24\) eV nm\(^{-1}\) for the triton and \(-180\) eV nm\(^{-1}\) for the alpha particle):

\[
\text{n} + ^{6}\text{Li} \longrightarrow ^{3}\text{H} + ^{4}\text{He}
\]

\[\text{FIGURE 4 FORMATION OF HYDROGEN AND OXYGEN AS A FUNCTION OF DOSE.}\]
Under these radiolysis conditions, the LET is comparable to, and the dose rate of $10^3$ Gy s$^{-1}$ is about 5% of that expected in an ITER/NET fusion reactor. Figure 4 summarizes the formation of hydrogen and oxygen in these solutions. The radiolysis does not approach steady-state conditions, as both gases are formed linearly with dose, with G-values of 0.8 and 0.2, respectively. Furthermore, the gases are not formed in stoichiometric amounts and this is attributed to impurity effects in the small (2x10$^{-3}$ dm$^3$) quartz ampoules.

Although a dose of only 1.75x10$^4$ Gy is expected on a single pass through an ITER/NET fusion reactor, the production of gases at the rate shown in Figure 4 will require very large systems operating pressures (~2-3 MPa) in order to keep these gases in solution, to prevent possible explosions (15). However, computer simulations at CRNL indicate that, for the radiolysis expected in a fusion reactor, the addition of modest amounts of hydrogen gas (~5x10$^{-4}$ mol dm$^{-3}$) to the solution should suppress the excess hydrogen and oxygen production (15). This effect is shown in Figure 5. Experiments are planned at CRNL to confirm these predictions.

**INTERNATIONAL PROGRAMS**

BEATRIX has been a successful example of international co-operation on solid-blanket technology. Canada, via the CRNL program, is a full partner in BEATRIX-I - the US, EEC and Japan are other participants. A recent matrix document showed 19 individual experiments within the matrix. Canada was the first to carry out testing under BEATRIX, on lithium aluminate from CEA (Saclay, France); experiments have also been performed on material supplied by Japan and the US.

Canada is now participating in BEATRIX-II, advanced testing of candidate blanket materials in FFTF, Hanford, with Japan and the US as other partners. During 1985, the BEATRIX-II program, Phase I (16-18), progressed through design efforts into fabrication of its systems in order to meet a scheduled beginning of irradiation late in 1989. Funds were made available by Canada, Japan, and the US to pursue the design and fabrication program at Pacific Northwest Laboratory, Westinghouse Hanford Company and AECL. In addition, the Japanese Atomic Energy Research Institute (JAERI) fabricated ceramic electrolysis cells for incorporation into the sweep-gas analysis lines. It appears that the final tasks before insertion into the reactor will be completed in time. Both the IEA and Hanford site-design-review meetings have approved the final design of the system.

The purpose of BEATRIX-II is to conduct an in-situ tritium recovery experiment in the high-energy-neutron environment of FFTF. During Phase I, two in-situ capsules containing Li$_2$O will be irradiated, with higher tritium production levels than in previous experiments. One capsule possesses the capability for temperature-change experiments, while the other exhibits a temperature gradient for evaluating the temperature stability in an engineering blanket configuration. Nonvented capsules are included for measurement of irradiation damage, thermal diffusivity, release kinetics, and beryllium compatibility. The tritium produced in the Phase I test and accumulated in the tritium removal system getter beds will be shipped to the Tritium System Test Assembly at Los Alamos National Laboratory. The tritium will be processed into a form compatible with fusion fueling, thus demonstrating the complete fusion-fuel cycle. The scope of Phase II (Cycle 12) irradiations, scheduled for late 1990, has just been established. CRNL-fabricated lithium zirconate spheres will be included in one of the vented capsules.

In addition to the BEATRIX program, we are also participating in the JAERI Tritium Project. Laboratories from Canada, US, Japan and Europe have provided lithium-based ceramics for irradiation in the FNS and LOTUS high-energy (14 MeV) neutron facilities. On completion of irradiation, the ceramic will be measured for tritium content; all data will be pooled and reported, to improve blanket neutronics calculations.

**CONCLUSIONS**

(1) There has been substantial progress in the three main elements of the program,
- completion of the long-term, CRITIC-I in-reactor test that has provided significant reactor-relevant data on Li$_2$O, and preparation for CRITIC-II, with lithium zirconate spheres,
- good progress in lithium-based powder synthesis and the development of high-speed, sphere-fabrication technology, and
- selection of LiOH as the prime candidate for the aqueous lithium salt blanket concept following gamma-cell and in-reactor radiolysis testing.

(2) In international programs, lithium zirconate spheres fabricated at CRNL will be tested in Phase II of the BEATRIX-II irradiation in FFTF, Hanford. Additionally, the JAERI Tritium Project will result in improved blanket neutronics data.

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In the operation of magnetic confinement fusion reactors, first wall materials are exposed to large fluxes of plasma particles, predominantly hydrogenic species. This exposure leads to wall erosion and subsequent plasma contamination, both of which have strong negative implications for reactor operation. This paper considers the effects of various bombardment conditions on the erosion behaviour of carbon and carbon-based materials employed in present day tokamaks. Erosion measurements were performed in three temperature regimes, each of which involves a different erosion mechanism. At temperatures below about 500 K, physical sputtering dominates, and while the sputtering yields are generally considered to be low, the angle at which the ions are incident on the sample is an important factor. In the range 500 to 1000 K, chemical sputtering - hydrocarbon production - dominates for most forms of carbon (graphite, thin carbon films, and carbon/carbon composites). Carbon containing compounds, such as boron carbide, however, appear to have reduced chemical erosion properties. At temperatures above about 1300 K, radiation-enhanced sublimation becomes important. We have studied this effect as a function of flux density and ion energy.

1. INTRODUCTION

In the operation of a fusion reactor, hydrogenic fuel, in the form of a high-temperature plasma, must be contained for a sufficient period of time, such that the energy released from fusion reactions is greater than the energy required to heat and contain the plasma. Since its development in the 1950's, one of the most successful techniques for magnetically containing a hydrogen plasma is the toroidal 'tokamak' configuration. Such a device employs both toroidal and poloidal magnetic fields to stabilize and contain the plasma. A major advantage of the technique is the ability to heat the plasma by driving a very large current through it.

Even with very strong magnetic fields, however, the containment of the plasma is not complete, and there is leakage of energetic particles from the main body of the plasma outward, and thus the plasma edge inevitably comes in contact with the reactor wall structure. This contact results in the loss of plasma particles to the wall, and contamination of the plasma by eroded materials. Also, the recycling of the hydrogenic fuel from the walls during a discharge may have a strong impact on the plasma density, and therefore, its stability. A schematic showing the primary mechanisms involved in tokamak plasma-surface interactions is presented in Figure 1.

Contamination of a fusion plasma by impurities has two pronounced negative effects on energy generation. Firstly, the impurities dilute the fuel since the impurity atoms generally contain many more electrons than do hydrogen atoms. Secondly, the impurities may radiate energy away from the plasma via line radiation if they are not fully ionized, and they also result in increased bremsstrahlung radiation due to their higher atomic number. The heavier the impurity, the greater the effect on all accounts. The maximum permitted concentrations of impurities in tokamak plasmas are on the order of $10^{-2}$ for impurities with $Z < 8$ and $10^{-3}$ for high-Z elements (1).

In the past few years, carbon has become a leading contender for all plasma-facing surfaces in tokamak reactors because of its good mechanical properties, high melting point, resistance to thermal shock, and low atomic number. Graphite is frequently used for limiters, and is a common material for protective tiles. The complete coating of all plasma-facing surfaces with a layer of carbon, by plasma-deposition, is also being used (2). Two other low-Z elements, Be and B, have been much more limited application, but remain serious alternatives.

The present paper reviews some of the more recent results in the fusion materials erosion field, with emphasis on our latest findings.

2. HYDROGEN EROSION OF CARBON

A difficulty with using light elements as plasma-interactive surfaces is that while a higher concentration is acceptable in the plasma, the erosion rates due to plasma bombardment are also higher. This is especially true for carbon in the 500-1000 K temperature range where chemical reactions between incident hydrogen ions and the near-surface carbon atoms lead to the formation of volatile hydrocarbons. Such processes, which involve both energy deposition and chemical reactions, are generally referred to as chemical sputtering. They are characterized by an erosion yield which has a noticeable temperature dependence, and the production of volatile molecules, in this case, hydrocarbons - primarily methane.

Occurring for all temperatures, and all materials, physical sputtering involves the physical transfer of momentum from the incident ions to the target atoms. If sufficient momentum is transferred, in an appropriate direction, the target atom can be ejected from the surface. The erosion yield due to physical sputtering by hydrogen is generally significantly smaller than the maximum chemical sputtering yield for carbon. Thus it is the dominant erosion mechanism only when the surface temperature is
reduced to the point where most of the chemical reactions are suppressed. For H\(^+\) energies above ~100 eV, the transition point is about 400–500 K; however, for lower ion energies, especially near the theoretical threshold for physical sputtering, chemical processes may still be important at temperatures as low as 300K (3).

At temperatures above ~1300 K, the erosion yield of carbon atoms due to energetic ion (not necessarily hydrogenic) impact is found to increase exponentially with temperature (4,5). Because of this exponential temperature dependence, and an absence of chemical products, the process has come to be known as ion-induced or radiation-enhanced sublimation, RES. At temperatures near 2000 K, the C/H\(^+\) yield reaches or exceeds the maximum chemical sputtering yield of ~0.1 C/H\(^+\) for H\(^+\) ion energies > 200 eV. It must be remembered, however, that most other materials, and certainly all light elements would have melted by this point.

The three erosion regimes outlined above are shown in Figure 2 for 1 keV H\(^+\), D\(^+\) and 3 keV He\(^+\) ions incident on pyrolytic graphite over the temperature range 350 to 2000 K (4). At temperatures above ~2200 K, thermal sublimation becomes appreciable, and quickly dominates the erosion, independent of any ion effects.

2.1 Physical Sputtering

Most surfaces in a fusion reactor will not receive high enough heat loads to take the surface out of the physical sputtering regime. Even here, however, the potential exists for a run-away situation where the sputtering yield (i.e., the number of sputtered atoms per incident ion) could exceed unity. As indicated in Figure 2, at normal incidence the physical sputtering yield for H + C is about 10\(^{-2}\), for 1 keV H\(^+\) bombardment. As the angle of incidence is decreased, the sputtering yield increases, by up to an order of magnitude at grazing incidence. For the H + C case, this brings the yield up to ~10\(^{-1}\) C/H\(^+\), which may
lead to serious problems, but by itself may not be catastrophic. The sputtered carbon atoms, however, may be ionized in the edge plasma, and subsequently impact on the carbon wall, thus triggering $C + C$ self-sputtering. While self-sputtering yields at normal incidence are $\sim 10^{-2}$, at large angles, yields greater than unity could result (6). A similar result has been found for Be as well (7). Such a situation could well be critical for a long-pulse fusion reactor.

We have performed experiments at UTIAS investigating the effect of ion incidence angle for the $H^+ + C$ system, with serious consideration given to the nature of the surface (8). In Figure 3 we show the $H^+ + C$ physical sputtering angular dependence for a highly-oriented graphite surface bombarded by 50 to 300 eV $H^+$ ions. These results agree reasonably well with previous measurements using different techniques (9), and also with numerical simulations (10, 11). Figure 4 shows similar experimental results for materials with different surface roughness characteristics. It is evident that surface roughness plays a key role in the physical sputtering process. A further point of interest in these results is the curve of Figure 4(b), where the surface was originally highly-oriented pyrolytic graphite, but was modified by high fluence ion bombardment. It was found that such surface modification significantly alters the physical sputtering yield at large angles. The next step will be to demonstrate that a similar surface roughness dependence occurs for the $C^+ + C$ self-sputtering case.

2.2 Chemical Sputtering

In the temperature range 500 to 1000 K, chemical reactions can occur when carbon is bombarded by hydrogenic species. In the last 10 to 15 years, carbon, primarily in the form of graphite, has been thoroughly tested to determine the rate at which hydrocarbons are produced under various bombardment conditions. An example of this is given in Figure 3, which shows the erosion of pyrolytic graphite due to bombardment by 300 eV $H^+$ ions; the temperature dependence of the hydrocarbon production is clearly shown. In a reactor situation, however, it is likely that different forms of carbon other than pyrolytic graphite will be employed. For example, with the use of carbon/carbon composites, it is possible to produce components with improved structural and thermal characteristics. Such materials are likely to be used in the next generation tokamaks (12, 13). Recently, we have undertaken several studies in order to compare the erosion characteristics of different forms of carbon. Plasma-deposited thin films and carbon/carbon composites represent very different forms of carbon, and one might well expect major differences in the erosion behaviour. As indicated in Figure 6 (14) and Figure 7 (15), however, only very minor, if any, differences were observed. This phenomenon is most likely due to the fact that during energetic hydrogen ion bombardment, carbon surfaces become amorphous and hydrogen saturated, in effect making them very similar to the plasma-deposited thin film. Thus the erosion yields tend to be independent of whether the sample was initially graphite or other forms of carbon.

Much of our work on chemical erosion has involved...
the study of synergistic effects occurring during simultaneous bombardment of graphite by \( \text{H}^+ \) ions and thermal \( \text{H}^0 \) atoms. On their own, \( \text{H}^0 \) atoms have a maximum erosion yield of \( \approx 10^{-3} \text{C/H}^0 \), involving a number of hydrocarbon species (16). During simultaneous ion bombardment, though, this yield may be increased to the order of \( 10^{-1} \text{C/H}^0 \), provided the \( \text{H}^+/\text{H}^0 \) flux ratio is \( > 0.1 \) (16, 17); see Figure 8. This synergism is quite relevant in a reactor situation, where electron dissociation of \( \text{H}_2 \) molecules desorbed from the walls will produce Franck-Condon \( \text{H}^0 \) atoms of \( \approx 3 \text{eV} \) which will impact on the walls. The magnitude of the flux will be roughly equivalent to the incident ion flux, and thus a near doubling of the hydrocarbon production due to ions alone may be expected in some circumstances.

Another synergistic effect, which we have recently undertaken to investigate, is the co-bombardment of graphite by \( \text{H}^+ \) ions and electrons. A very significant enhancement of the erosion yield (by up to \( 25\times \)) was reported by Guseva et al in \( \approx 1983-1984 \) for the synergism (18, 19). At about the same time, experiments at UTIAS studying the \( \text{H}^+ + \text{S} \) interaction indicated that there was no significant synergism for this case (20). Similarly, our recent measurements of \( \text{H}^+ + \text{S} \) chemical sputtering failed to find any enhancement of the \( \text{H}^+ \) erosion yield due to the addition of the electrons. Confirmation of our results by Vietzke et al (21) leads us to have serious doubts about the earlier experiments (18, 19).

2.3 Radiation-Enhanced Sublimation

For the next generation of fusion devices, it has been proposed that some high heat flux surfaces be protected by carbon tiles which are mounted in a manner which allows cooling by radiation only (22). It is projected that the temperature of some of these

![Figure 5: Hydrocarbon yields from pyrolytic graphite as a function of sample temperature, for 900 eV \( \text{H}^+ \) ion bombardment.](image)

![Figure 6: Hydrocarbon yields from Aerolorb carbon/carbon composite as a function of sample temperature, for 900 eV \( \text{H}^+ \) ion bombardment (from Ref. 14).](image)

![Figure 7: Methane production as a function of sample temperature for \( \text{H}^+ \) bombardment of amorphous hydrogenated carbon films produced by plasma-deposition (from Ref. 15).](image)
tiles will reach the 1800 to 2000 K range, leading to erosion by radiation-enhanced sublimation.

The suggested mechanism for the RES process involves the formation of carbon interstitials due to nuclear energy deposition in the near-surface layer by the impacting energetic particles, and subsequent diffusion of the carbon atoms to the surface (4, 5, 23-25). Some calculations based on this mechanism have indicated that a strong dependence on flux density may occur (26), and that yields measured in beam experiments, with fluxes limited to \( \leq 10^{16} \) ions/cm\(^2\)s, may be significantly higher than those found in a reactor.

To address this flux dependence question, we have performed laboratory experiments where the flux density was varied over three orders of magnitude (27). Unfortunately, the results (Figure 9) display only a very minor dependence on flux density, and the yield is reduced only by about a factor of two over the three orders of magnitude flux range.

To extend our results to higher fluxes - closer to reactor conditions - we are collaborating with Sandia National Laboratories, Albuquerque, using their high heat flux facility. Our first results, with fluxes of \( \approx 10^{17} \) ions/cm\(^2\)s, strongly suggest that the linear flux dependence of RES continues at least to this flux density (28).

### 3. OTHER LOW-Z MATERIALS

While carbon has many advantages as a first wall material, it is not the only low-Z material considered for plasma-facing components. Both boron (Z = 5) and beryllium (Z = 4) have seen limited testing in a reactor environment. Boron has been used, in combination with carbon, as a thin film coating in TEXTOR, providing a plasma impurity reduction by about a factor of two, compared to carbon coatings (29). This reduction is attributed to a suppression of H + C chemical reactions.

In an attempt to produce improved first-wall materials, we at UTIAS have started an R & D program on boron-containing compounds. First results have been obtained on the chemical sputtering of sintered boron carbide, B\(_4\)C. While chemical removal of the carbon atoms occurs, as indicated by the production of hydrocarbons, see Figure 10, the normal temperature dependence associated with chemical sputtering is not evident. This suggests that some mechanism, besides the chemical reactions themselves, is limiting the rate at which the reactions can proceed. Possible mechanisms are physical sputtering of the boron component or ion-induced mixing of the near-surface.

The testing of bulk Be limiters has been carried out in the ISX-B tokamak (30), and full-scale testing of Be in JET is planned for later this year - Be will be evaporated on all internal surfaces. It is hoped that several advantages will be gained by this, the most important being the gettering of oxygen. Provided that the Be sputtering yield is not substantially higher than that for carbon, and oxygen impurities are trapped by the beryllium, the plasma purity should be improved, as long as large-scale vaporization and melting do not occur.

### 4. CONCLUSIONS

At the present time, it appears that the light
elements Be, B and C are the best candidates for first wall limiter and divertor surfaces in a tokamak-like fusion reactor. In the case of carbon, self-sputtering at large angles of incidence and radiation-enhanced sublimation may, however, impose an upper limit on the operating wall temperature. A major consideration which has yet to be studied for these or any other materials is the effect of high-flux, high-energy neutrons. Experience with carbon in fission reactors is that it will not stand up very well structurally to the neutron bombardment. Boron, on the other hand, can be a strong neutron absorber. Attempts are being made at improving carbon's neutron irradiation resistance, by specially tailored carbon/carbon composites, to withstand the neutron fluences expected in the next generation fusion devices, e.g., International Thermonuclear Experimental Reactor, ITER. Further efforts are needed, however, to develop advanced materials for long-life fusion power reactors.

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DIFFUSION AND RETENTION OF TRITIUM IN GRAPHITE

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ABSTRACT

The retention of a mixture of atomic protium and tritium striking a pyrolytic graphite surface with thermal energies was measured by thermal desorption, and found to be similar to literature values for the retention of pure atomic protium, and pure atomic deuterium. Isotopic effects could not be detected. For incident fluences of between 10^{18} and 10^{20} atoms/cm^2, retention is approximately 9.3 \times 10^8 F^{-33} atoms/cm^2, where F is the incident fluence. In a separate experiment, tritium imaging is used to monitor tritium implanted in a pseudo-monocrystal graphite sample, in an attempt to measure its diffusion coefficient. The implanted tritium proved to be very immobile, and was only partially desorbed after prolonged heating of the sample to temperatures of almost 2000 K.

I. INTRODUCTION

Graphite has many desirable properties as a material for components facing the plasma in magnetically confined fusion reactors. It can withstand high temperatures, and does not melt; it has or relatively high thermal and electrical conductivity, and an adequate mechanical strength. The low atomic number of carbon is desirable, since high-Z impurities are a source of radiation and hence unwanted cooling of the plasma. Nevertheless, graphite unfortunately does interact in complex and possibly undesirable ways with the hydrogen isotopes that fuel the reactor.

One important process is the diffusion of tritium into graphite, and its retention there. Under certain circumstances, tritium-to-carbon ratios as high as 0.6 can be obtained (1), so that a relatively small amount of graphite might retain substantial amounts of a reactor's fuel, increasing the necessary inventory with obvious consequences for both economy and safety.

Four processes have been identified by which tritium may be retained in or on graphite in a magnetically confined fusion reactor (1); saturation by energetic implantation, co-deposition with carbon, adsorption on graphite surfaces, and trans-granular diffusion with trapping. The first two processes shall not concern us here. Adsorption on surfaces is significant because of the great porosity of graphite, which makes its effective adsorptive area much greater than its macroscopic area. Trans-granular diffusion represents passage of tritium into the bulk of the graphite, where trapping may make it difficult to dislodge. Here we present results from experiments examining these two processes.

2. RETENTION OF A THERMAL FLUX OF ATOMIC TRITIUM AND PROTIUM IN PYROLYTIC GRAPHITE

2.1 Experiment

Although graphite is substantially inert to molecular hydrogen, it reacts readily with the atomic form, H^0, adsorbing it on the surface, forming hydrocarbon precursors, and eventually volatile hydrocarbons, or possibly hydrocarbon radicals (2-5). It is of interest to determine the total amount of hydrogen retained on a graphite sample as a function of the incoming fluence of atomic hydrogen, and to know if there are significant isotopic effects.

To this end, a sample of pyrolytic graphite was exposed to a flux of atomic hydrogen, composed of a mixture (about 2:1) of protium and tritium, for varying lengths of time. The sample was then heated to desorb all of the hydrogen. From the amount of hydrogen desorbed and its isotopic ratios, the retention of each isotope as a function of incident fluence could be calculated.

The experiments were carried out in the UTIAS Low-Level Tritium Facility, a simplified schematic of which appears in Figure 1.

A backfill of about 4 \times 10^{-4} torr of hydrogen was produced by heating SAES getters G_0 and G_2. Pressure and isotopic composition were monitored by the ionization gauge and the quadrupole mass spectrometer, respectively. Cryocondensation pump P_2, which had been specially modified so as not to pump hydrogen, was used to control undesirable condensible gases. Valve V_P_1, leading to pump P_1, which did pump hydrogen, remained closed during the backfill. The backfill was dissociated by means of a 12 mm long, 1 mm wide, hot tungsten filament placed approximately 6 mm in front of the sample, which consisted of a strip of Union Carbide HPG 99 pyrolytic graphite 33 mm long \times 8 mm wide \times 0.2 mm thick. The tungsten filament temperature was estimated to be 1900K - hot enough to produce a substantial flux of atomic hydrogen, yet not hot enough to produce...
Therefore, in order to determine the isotopic and would interfere with the measurements. Knowing the volume of the chamber, the rise in pressure observed with the ionization gauge allows the amount of hydrogen desorbed to be calculated.

Once exposure was complete, the filament was turned off, valves $V_{E}$ and $V_{Q}$ were closed to isolate the getters, and the test chamber was evacuated to $\sim$10$^{-9}$ torr by means of cryocondensation pump $P_{1}$. After isolating the test chamber by closing $V_{P1}$ and $V_{P2}$, the sample was heated to above 1600K for a few seconds. This was sufficient to drive off all the hydrogen adsorbed during the previous exposure. Knowing the volume of the chamber, the rise in pressure observed with the ionization gauge allows the amount of hydrogen desorbed to be calculated.

It would have been desirable to use the quadrupole to monitor the isotopic composition of the desorbed gas, but the getter $G_{Q}$, placed in the quadrupole chamber for an earlier experiment, can readily pump or desorb hydrogen at a rate not easily predicted, and would interfere with the measurements. Therefore, in order to determine the isotopic abundances in the desorbed gas, the test chamber was isolated by closing $V_{Q}$, and sampled through a 1 mm orifice by the quadrupole mass spectrometer, which was simultaneously differentially pumped by cryocondensation pump $P_{1}$. The high, more-or-less constant pumping speed of the cryopump dominates any effects of the getter $G_{Q}$. Because the pumping speed of the sampling orifice depends on the mass of the species being sampled, the composition of the gas in the test chamber can be observed to change with time in an easily predictable manner. It is necessary, therefore, to measure the relative abundances of the different isotopes at the moment sampling is begun, i.e., the moment $V_{Q}$ is opened.

### 2.2 Results and Discussion

During exposure of the sample to atomic hydrogen, there was an equilibrium distribution of molecular species, $H_{2}$, $HT$, and $T_{2}$, in the backfill gas. The amount of deuterium present was negligible.

The fluence to the sample was calculated according to the geometry of the sample-filament arrangement, and making use of the results of Hickmott (6), with the following modification. Hickmott calculated a rate of production of atomic protium, roughly proportional to the $H_{2}$ backfill. In our case, the $HT$ present will also contribute, modified by a factor of $1/2$, since only half of the incoming molecule is protium, and by a further factor of $(1/2)^{1/2}$, because, for a given pressure, the flux of molecules incident on a surface is inversely proportional to the square root of the molecular weight. It will also contribute to the atomic tritium flux to the same extent that it contributes to the atomic protium flux. Finally, the $T_{2}$ will contribute to the atomic tritium flux to the same extent as the $H_{2}$ contributed to the $H^{0}$ flux, except for a factor of $(1/3)^{1/2}$, again arising from the greater mass of $T_{2}$ with respect to $H_{2}$. Thus, knowing the pressure of the three different forms of hydrogen present, the flux (and by integrating over time, the fluence) of atomic protium and tritium on the sample can be calculated. It is important to note that the proportion of protium in the atomic fluence is greater than the proportion of protium in the backfill gas.

The hydrogen sampled after the desorption process contains, like the backfill, equilibrium proportions of the three molecular species, indicating no isotopic separation during the release process. The atomic percent of protium is approximately equal to that in the bombarding fluence, which, in most experiments, represents approximately a 6% enrichment with respect to the original backfill (see Table 1). The relation that appears from its slower molecular speed in the gaseous state as a result of its greater atomic weight, the behaviour of tritium was very similar to that of protium with regards to retention in the near surface of our graphite sample.

The total hydrogen ($H$ and $T$) retained in the sample as a function of bombarding fluence is plotted in Figure 2. Measured values are in relatively good agreement with previous results for protium and deuterium at similar sample temperatures (3, 7). The earlier results obtained by Hicks et al (2), using a tritium tracer technique ($D/T = 1000$) and post-exposure combustion of the sample show retention levels about an order of magnitude higher than the present results with tritium. This anomaly still remains to be resolved. Figure 2 also shows the retention of energetic deuterium in graphite. It is generally accepted that the mechanism of retention of energetic hydrogen is substantially different from that of thermal atoms, and approaches 100%, up to a saturation value in the graphite. This is because energetic hydrogen implants itself within the graphite crystals rather than on the internal porosity of the material.
The present results show a one-third power dependence of retention on irradiating fluence, with no significant departure from this relationship at either extreme. A least-squares-fit of the points shown in Figure 2 gives retention \( R \) (atoms/cm\(^2\)) as

\[
R = 9.33 \times 10^8 F^{0.33} \text{ atoms/cm}^2
\]

where \( F \) is the total incident fluence in atoms/cm\(^2\).

It can be shown (8) that for a thick slab of material, in the diffusion limited regime, where an incident flux maintains a constant density on the sample surface starting at time \( t = 0 \), and diffuses into the material, the inventory in the material will be proportional to the square root of time, or to the one half power of total fluence. Such calculations are based on Fick's Law of Diffusion, which is clearly applicable to diffusion within a bulk material, but which may not apply to our case, since low energy hydrogen is thought to diffuse through graphite along the surfaces of its internal porosity (3). The possibility of trapping sites of unknown energy may also complicate analysis of the situation.

3. TRITIUM IMAGING AND DIFFUSION IN GRAPHITE CRYSTAL GRAINS

3.1 Tritium Imaging Technique

In an effort to understand trans-granular diffusion, an attempt was made to follow the movement of tritium in a pseudo-monocrystal graphite using a technique known as tritium imaging, first developed by Malinowski (9-12) as an alternative to autoradiography.

Using a flat, smooth surface as the cathode in an electrostatic three-element immersion lens, electrons emitted from that surface can be focused on an image intensifier such as a MicroChannel Plate Detector (MCP). If the source of the electrons is near surface tritium, the image will show the distribution of that tritium. Because the spectrum of energies associated with the tritium beta-particles is large, (0-18 keV), attempts to use them for this type of imaging would suffer from severe chromatic aberration or would require enormous accelerating voltages within the lens. However, the flux of secondary electrons released from the surface by the betas can be conveniently used for imaging since they emerge with a narrow energy spread (0-50 eV) and are thus much more amenable to focusing. Furthermore, the number of secondaries approximately equals the number of primaries, so there is little loss of image intensity as a result of using secondaries. Even so, the flux of imaging electrons is very small, hence the need for an image Intensifier. In fact, imaging cannot be carried out during annealing of the sample at temperatures above about 900-1000K, since thermionic electrons will bury the tritium signal. It must also be noted that the short range of the tritium beta in graphite limits the depth at which tritium can be detected by this means to about 300 nanometres.

3.2 Experiment

The imaging system used in these experiments was designed and built at UTIAS, and is shown schematically in Figure 3. An oil-free vacuum of less than \( 10^{-6} \) torr is required for the experiment, and this was provided by a turbomolecular pump backed by a molecular sieve chilled to liquid nitrogen.
The best tritium images of these samples were obtained when the sample-grid distance was smallest (about 0.35 mm). Magnification was approximately x16. The theoretical limit of resolution of this technique is the range of a tritium beta in graphite, or about 0.5 μm. The actual resolution, based on the size of the smallest regularly discernible feature in the image, was estimated to be 10-20 μm.

It has been suggested that trans-granular diffusion in graphite occurs almost entirely in the a-b (basal plane) direction (14). This condition must hold if lateral diffusion of the surface tritium is to be observed before it diffuses either out of or too deeply into the sample to be imaged.

Discussion

Figure 4a shows the tritium image of sample I as it was received after implantation. A number of features are visible. The dark irregularities around the edge might be attributed to burrs on the mask which defined the irradiation spot. However, a couple of dark features which are plainly completely within the tritiated region. Because the scanning electron micrographs of the sample surface before implantation did not show features of this size or shape, it may be assumed that they are not topographic. They could represent regions where particles were clinging to the sample during irradiation. Finally, there is a fine, bright line which runs across the diameter of the tritiated region. Although it is possible that this may be a result of a "hot spot" in the irradiating beam, this feature may correspond to the step in the cleaved surface mentioned above.

The sample was then annealed for 1 hr periods at increasing temperatures, as shown in Table 2. After annealing to 975K the bright line faded. After 1375K the line became dark (as in Figure 4b). Except for possibly a slight darkening of the image, no further changes in the image were observed after annealing up to 1775K.

The second sample has fewer features than the first (see Fig. 4c). There was a large, irregular dark spot near the centre. The boundaries of the image were smooth, but appeared to be defined by two slightly non-concentric circles. It is believed that this sample was irradiated in two exposures, and that the sample holder may have shifted between exposures. Despite the fact that the tritium had been implanted somewhat more deeply, there was no significant change in image brightness.

After annealing at 1775K, the second sample immediately developed cracks emanating from the dark central spot. This seems to indicate that this spot is not just an unirradiated area, but a physically different region of the sample, possibly a hole. The cracks became more noticeable as the sample was annealed to 1875K for an hour, and then 1975K for 1/2 hour (see Figure 4d). After a further twenty minute anneal at 1975K, a section representing approximately 40% of the tritiated spot had flaked off (see Figure 4e). However, throughout this process, no significant lateral motion of tritium was detected.

By using very low f-stops, photographs of short exposure were taken, which show dots representing the individual secondary electrons emitted in that time.
TABLE 2: HISTORY OF ANNEALING OF SAMPLES USED IN TRITIUM IMAGING

<table>
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density, as the situation may be complicated by depth-of-distribution effects. The results of this technique are shown in Figure 5. The error bars are equal to the expected standard deviation of a Poisson probability distribution, which should govern these events. The scatter is slightly greater than the error bars would suggest, particularly for the unannealed sample. Among other things, this may be a result of the difficulty of counting large numbers of faint dots spaced very close together.

Without being too quantitative, several factors are evident. A significant fraction (tens of percent) of the tritium is lost from the near-surface during the 1 hr 1775K anneal. More is lost during the subsequent 1 hr 1875K anneal. However, a significant amount (again on the order of tens of percent) remains, and furthermore, this remaining fraction is not significantly depleted during annealing to 1975K.

Because lateral diffusion of tritium was not visible even on a scale of 20 μm after annealing at 1975K for about 50 minutes, we can establish an approximate upper limit on the diffusion coefficient in the basal plane at that temperature, i.e.,

$$D < \frac{2 \times 10^{-3} \text{ cm}^2}{3000 \text{ s} \times 10^{-9} \text{ cm}^2/\text{s}}$$

at 1975K. This limit is compatible with the results of Causey et al (15), Saeki (14) and Rohrig et al (16), but not with those of Aramal et al (17), and Malka et al (18) (see Figure 6). It should be noted, however, that these authors have studied diffusion in small-grained graphite. Saeki, for example, reports grain sizes of 2-4 nm. In addition, the situation is not as simple as mere numerical agreement might suggest. When Saeki measured diffusion in a highly oriented pyrolytic graphite, for example, he was actually looking at diffusion in the c-direction. The relatively low diffusion coefficient was then attributed to the great distance that tritium had to migrate in the basal plane in order to find a defect or grain boundary that would allow movement in the c-direction.

If near surface tritium is lost by diffusion into or out of the material, it is possible to obtain a rough estimate of the diffusion coefficient in the
order of 250 nm, and at the temperatures at which c-direction. The distance it diffuses will be on the 
about $(2.5 \times 10^{-5}) / 10^3 \text{ cm}^2 / \text{s}$, or about $10^{-12} \text{ cm}^2 / \text{s}$.

On the other hand, processes other than simple diffusion can be postulated to explain the observations. Sawicki et al (19), for example, have used nuclear reaction depth profiling to observe depletion of implanted tritium in fine grain isotropic graphite after annealing. They did not see the change in depth profile that might be expected to accompany a diffusive process, and have suggested that it was lost through recombination of the tritium and its subsequent rapid movement out of the material in molecular form.

The most remarkable thing about the observations described here is the very high temperatures at which the sample can be maintained without dislodging all of the tritium. Several authors report that for low fluences of implanted hydrogen, 90% has been desorbed by 1500K while for high fluences desorption is 90% complete by about 1200K (see summary of results in Ref. 19). The work of Beutler et al (20) suggests that the inherent hydrogen in a pyrolytic graphite can be reduced by three orders of magnitude by means of a 2000K bakeout of only a few minutes duration. These results, it should be noted, were all obtained by measuring hydrogen release in small-grained graphites. It is possible that what was actually being measured was diffusion out of the small grains, which can occur relatively more rapidly, followed by diffusion along grain boundaries.

The lack of change in surface tritium density during the 1975K anneals in the present experiments suggests that the remaining tritium is somehow less mobile than the rest. Numerous authors (1, 7, 17, 21) have suggested that graphite contains trapping sites of a number of different energies. The high temperature to which we can expose our samples without detrapping implies fairly deep traps, although we do not have sufficient data to estimate their energies quantitatively.

Causey (1) suggests that trapping sites are found on crystalline edges. If this were so, the crystal grain size of the samples used here (10-80 μm, 35 μm ave.) is large enough that tritium imaging ought to be able to resolve grain boundaries, which is not the case. It may be possible that the process of implantation created traps uniformly through the implanted area. The density of such traps would have to be very high however - on the order of 1 atomic percent - to accommodate all the tritium implanted. Causey also states (1) that where high trap densities are found "Tritium release...is similar to...classical diffusion...[and] the activation energy for the apparent diffusion coefficient is that of the trap energy". In other words, trapping phenomena and diffusion become indistinguishable.

It is interesting to note that Sawicki et al (19) did not observe such trapping, despite using an identical implantation technique at similar energies. On the other hand, Saeki did observe residual tritium in his samples, even after annealing to 1600K.

Malinowski et al (12) have used tritium imaging to examine a sample substantially similar to our own, implanted with tritium at a somewhat lower energy (25
Their experiment differed from ours in that their surface of their sample. In addition, their surface was in as—received condition and may have had more inherent imperfections than a carefully cleaved surface. The as-received surface of our sample was shown by SEM to be much rougher than the freshly cleaved surfaces which were used for implantation.

4. SUMMARY

Retention of atomic tritium with thermal energies in pyrolytic graphite surfaces has been shown to be similar to retention of protium, with no detectable isotopic effects.

Diffusion within the crystal structure of graphite has been shown to be less than $10^{-9}$ cm$^2$/s in the a-b plane for temperatures below 197.5K. It was demonstrated that tritium implanted in large crystal grains may be retained even after heating to nearly 2000K for prolonged periods. This is in contrast to the much lower temperature thermal release of hydrogen (including tritium) observed by others in smaller-grained graphites.

5. ACKNOWLEDGEMENTS

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APPLICATION OF RISK ASSESSMENT TECHNIQUES TO THE JOINT EUROPEAN TORUS (JET) ACTIVE GAS HANDLING SYSTEM

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ABSTRACT

Risk assessment techniques are used for the safety analysis and licensing of the tritium handling system of the Joint European Torus (JET), one of the largest tokamak fusion devices in the world. The methods used are described with representative examples and results. Finally, the usefulness of applying risk assessment techniques in this case is discussed.

INTRODUCTION

The Joint European Torus (JET) is the largest project in the coordinated fusion programme of the European Atomic Energy Community which is aimed at proving the feasibility of using nuclear fusion as a source of energy(1,2).

The essential objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermonuclear reactor. These studies are aimed at defining the parameters, the size and the working conditions of a tokamak reactor. The realization of this objective involves four main areas of work:

1. The study of scaling of plasma behaviour as parameters approach the reactor range;
2. The study of plasma-wall interactions in these conditions;
3. The study of plasma heating;
4. The study of alpha particle production, confinement and consequent plasma heating.

Two of the key technological issues in the subsequent development of a fusion reactor are found for the first time in JET: these are the use of tritium and the application of remote maintenance and repair techniques.

JET is located on a site near Oxford, UK, adjacent to the Culham Laboratory of the United Kingdom Atomic Energy Authority (UKAEA) which acts as the host organization. First plasma was achieved in June 1983.

Operating parameters are listed in Table 1. JET has obtained a fusion product \( N_{i}.T_{g}.T_{j} \) (ion density, energy confinement time, ion temperature) of \( 2.5 \times 10^{19} \text{ m}^{-3}.\text{s. keV} \). In a fusion reactor, the values must be such that their product exceeds \( 5 \times 10^{21} \text{ m}^{-3}.\text{s. keV} \).

So far, experiments have been carried out using hydrogen or deuterium plasmas. In the final stage of the programme, it is planned to operate with deuterium-tritium plasmas so that abundant fusion reactions occur. The alpha particles liberated from these reactions should then produce significant heating of the plasma. During this plasma operation, the machine structure will become radioactive to such an extent that all repairs and maintenance will have to be carried out using remote handling systems.

To achieve the aims, a planned programme of enhancements of the JET machine and its ancillaries is being undertaken, leading to the capability of plasma operation with tritium from mid 1991 onwards(3). The major addition is the Active Gas Handling System (AGHS) whose main function is the extraction and separation of hydrogen isotopes from the torus exhaust so that tritium and deuterium can be recycled (Figure 1). The AGHS is planned to be in operation in 1990 so that ample experience can be accumulated prior to operation of JET on a closed cycle with tritium.

ACTIVE GAS HANDLING SYSTEM(3,4)

The function of the JET AGHS, currently being designed and procured, is to gather mixtures of hydrogen isotopes (H, D, T) and impurities (tritiated water, hydrocarbons, etc.), to purify these mixtures and to resupply isotopically pure \( \text{D}_2 \) and \( \text{T}_2 \) for operation of the torus and its subsystems (neutral beam injectors, multi-pellet injectors). The design is based on a maximum total tritium inventory of 90 g \( \text{T}_2 \) with a maximum daily throughput of about 30 g \( \text{T}_2 \).

The plant comprises several major subsystems (Figure 2).

Cryogenic

For pumping hydrogen isotopes and impurities by cryocondensation, and, helium by cryosorption, at liquid helium temperatures (4 K). Hydrogen isotopes are transferred to Intermediate Storage, and helium and impurities to Impurity Processing.

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Impurity Processing

For extracting hydrogen isotopes from impurities. Impurities are recombined over a catalyst to tritiated water and non-active products and the tritiated water is reduced over hot uranium to hydrogen gas (H/D/T) and uranium oxide.

Intermediate Storage

For temporary storage of purified hydrogen isotope mixtures using uranium beds. Gases (H/D/T) are received, stored and transferred to isotopic separation.

Cryogenic Distillation

To separate the three hydrogen isotopes (H/D/T) using three cascaded columns at liquid hydrogen temperatures.

Gas Chromatography

To separate D₂ and T₂ using columns packed with palladium coated alumina spheres.

Product Storage

To receive and store isotopically pure D₂ and T₂ on uranium beds and deliver these to other JET systems via Gas Introduction.

Gas Introduction

Used to monitor and direct the feeds to other JET systems.

Mechanical Forevacuum

For evacuation of the tokamak and other systems using dry scroll, tritium-compatible vacuum pumps.

Exhaust Detritiation

To remove residual tritium from the torus and AGHS exhausts in a once-through manner.

The AGHS will be installed in a separate dedicated building connected to the main JET building by a bridge containing the necessary process and service pipework and cabling. Components routinely containing tritium at near or above atmospheric pressure have a secondary containment, generally of stainless steel, which is monitored for tritium. The Cryogenic Forevacuum system, Cryogenic Distillation and Gas Chromatographic Isotope Separation systems have secondary containments in the form of 1.2 m diameter stainless steel vessels. Where possible, these vessels are evacuated allowing very prompt detection of leaks. Components dissipating power, instruments, etc. are generally housed in compartments filled with helium or nitrogen. Large items like Normetex pumps, etc. are grouped together and the secondary containment is formed by an enclosure which is monitored for tritium and can be connected to Exhaust Detritiation, if necessary. Over-pressure protection and over-temperature protection are generally achieved by local temperature and pressure switches tripping at a level slightly higher than the setpoints of the automatic controller. Additionally, rupture discs are provided where required. Some of these release into a series of permanently evacuated buffer tanks.

Included in the design of the AGHS are six 10 m³ tanks that provide the opportunity to delay exhausts to the environment to allow tritium monitoring and provide the opportunity to recover from overpressure relief via bursting discs or from faulted process boundary components. Each of these tanks has sufficient volume to hold the entire inventory of the AGHS at near atmospheric pressure. They also provide the opportunity for the detritiation of routine releases that become abnormally contaminated with tritium.
All plant subsystems are designed to fail to a safe state in case of failure of services, i.e., cryogenic supplies, compressed air supply or electrical power. In the safe state, subsystems are isolated from each other with process gas at subatmospheric pressures and from external sources of heat and pressure. The intended use of uranium hydride storage is attractive in this regard - when all services are interrupted, the plant subsystems will approach room temperature, in which case all hydrogen isotope pressures will settle at the equilibrium pressure of uranium hydride (\( \sim 10^{-3} \) Pa).

Operation of subsystems and coordination of the overall plant operation will be automatically controlled by a micro-processor based control system. A dedicated control room will be installed in the non-active area of the gas handling building. It is expected that during certain phases of the programme 24 hours manning will be required. A link will be maintained to the main JET control room to make available alarm and main status signals as required.

In addition, extensive analytical capabilities are being provided.

SAFETY APPROVALS

The use of radioactive materials in the UK is constrained by government legislation which controls the licensing of plants, the exposure of employees, the disposal of waste and transport. The main statutes which JET is subject to are listed below:

Radioactive Substances Act 1960

The main impact of the Act on JET is that all routine discharges of tritium or activated substances to the environment and for disposal as waste must be approved in advance by Her Majesty's Inspectorate of Pollution (HMIP). JET is required to make a submission to HMIP showing that "Best Practicable Means" have been used in minimizing the environmental impact and showing that in all cases radiation doses to the most exposed members of the public are within targets.

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Ionizing Radiation Regulations 1985 (IRRs)

The legislation is concerned with limiting radiation doses to employees and the compliance is monitored by the Health and Safety Executive. JET occupational dose targets are considerably below the legislative limit of 50 mSv for annual committed dose to exposed workers. The legislation requires a contingency plan to be submitted to the HSE if quantities of radioactive material exceed certain limits. The quantity of tritium to be used at JET (90 g) is less than this limit (540 g) but a contingency plan will be produced nevertheless.

The accounting requirements of the IRRs will be met by the arrangements, using periodic pressure/volume/temperature measurements which are needed for process monitoring.

Transport Regulations

UK legislation is based on the International Atomic Energy Agency Code of Practice for safe transport of radioactive materials. It is not JET's intention to develop special transport containers for either tritium or waste as the timescale for approval, particularly in the case of licensed (Type 'R') containers is of the order of two years: therefore, standard containers (such as LP50) will be used for tritium transport.

Nuclear Installations Regulations

Under the 1965 Nuclear Installations Act, nuclear sites in the UK are required to be licensed by the Nuclear Installations Inspectorate (NIJ). At present, however, UKAEA sites are exempt from licensing and a semi-independent internal unit of the UKAEA, the Safety and Reliability Directorate (SRD) is responsible for ensuring that the same standards are applied to UKAEA sites as if they were licensed by the NIJ. Under the JET Host Agreement, the UKAEA ensures that JET applies similar standards and JET is required to satisfy SRD that the design of the plant, its construction and commissioning, and its operating procedures are acceptably safe. When satisfied, SRD will endorse the issue of an Authority to Operate (ATO) for the plant.

DESIGN TARGETS FOR ACCIDENTAL RELEASES

SRD will judge the acceptability of the AGHS with reference to UK NIJ guidelines for nuclear chemical plant. Internal JET safety design guidelines have been generated to ensure that designs can meet these guidelines. For example, the containment philosophy employed in the AGHS is that all main tritium process lines are doubly contained apart from certain exceptions, e.g., those which are of thick-walled, all-welded construction and which operate at room temperature and sub-atmospheric pressure such that permeation losses and the probability of a significant release are low.

SRD has also set a design target for each identifiable accident sequence - the product of frequency and magnitude of release (of tritium oxide) to the environment should be less than 0.37 (μCi/year). This target is more restrictive than the guidelines laid down for nuclear chemical plant by the UK NIJ. To demonstrate that this design target can be met, a probabilistic safety analysis is carried out as part of the Design Safety Review for each of the AGHS subsystems.

AGHS DESIGN SAFETY REVIEW

The AGHS is being reviewed in three stages - a Preliminary Safety Analysis Report (PSAR) based on the overall conceptual and preliminary design; a more in-depth Design Safety Review for each individual AGHS subsystem; and a Final Safety Analysis Report and other operational safety documentation based on the as-built AGHS. SRD review continues throughout the commissioning and operation period.

Routine releases, occupational exposure, and accidental releases of tritium, as well as conventional hazards, were analyzed qualitatively in the AGHS PSAR, and this was endorsed by SRD before the final plant design was started. For the second phase of the safety analysis, a Design Safety Review is carried out on each of the AGHS subsystems. The Design Safety Review is the prime mechanism for demonstrating to SRD that the AGHS subsystem as designed will be acceptable. Before proceeding to procurement, a Design Safety Review has been carried out on each subsystem of the AGHS and endorsement of SRD obtained. The Design Safety Review addresses for each subsystem:

1. Containment boundaries during normal operations.
   For accident situations and during maintenance including monitoring provisions:

2. Over-pressure protection:

3. A layout review including the effects of water leaks, cryogen spills, fire, external events, and personnel safety in a qualitative manner:

4. Failure modes and effects analysis (FMMA): 

5. Accident analysis:

6. A list of components important to safety and their test frequencies:

7. Waste arisings.

The accident analysis presents the key results of the probabilistic safety analysis to demonstrate that the design target has been met.

The FMMA is used to identify initiating events which require further analysis, i.e., those where a release of tritium is possible into the building or into a secondary containment. Each component or grouping of components in the subsystem is considered in turn to determine if failure could result in a release or require safety system action to prevent/mitigate a release. All of the foreseen operating modes of the subsystem are considered. The following topics are addressed for each component:

<table>
<thead>
<tr>
<th>Component</th>
<th>Failure Mode</th>
<th>Effects</th>
<th>Means of Detection</th>
<th>Remarks</th>
</tr>
</thead>
</table>

This thorough and systematic review of the entire subsystem provides assurance that all initiating events are identified and that the design is fault tolerant. The FMMA also documents for future operators the effects and indications of possible failures.
The initiating events identified in the FMEA are assessed in more detail, using event trees if required. An example of a typical tree is shown in Figure 3. These identify the safety systems available for preventing/mitigating the release and indicate the consequences of failure. Simple fault trees based on the preliminary design are used to estimate failure probabilities of mitigating systems and initiating event frequencies. A sample fault tree is shown in Figure 4.

Due to the lack of established reliability data relevant to fusion (cryogenic and ultra-high vacuum), a data base derived from generic fission reactor data, modified as necessary, has been developed, discussed with and endorsed by SUD.

Accidental tritium releases are estimated based on the likely maximum inventory available for release for a given accident sequence. In most cases, those releases will be into an intact containment. The AGHS design permits recovery of such releases with only a small (assumed to be 1 percent) residual contamination which will be released during maintenance via Exhaust Detritiation (ED) with a detritiation factor of about 1000. The ED is also used during routine purging of secondary containment. In some cases, accidental releases will be into a failed/leaking secondary containment such that releases into the building will be diffusion-driven. Here, an upper bound of 10 percent of the available inventory is assumed to be lost before long-term recovery actions can be

```
FIGURE 3 TYPICAL EVENT TREE ANALYSIS

OVERTEMPERATURE TRIP 2 X 10 -4
HARD WIRED OVERTEMPERATURE TRIP 2 X 10 -4

D.1 No release. Gas processed as normal.

D.2 Gas recovered and processed as normal. 0.001% released through EDS

D.3 Release. 10% by diffusion and 0.09% released through the EDS

D.4 Total release

FIGURE 4 TYPICAL FAULT TREE USED TO DERIVE INITIATING EVENT FREQUENCY

NITROGEN LEAKS INTO PROCESS GAS AND IS NOT DETECTED

FAILURE OF LEAK IN H20 OR LINDO

LEAK IN LN2 LEAK IN LINDO

FAILURE TO DETECT LEAK

FAILURE OF TRACER LINE 1 LEAK FROM LINE 1 LEAK FROM LINE 3

PRESSURE FAULT LEAK FROM LINE 1 PRESSURE FAULT LEAK FROM LINE 3

FIGURE 4 TYPICAL FAULT TREE USED TO DERIVE INITIATING EVENT FREQUENCY

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implemented. For pressure-driven releases into failed containment or catastrophic failures, total release is assumed. All releases are pessimistically assumed to be in the more hazardous oxide form to allow for possible conversion (HT → HTO) within the room.

To demonstrate compliance with the SHD design target, the event sequence frequency (as determined by the event tree and fault trees) and tritium release (as determined above) are plotted on a frequency versus consequence plot. An example of a frequency versus consequence plot is shown in Figure 5.

**Frequency vs Consequence Plot Impurity Processing Loop**

- **Sequence** Design Target (0.37 TBq a⁻¹HTO)
- **Process element failure** (secondary containment intact)
- **Process element and secondary containment failure**
- **Maintenance operations**

**FIGURE 5 IMPURITY PROCESSING LOOP ANALYSIS**

**HIGHLIGHTS**

Design Safety Reviews have been carried out on seven of the nine major subsystems of the AGHS. All designs have met the design target for accident sequences. Some of the highlights/insights gained from applying risk assessment techniques are noted below:

1. The FMEAs (and a previous Hazard and Operability (HAZOP) study) have confirmed that most valve or control malfunctions will lead to process upsets, but not lead to tritium releases or significantly higher risk of release than normal operation. The AGHS has been designed to allow considerable flexibility in operation to cope with abnormal processing requirements arising from the JET experimental programme, which also allows for recovery from such process upsets.

2. Specific tasks and hazards requiring operator training or procedural controls have been identified by the FMEAs.

3. The design of uranium beds has not been found to pose a significant hazard, despite the pyrophoric form required. Although the uranium bed systems have met the design targets, an experimental verification of deterministic calculations of air ingress events has been requested by SRD. These calculations show that nitrogen blanketing will limit the amount of oxygen available for reaction with uranium for events such as valve opening during replacement.

4. Secondary containment has been shown to be the key safety feature of the design. The importance of maintaining the containment intact and monitoring secondary containment to quickly detect failures has been emphasized. The use of high vacuum secondary containments for cryogenic and uranium bed components and inert gas filled "valve boxes" at sub-atmospheric pressures allows pressure measurements to be used for on-line process boundary or containment failure detection.

5. For a tritium processing plant like the AGHS the key process requirements to prevent a release are to prevent overheating (the AGHS incorporates separate over-temperature heater cut-outs) and to limit pressure excursions (the AGHS incorporates rupture discs venting to evacuated tanks on cryogenic components and pressure switches to trip pumps capable of creating over-pressures).

6. The sequence design target and event sequence analysis have been used to identify specific components needing testing and set preliminary test frequencies for components important to safety.

7. The ability of the plant to tolerate control system failures has been defined, and it has been shown that no credit needs to be taken for actions by the control system for abnormal events, thus lessening the degree of reliability required of the control system and allowing more flexibility in meeting the need for changes to control programming for such an experiment system. All safety related trips and interlocks are provided by an independent hardwired protection system.

8. Effective use of buffer tanks has provided a more defence in depth protection for the release of tritium to the environment.

**SUMMARY**

To date, Design Safety Reviews have been completed for all AGHS process subsystems, except Gas Introduction where the design is still in progress and Exhaust Detritiation where the supplier is preparing similar assessments. SRD have accepted these assessments and endorsed procurement to proceed, but have required clarification and more detail of some aspects prior to commissioning or operation.
The application of risk assessment techniques has demonstrated (within the limits of data availability) that the accident sequence release guideline is met. The importance of secondary containment monitoring and operator training have been highlighted. Testing requirements including frequencies have been derived from the assessments. In summary, the assessment process has contributed to providing good confidence to both JET and the authorities that the plant can be operated to an acceptably safe level.

REFERENCES


ABSTRACT

A heterogeneous Monte Carlo neutron transport code, HEMCNT has been developed in order to calculate neutronic parameters of interest for multi-regioned fusion and fission reactors and neutron shields. In Part I of this paper, calculational methods in HEMCNT are presented.

To verify the current code, benchmarking calculations in Part 2 are compared with the results obtained from CFFTP's Aqueous Lithium Salt Blanket concept for Engineering Test Reactors [1]. Also, the sensitivity of neutronic parameters was investigated by varying the microscopic and differential cross-sections (MCS and DCS) for the case of a 100 cm blanket composed of natural lithium.

Part 3 examines shielding calculations which have been performed for the acquisition of a 14.1 MeV neutron pulsed generator to be located in the Hot Storage Area Room of the Nuclear Reactor Building situated on the McMaster University campus. Radiation dose estimates were calculated as a function of concrete thickness.

In Part 4, the Monte Carlo neutron transport code was used to evaluate the energy deposition, tritium breeding profile and neutron leakage spectrum of a liquid lithium and inert gas mixture flowing in a Fusion reactor coolant channel bombarded by monoenergetic 14.06 MeV neutrons.

INTRODUCTION

The applicability and usefulness of Monte Carlo methods for neutronic calculations stem from two principle reasons. Firstly, since the interactions of radiation passing through matter are stochastic, these processes can straightforwardly be simulated by the use of a random number generator, making decisions based on the probabilities of various phenomena. Secondly, the Monte Carlo method has been a useful computational tool in solving neutron transport problems for complex multi-dimensional geometries without having to evaluate intricate Boltzmann equations.

This paper is divided into four sections and will focus on the application aspects that have been performed with the HEMCNT code.

Part I begins with a brief description of the HEMCNT code. In Part 2 are compared with the neutronic results obtained from CFFTP's Aqueous Lithium Salt Blanket concept for Engineering Test Reactors.

Because of uncertainties reported in nuclear cross sections [2] the sensitivity of neutronic parameters with microscopic and differential cross sections, it is also necessary to have a fundamental data base

In Part 3, the HEMCNT code was used to estimate the neutron flux density as a function of energy and space for a 14.1 MeV neutron pulsed generator (NPG). With the neutron flux known, radiation dose estimates can be determined by using appropriate dose quality factors. Radiation dose estimates are presented as a function of concrete thickness. A reflector is included so as to evaluate intricate Boltzmann equations.

Then, with the use of multigroup cross sections and the logic of the random walk process, important quantities of interest can be recorded. After several thousand histories the averaged results provide an estimate of the "true solution" sought.

Output parameters of the system include: a) neutron flux \( \phi(E) \), b) neutron leakage spectrum, c) equilibrium between the number of nuclear reactions and the heat released by each material, normalized per source neutron, d) reaction rates and e) power density rates.

1.1 Calculational Methods in HEMCNT

The two underlying strategies that Monte Carlo sampling methods are based on are the rejection and mapping techniques. The former is usually employed to describe a decision making event such as the selection of a material or reaction. Table look-up schemes may be employed here in the program. The latter scheme is useful for transforming a random number into a variable such as the azimuthal angle of flight of a neutron or its distance of travel. In both cases, random numbers uniformly distributed between (0,1) are assumed at our disposal.

(i) Isotropically Sampling. The polar and azimuthal angles can be sampled using the mapping technique to obtain

\[ \theta = 2\pi \xi_1 \quad \text{and} \quad \phi = 2\pi \xi_2 - 1. \]

However, for the case of anisotropic scattering, \( \cos \phi \) must be sampled differently.
(ii) Anisotropic Scattering. For energies above 3 MeV, neutron scattering is usually anisotropic. The anisotropic scattering angle can be sampled as follows [5]. Let \( \sigma(\theta_c, E) \) be the cross section for a neutron with energy \( E \) to scatter within a differential angle \( d\Omega \) at \( \theta_c \). Then the probability of a neutron scattering through an angle less than \( \theta_c \) is equal to

\[
\xi = P(\theta_c, E) = \frac{\int_{0}^{\theta_c} \sigma(\theta, E) d\Omega}{\int_{0}^{\pi} \sigma(\theta, E) d\Omega} = \frac{\int_{0}^{\theta_c} 2\pi \sigma(\theta, E) \sin \theta d\theta}{\int_{0}^{\pi} 2\pi \sigma(\theta, E) \sin \theta d\theta}
\]

The scattering angle can be found by equating \( P(\theta_c, E) \) to a random number and solving the equation for \( \theta_c \). This was accomplished by first tabulating \( P(\theta_c, E) \) against \( \theta_c \) and \( E \), then inverting the table to get the value of \( \theta_c \) corresponding to any given incoming neutron energy \( E \) and \( P(\theta_c, E) \) equal to \( \xi \), a random number between 0 and 1.

(iii) Direction Cosines. If \((u, v, w)\) are the original direction cosines of the neutron relative to the x, y and z axes, and \( S \) and \( \phi \) represent a new direction, the updated direction cosines become

\[
u' = u \cos \phi + w \cos \phi \frac{1 - w^2)^{1/2}}{1 - w^2 \cos \phi}
\]

\[
v' = v \cos \phi + w \cos \phi \frac{1 - w^2)^{1/2}}{1 - w^2 \cos \phi}
\]

\[
w' = w \cos \phi - (1 - w^2)^{1/2} \sin \phi \cos \phi
\]

(iv) Path Length Sampling. The probability of a neutron having an interaction within a distance \( I \) is

\[
P(I) = 1 - \exp(-l\Sigma) = \xi.
\]

Equating \( P(I) \), which ranges between 0 and 1, to a random number \( \xi \), we can determine how many mean free paths the neutron will travel,

\[\text{No. of mean free paths} = \ln(1) = l\Sigma\]

(v) Material and Reaction Sampling. Materials and reactions can easily be sampled using the rejection technique. Figures 1 and 2 illustrate the algorithms used. Microscopic cross sections were obtained from the ENDF/B-1975 library [2].

(vi) Neutron Flux. The neutron flux density can be evaluated using either the collision density estimator or the track length estimator. The collision density estimator is given by

\[
\phi(r_k, E_i) = \frac{N_{elk}}{\sum_i \Sigma_i(r_k, E_i)E_iV_k}
\]

whereas the track length estimator is

\[
\phi(r_k, E_i) = \frac{N_{elk}L_{el}W_{el}}{\sum_i E_iV_k}
\]

where the following nomenclature is used:

- \( \phi(r_k, E_i) \) = flux in region \( r \), at energy \( E_i \)
- \( W_{el} \) = weight of the \( n^{th} \) particle
- \( E_i \) = energy of the \( i^{th} \) neutron group
- \( \Sigma_i(r_k, E_i) \) = total macroscopic cross section for group \( E_i \) in region \( r_k \)
- \( V_k \) = volume of region \( k \)
- \( N_{elk} \) = number of particles with energy in group \( E_i \) in region \( k \) with volume \( V_k \)
- \( L_{el} \) = track length of the neutron.

(vii) KERMA: Kinetic Energy Released to Material. Subroutine KERMA calculates the kinetic energy released in the material. It is important to note that HEMCNT does not transport photons, but does deposit their energy locally. In the following sections, we summarize the kinetic equations pertaining to the nuclear heat liberated for elastic and inelastic scattering, neutron capture, charged particle break-up and \((n,2n)\) reactions.

(a) Elastic and Inelastic Scattering. If the target nucleus of atomic mass \( A \) receives an excitation energy \( E \) from a neutron with initial energy \( E_i \), its final energy \( E' \) may be expressed as

\[
E' = \frac{1}{(A+1)^2} \left[ 1 + A^2 \left( 1 - \frac{E'}{E} \right) + 2A \cos \theta_c \left( 1 - \frac{E'}{E} \right)^{1/2} \right]
\]
Hence the total amount of nuclear energy deposited in the material is

\[ E_H = E' - E - \epsilon \]

The neutron scattering angle as measured in the laboratory system is given by the following conversion equation.

\[ \cos \theta_i = \left( \frac{1 + l}{1 + \lambda^2 (1 - \lambda^2)} + 2 \lambda \sin \theta_i \left( 1 - \lambda^2 \right)^{1/2} \right) \]

where \( \epsilon = 0 \).

(b) Neutron Capture. The total energy deposited from a capture reaction is simply given by

\[ E_H = E \cdot Q \]

where \( Q \) is the reaction Q value.

(c) Evaporation Model. The secondary neutron energy distribution of many nuclear reactions such as \((n,2n)\) and charge break-up reactions like \(n(n,\alpha)\) can be modelled by an evaporation spectrum. In an evaporation model the emergent neutron energies are assumed to have the following distribution in the center of mass, COM, system

\[ f(E \to E') = \left( \frac{E'}{E} \right)^{\alpha} \exp \left( -\frac{E'}{E} \right) \]

where \( \theta(E) \) is the nuclear temperature of the nucleus which can be approximated by a Fermi gas model as

\[ \theta(E) = \left( \frac{10E}{A} \right)^{1/2} \]

and where \( I \) is a normalization constant that depends on \( E_{\min} \) (usually assumed to be zero) and \( E_{\max} \). By definition the upper limit \( E_{\max} \) is the maximum energy available to the emerging neutron. The upper limit of energy available corresponds to the excitation of the minimum level \( Q_{\min} \) within the nucleus. It can be shown [6] that

\[ E_{\max} = \frac{A}{A+1} \left( \frac{A}{A+1} - E - Q_{\min} \right). \]

The average emergent neutron energy \( E \) in the COM system may be evaluated as

\[ \bar{E}_c = \frac{\int_{E_{\min}}^{E_{\max}} E' f(E \to E') dE'}{\int_{E_{\min}}^{E_{\max}} f(E \to E') dE'} = \theta(E) \left( \frac{E'}{E} \right)^{\alpha} \exp \left( -\frac{E'}{E} \right) \exp \left( -\frac{\theta(E)}{\theta(E) + \left( E' - E \right)} \right) \]

where

\[ x = E_{\max} / \theta(E). \]

Reverting the COM energy to the laboratory frame yields

\[ \bar{E}_c = \frac{E}{\left( \bar{X} + 1 \right)^2} + \bar{E}_c. \]

(d) Charged Particle Break-up Reaction. Charge particle break-up reactions are generally written in the form

\[ X_Z A_1^{A_1} (n, \alpha') \alpha_1, \alpha_2, ..., X_Z A_2^{A_2}. \]

Assuming that the neutron is emitted first, its energy \( E_n \) can be determined by the evaporation model previously discussed. Since we are not interested in the equipartition of energy between particles, the total nuclear heat deposited can simply be written as

\[ E_H = E + Q - E_n. \]

For the cases where there is no secondary neutron emission, as in the \( n(\alpha,\alpha') \) reaction, then \( E_n = 0 \) in the above equation.

1.2 HEMCNT Flow Structure.

Figure 3 is a computer flow diagram summarizing the sequence of events that can take place when a neutron is followed from one collision to the next. A simulation run time depends mainly on 1) the number of neutron histories, 2) the number of energy groups, and 3) the number of zones used. Typical CPU times are of the order of 200 seconds on the VAX-8600 for \( 10^4 \) neutron histories, using 20 energy groups and 1 meter thick breeder blanket divided into 5 zones.
2. FUSION REACTOR BLANKET DESIGN FOR HOMOGENEOUS MODELS

2.1 Benchmarking Results

Figure 4 illustrates the arrangement and composition of the aqueous lithium salt blanket for Engineering Test Reactors proposed by Gierszewski et al. [1] that was homogeneously modelled. In the first region the thick stainless steel walls are cooled by light water. Tritium breeding occurs in region two which contains a solution of D$_2$O and 5% Li. Region three is designed for removing the remaining heat with H$_2$O coolant and to act as an efficient neutron moderator/reflecter. As depicted in Figure 5, excellent agreement in the tritium breeding ratio as a function of the steel structural volume in the breeding region of the blanket was obtained. This comparison is based on the findings of a continuous energy MCNP and ONEDANT multigroup analysis [1].

![Figure 4](image_url)

**Figure 4.** Representative arrangement of the aqueous salt breeding blanket depicting thick stainless steel 316 walls and thin breeding region.

![Figure 5](image_url)

**Figure 5.** Tritium breeding ratio as a function of steel structural volume in region 2 of the breeder blanket shown in Figure 4.

2.2 Sensitivity of Neutronic Parameters with Microscopic and Differential Cross-Sections

The sensitivity of the microscopic and differential cross-sections (MCS and DCS) on the tritium breeding ratio (TBR), energy deposition distribution and neutron leakage have been investigated for a conceptual fusion reactor blanket.

For the geometry shown in Figure 6, a 100 cm thick spherical breeder blanket with an inner radius of 300 cm and with 14.06 MeV neutrons emitted radially outwards from the center, the results presented in Table 1 indicate that a 10% change in the lithium-7 MCS produces a 3.5% change in TBR, a 1.6% change in the EMF and a -2.7% variation in the neutron leakage fraction. Note however, a similar 10% variation in the MCS of Lithium-6 yields very little change on the original neutronic parameters.

![Figure 6](image_url)

**Table 1.** Sensitivity of neutronic parameters with microscopic and differential cross-section for a 100 cm natural lithium breeder blanket.

<table>
<thead>
<tr>
<th>CASE</th>
<th>T$_x$</th>
<th>T$_y$</th>
<th>T$_{x+y}$</th>
<th>Neutron Leaks</th>
<th>E(15MeV)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural Lithium</td>
<td>0.532</td>
<td>0.771</td>
<td>1.30</td>
<td>0.554</td>
<td>14.88</td>
</tr>
<tr>
<td>dT$_x$=10%</td>
<td>No change</td>
<td>12%</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>dT$_y$=10%</td>
<td>0.544</td>
<td>0.794</td>
<td>1.34</td>
<td>0.541</td>
<td>15.12</td>
</tr>
<tr>
<td>dT$_{x+y}$=10%</td>
<td>0.521</td>
<td>0.727</td>
<td>1.25</td>
<td>0.570</td>
<td>14.65</td>
</tr>
<tr>
<td>isotopic</td>
<td>0.647</td>
<td>0.589</td>
<td>1.24</td>
<td>0.402</td>
<td>16.96</td>
</tr>
<tr>
<td>$\mu$=cos90°</td>
<td>0.531</td>
<td>0.810</td>
<td>1.12</td>
<td>0.788</td>
<td>10.20</td>
</tr>
<tr>
<td>strong forward scattering</td>
<td>0.292</td>
<td>3.81</td>
<td>2.91</td>
<td>2.06</td>
<td>1.32</td>
</tr>
</tbody>
</table>

Changing the scattering differential cross-sections altered the neutron leakage fraction, the total TBR, and the energy multiplication factor (EMF) including its radial deposition profile within the blanket. The error in assuming an isotropic DCS greatly lowers the T7 but increases the T6 contributions. Here T6 and T7 designate the number of tritons produced per incident neutron via the n(Li,at)

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or the n(³Li,α)³He reactions. The EMF is increased by 14% while neutron leakage decreases 27%. As expected with more large angle scatterings, the energy distribution has a greater peak close to the first wall. For the extreme case of strong forward scattering the fraction of neutron leakage is very "high" and therefore the energy deposited in the material is low. The T7 contribution is slightly greater because the average energy loss between scatterings is less but the T8 value is significantly lower due to the harder neutron spectrum.

3. NEUTRON SHIELDING CALCULATIONS

The purpose of this section is to report on shielding calculations performed for a proposed installation of a neutron pulsed generator (NPG), to be located in the Hot Waste Storage Area of the Nuclear Reactor Building situated on the McMaster University campus. The acquisition of a 14 MeV neutrons pulsed generator will enable research studies at McMaster University in the fields of 1) two phase flow diagnostics 2) fusion reactor blanket studies, and 3) neutron activation analysis of oxide materials. The neutron generator is a small low power (300 Watt) consuming portable unclassified accelerator developed by Sandia National Laboratories. Such a unit can produce 10⁶ neutrons/pulse at a maximum repetition rate of 12 pulses/minute. However, due to tube lifetime restrictions and operational costs, only 1000 pulses/year are envisaged. The neutron pulsed generator consists of a miniaturized self-contained linear accelerator in which deuterium ions are accelerated onto a tritium target, producing a fusion reaction and approximately 10¹⁰ neutrons of 14 - 15 MeV in energy. The maximum repetition rate for the neutron pulsed activation accelerator is 12 times per minute. Each pulse or burst of neutrons is 1 millisecond in duration and can be used to activate materials.

In order to ensure adequate shielding protection to personnel the neutron generator will be kept in the Hot Waste Storage Area of the McMaster Nuclear Reactor Building. To monitor the neutron output, a semiconductor neutron detector is built into the NPA generator package. In addition, a ³He detector, located in the vicinity of the control unit, will simultaneously be used to monitor the thermal neutron fluence. From these readings and our HEMCNT analysis, one can then deduce the total dose to the operator, albeit very small. A single operator (stationed approximately 10 meters away) may remotely and efficiently operate the generator.

In order to take into consideration the thermalization of the high energy neutrons and take into account the albedo of the Hot Storage Waste Room as well as the external walls of the Nuclear Reactor Building, a Monte Carlo neutron transport analysis was performed. The geometry selected to model the NPA generator's shielding is shown in Figure 7. For simplicity, the concrete walls were simulated using H₂O since their macroscopic cross-sections are similar. Several cases were studied by varying the thickness of the wall with and without a water reflector. The Hot Storage Area walls were assumed to be 1 meter from the source. In order to model the room's differences in wall and ceiling thicknesses, the Hot Storage Waste Area walls were assumed to be between 50 and 50 centimeters thick one meter from the point source. For a better estimate of the dose, a 50 cm thick H₂O reflector shield, simulating the external walls of the Nuclear Reactor Building, was artificially positioned 10 meters away.

A 20 energy group representation yields satisfactory energy-flux and energy-dose characterizations without exacerbating CPU charges. Two thousand neutron histories were deemed necessary to obtain adequate statistics for the spherically symmetric cases studied. The neutron flux was estimated using both the tracklength estimator and the collision density estimator. Corresponding dose rates were determined using Figure 8 [7]. Figure 8, taken from Title 10 of the Code of Federal Regulations, Part 20 (10CFR20) gives the neutron flux which will deliver a dose equivalent of 2.5 millirems per hour, i.e. 5 rem/year. Large variations in the dose equivalent per neutron as a function of neutron energy exist. This occurs since, in evaluating the dose equivalent from neutrons, account must be taken of the different quality factors associated with the various secondary radiations. The radiation dose from neutrons is delivered in the form of secondary radiations produced by neutron interactions. At low energies, these are primarily the gamma rays produced by neutron capture in the constituent elements of the body. For fast neutrons, the recoil nuclei created by elastic scattering interactions are the ionizing secondary radiations of greatest importance [7].

![Figure 8](image-url)  
**Figure 8.** The above values taken from 10CFR20 give the neutron flux as a function of energy that will deliver a dose of 2.5 millirem/hour. If the energy dependent neutron flux is known, the figure can be used to provide an estimate of the corresponding dose rate.

The thermal and 14.06 MeV components of the neutron flux as a function of radial position from the NPA generator are shown in Figure 9. It can be seen that the flux decreases with wall thickness and radial distance. Similarly, the neutron dose in Figure 10 illustrates this behavior. Moreover, it can be noted that most of the dose is due to the high energy neutrons because of its associated quality factors. The total dose 10 meters from the NPA generator ranges between 7.3 and 21.5 millirems/year for the two cases discussed. Figures 11 and 12 depict the neutron flux and neutron dose as a function of energy a) inside the room, b) directly outside the room and c) within 8.5 to 10 meters of the accelerator. Here again, it is clearly visible that most of the dose is due to the high energy flux even though there is significant neutron thermalization. Furthermore, it has been found that the effect of removing the reflector shield decreases the dose to 3.3 and 16.7 mrem/yr for the 50 and 30 cm thick walls. Increasing the reflector thickness beyond the 50 cm value chosen for the external walls of the Nuclear Reactor Building did not significantly change the total spatial dose distribution. Further investigations in varying the thermal microscopic scattering and absorption cross-sections showed little influence on dose. However, the thermal neutron flux profile does change and this parameter essentially governs how many counts the Helium-3 detector will register.

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**Figure 7.** The spherically symmetric geometry selected to model the Hot Waste Storage Room and the external walls of the Nuclear Reactor Building.
4. FUSION REACTOR BLANKET DESIGN FOR HETEROGENEOUS MODELS

Over the years, several liquid metal (LM) and liquid metal gas two-phase (LMG-2P) flow coolants have been investigated for possible uses in future fusion reactors [8-11]. Bender and Hoffman [9] postulated that a mixture of liquid metal and inert gas can significantly reduce the effective electrical conductivity of the mixture such that the MHD forces would essentially vanish. This idea was initially borrowed from the early works on LM-MHD studies and the mechanism works as follows. A coolant in a fully dispersed phase i.e., mist flow, exhibits the electrical conductivity of the gas phase which is orders of magnitude lower than the electrical conductivity of the single phase LM. Since the ratio of LM density to gas density is in excess of 500, only a small gas quality (mass of gas/total mixture of mass fraction) is required. The entrainment of the LM by the gas phase and the relatively lower MHD forces encountered result in reduced pumping power requirements. This consequently disburdens the earlier mentioned pumping difficulties associated with LM coolants.

However, trying to alleviate the MHD pressure drop by introducing a gas phase with large void fractions beyond 85% in reactor coolant channels may create new problems. Firstly, and probably most importantly is that the proposed scheme may lead to structural failure if the heat flux accidentally ventures beyond the critical heat flux (CHF). Under CHF conditions, the LM droplets or the thin film inside the pipe may quickly vaporize making the cladding temperature unstable and susceptible to local hot spots and may lead to a burn-out situation. Secondly, reactor blankets must be of the order of 1 meter in thickness if they are to breed sufficient quantities of tritium, as well as to thermalize and attenuate the energetic neutrons before they enter the shield region. To satisfy these constraints while maintaining large voids may make the reactor too bulky and economically unfeasible. Furthermore, the oversized current carrying magnetic coils would create enormous forces and stresses beyond their yield stress limit (about 290 MPa for copper), vastly as the void fraction increases, the TBR decreases as the ratio of lithium to structural material decreases.

Figure 13 illustrates a transverse cross-sectional view of the flow channel. In this example, an annular flow regime is depicted. The thin film is partitioned into five concentric rings and each annulus is further divided into eight sectors. The numbering of these sectors is as indicated and will not change with respect to the x-y axis. On the other hand the N-E-S-W directions may be rotated but the 14.06 MeV neutrons are always incident from the bottom side i.e., sections 5 through 8.

Figures 14 and 15 represent the local radial tritium breeding ratio and the local energy distribution for three annular flow conditions inside a pipe of 1.575 cm. radius. The void fraction was varied between 1.56, 56.23 and 76.56 percent in a step-wise manner. Figures 14 and 15 show that the energy (or tritium) deposited in the concentric rings increase geometrically with increasing volume.
Figure 13. Channel geometry indicating zone numbers and radial regions for annular flow. The magnetic field is orientated in the N-S direction while the 14.06 MeV neutrons are incident from the bottom i.e., sectors 5 through 8.

Figure 14. Tritium breeding ratio profiles in an annular flow regime for various void fractions: a) V.F.=76.56%, b) V.F.=56.25% and c) V.F.=1.56%. In the following calculations the flow channel radius was R=1.575 cm.

The energy distribution in large pipes is nonuniform. A cylindrical channel, similar to the one shown in Figure 13, but with a 10.0 cm radius and a 1.25 cm thick film, yielding an equivalent void fraction of 76.56%, was irradiated by a neutron flux. Figure 16 clearly indicates that more energy is deposited in sectors (closest to the impinging flux) 6 and 7 than in zones 1 and 4. This can easily be understood because the 14.06 MeV neutron flux is directly incident on the surfaces of zones 6 and 7. Only an attenuated flux can reach sectors 1 and 4. Thus, the larger the pipe and the lower the void fraction the more pronounced this effect will be.

Figure 15. Energy deposition profiles in an annular flow regime for various void fractions: a) V.F.=76.56%, b) V.F.=56.25% and c) V.F.=1.56%. In the following calculations the flow channel radius was R=1.575 cm.

Figure 16. Energy deposition in zones of a coolant channel with an annular flow regime that has a 10.0 cm radius and a 1.25 cm thick annular film.
The radial energy distribution in zones 1, 2, 7, and 8 is shown in Figure 17. The amount of energy deposited in these zones depend not only on the volume of the concentric rings but also on the mean free path (MFP) of the neutron. The MFP for 14.06 MeV neutrons in natural lithium is 14.53 cm. Smaller channels of the order of 1 cm essentially appear transparent to these high energy neutrons. Therefore a non-symmetrical heat generation term within the LMG-2q flow coolant should only be expected in large pipes of the order of one neutron MFP or more.

Figure 17. Radial energy distribution in zones 1, 2, 7, and 8 for the annular flow regime case shown in Figure 16.

Figure 18 indicates the number of neutrons that leaked out of each sector of the annulus. It can be noted that most of the neutrons pass right through the channel, i.e., only a very small percentage are backscattered. Figure 19 is a neutron energy leakage spectrum of the flow channel. The figure confirms that about 90% of the neutrons, for this case, pass through unimpeded. Such a result indicates that natural lithium is a poor neutron moderator and that it would probably be beneficial to choose a two phase flow pattern that had as low a void fraction as possible if the reactor assembly is to be kept compact. The optimum conditions must be calculated by maximizing the neutronic parameters in terms of heat transfer rates and MHD pumping power losses.

Figure 18. Normalized neutron leakage fraction for the case shown in Figure 16 as a function of zone number. Note, most of the neutrons pass straight through the channel.

Figure 19. Neutron energy leakage spectrum for the case shown in Figure 16.

CONCLUDING REMARKS

A Monte Carlo neutron transport code has been developed, tested, and applied towards homogeneous and heterogeneous models. HEMCNT neutronic results obtained for the aqueous salt breeder shield agreed well with results reported by Gierszewski et al.

A sensitivity analysis performed on the microscopic and differential cross-sections of natural lithium for a 100 cm thick blanket caused relatively minor changes on the neutronic parameters.

The 14.06 MeV NPG study revealed that the radiation dose to personnel, based on 1000 pulses/year, — located 10 meters from the accelerator shielded by 30 to 50 centimeters of concrete one meter from the neutron point source -- ranges between 10 to 30 millirem/year.

The HEMCNT code was also used to evaluate the tritium production, neutron leakage and energy deposition profile of a liquid lithium and inert gas annular two phase flow mixture within a single coolant channel. Numerical simulations indicate that the TBR and total energy deposited decreases with increasing void fraction and vice versa for neutron leakage.

The volumetric nuclear heat generated in the larger flow channels was shown to be concentrated towards the first wall region directly incident to the neutron flux. In this case it is speculated that the internal heat source will not only increase the bulk temperature of the LM but may alter the temperature profile and heat transfer coefficient. This effect may reduce the heat transfer coefficient and result in higher film and temperature drops.

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RF CURRENT DRIVE AND HEATING ON THE TOKAMAK DE VARENNES

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ABSTRACT

A lower hybrid system is under development for current drive and heating on the Tokamak de Varennes. Near steady-state RF current drive and/or transformer recharge should allow long pulse operation of a diverted plasma for up to 30 sec at nearly full machine parameters ($I_p = 250$ kA, $n_{e0} = 3 \times 10^{19}$ m$^{-3}$). The system design is for 1 MW of RF power at a relatively high $N$ (variable around 3.1) in a narrow spectrum (0.4 FWHM) at 3.7 GHz. Design considerations and technical solutions are presented.

INTRODUCTION

The Tokamak de Varennes (T de V) started operation in March 1987. The machine has been operating reliably in the Ohmic regime and all the initial design criteria have been met. Typical plasma parameters are: major radius $R_\text{maj} = 0.86$ m, minor radius $a = 0.24$ m, toroidal field $B_t = 1.5$ T, plasma current $I_p = 250$ kA, central electron density $n_{e0} = 3 \times 10^{19}$ m$^{-3}$ and central electron temperature $T_e = 0.85$ keV. The main areas of research during the first year and a half of operation were internal kink modes, biasing of the plasma, vertical current effects, plasma-wall interactions, impurity control and material studies. The T de V is now being upgraded for fast plasma positioning and divertor operation. It should resume operation in September 1989.

One major characteristic of the T de V is that all the equilibrium field coils and power supplies are designed for long pulse operation, although the present Ohmic regime allows only ~1 s pulses. In order to take full advantage of this long pulse capability, a non-inductive current drive system is now being developed on the T de V. The planned lower hybrid current drive (LHCD) system should allow operation of the diverted plasma at nearly full machine parameters for about 30 s. In addition, it should provide significant heating of the plasma. Implementation of LHCD on the T de V should insure the long-term relevance of research programs on impurity control in divertor plasmas, plasma-wall interactions and materials development for fusion devices. Furthermore, it will open areas of research on non-inductive current drive such as current profile control, mode stabilization, plasma startup and transformer recharge with the divertor, all of which are important to future reactor operation. This paper describes basic considerations and features of the LH system design.

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**Fig. 1.** Schematic view of the antenna.
The LH system of the T de V will use 2 klystrons for a total power of about 1 MW for 30 s or 1.4 MW for 5 s. The 3.7 GHz frequency should be high enough to avoid density limitations for most cases on the T de V. The grill antenna will have to launch waves with relatively high N (c/v ph) because of poor wave accessibility towards the plasma center at low toroidal fields. For example, with n_e=3x10^{19} m^{-3} and B_t=1.5 T, a N greater than 3.0 is required to reach the plasma center, proportionally reducing the current drive efficiency \( (\eta_{\text{eff}} = n \eta_{\text{ref}} / P_{\text{RF}} = 1/N^2) \). With n_e=2x10^{19} m^{-3} and R_{sep}=0.86 m, this implies that about 1 MW is required to maintain the plasma current at 200 kA if the LHCD efficiency is 0.35 \( (10^{19} \text{ m}^2\text{A/W}) \). Such an efficiency is consistent with typical experimental results \( (\eta_{\text{eff}} = 1 - 2) \) and simulations performed with a modified version of the Bonoli-En- glade code. In order to obtain maximum efficiency at all plasma densities, the N spectrum launched by the antenna should be adjustable.

Several current drive scenarios are considered for the 30-s pulse duration. In addition to the long pulse condition entirely with RF current drive, an alternating cycle of current drive-enhanced ohmic operation (N\sim 3.5, \sim -5 s) with low density transformer recharge (N\sim 2, \sim -5 s) will be examined experimentally. The RF-ohmic part of the cycle would permit higher density operation than pure RF current drive, although at a still higher N. Operation in a higher confinement mode for the plasma energy (H-mode), although not yet demonstrated with LH alone, is also a possibility that could require rapid changes of the N. during an experiment.

A schematic view of the antenna is shown in Fig. 1 and an overview of the complete RF system is presented in Fig. 2. The antenna consists of 2 rows of 8 four-way multijunction modules. One klystron drives 2 rows of 4 modules each, with no phase difference between the top and bottom row. As shown in Fig. 3, each multijunction module is separated into 4 guides with the length of the dividers (E-plane) adjusted to minimize reflections and increase the directivity at the optimum coupling plasma density. An N centered at 3.1 with a directivity of 62% is obtained with the built-in 90° phase difference and a 6.5 mm distance between waveguides (including a 2 mm septum). The height of the waveguides is 72 mm. The spectral width \( \Delta N = 0.5 \) (FWHM) is determined by the total grill width of 234 mm including 2 dummy guides at each extremity. Six high power motor-driven phase shifters can be used to vary the relative phase between adjacent multijunctions from -90° to 70° (2.3\leq N \leq 3.7) in less than 200 ms if required. Typical spectra, obtained from simulations using a multijunction coupling code are shown in Fig. 4. The antenna can be moved ±70 mm with respect to its normal position 20 mm away from the separatrix (R_{sep}=0.86 m, a=0.26 m). The multijunctions will be maintained at 200°C with water cooling/ heating to minimize outgassing. With a 1 MW power input, the power density at the grill mouth is about 4.5 kW/cm². Total power losses from the klystrons to the plasma are estimated to be about 15%, including a few % within the antenna (plated stainless-steel or copper). High power isolators should permit operation of the system with up to 10% of the power reflected back towards a klystron. The transmission line uses standard WR-284 waveguide, pressurized with SF₆, between the klystron room and the antenna.

![Fig. 2. Overview of the system including the RF feedback/protection.](image-url)
Fig. 3. Schematic view of a multijunction module with 3-e plane dividers. Step transformers determine the relative phase between secondary guides.

Fig. 4. N. spectra for different phases $\Delta \phi$ between the multijunctions. The directivity represents the relative energy content of the main peak.

The fast digital feedback and protection for the RF system is described in Figs. 2 and 5. The power and phase control of a klystron is based on continuous measurements performed at various points: incident and reflected power on each multijunction, output and reflected power of the klystron, reflected power on the matching loads of the 3 high power dividers next to the klystron and the relative phase of the first multijunction. These measurements are processed by a fast controller during a $\sim 40 \mu$s cycle to determine the power and phase feedback to apply at the input of the klystron, the power balance on the transmission line and the reflection coefficient at each multijunction for protection purposes. Reference waveforms used for the feedback loops can be preprogrammed and/or calculated in real time from plasma parameters ($I_p, n_e, V_e$, etc.).

A general overview of the control and feedback system is shown in Fig. 5. Both local and remote consoles are available for the user interface and access to the database used to initialize the whole system. Programmable logic controllers (PLC) are used for local control, event sequencing, fault detection recording, slow monitoring and phase control of the mechanical high power shifters.

CONCLUSIONS

The LH system for the Tokamak de Varennes has been designed for reliable and flexible operation in a quasi-continuous regime and should permit significant experiments on long-pulse diverted plasmas. Main features include a narrow spectrum at high $N_e$ that can be changed in real time, and a fast digital control system. The expected commissioning date is 1992.
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Session 14:

Reactor Safety and Licensing

Chairman:

J. Waddington, AECB
The evolution of a distinctive Canadian nuclear safety philosophy is traced from the days of the Montreal Laboratory, during World War II, to the present. Suggestions for future directions are offered.

The Early Days

Safety has been a major consideration since the beginnings of the Canadian nuclear program, almost a half century ago. In 1940, during World War II, and just a year after the announcement of the discovery of fission, C. C. Lawrence started the construction of an "atomic pile" in the old National Research Council building on Sussex Drive in Ottawa. (1)

The pile was built of uranium oxide in small paper bags interspersed among larger bags containing fine petroleum coke. (Fig. 1)

As the senior scientist working with radium and x-rays, Lawrence was well aware of the hazards of working with radioactive materials. Working largely alone and frequently interrupted by urgent war work, it was 1942 when Lawrence and his summer-time associate, B. W. Sargent, realized that they could not achieve criticality - there were too many impurities in the uranium oxide and coke.

Their work, nevertheless, provided valuable background for the international group of scientists who came together in the Montreal Laboratory in late 1942 and gave Canada entry into the restricted developments in the USA. The talented members of the Montreal Laboratory, despite many difficulties, scientific and political, developed the theories necessary to design a heavy water moderated, natural uranium fuelled reactor.

Even at that early stage safety was a concern. Initially the focus was on the biological effects of the radiation that would be associated with the reactor. The concept was developed of keeping radiation exposure as low as practical, some three decades before ALARA (as low as reasonably achievable) became one of the principle tenants of the International Commission on Radiological Protection (ICRP).

In the spring of 1944, after the Montreal Laboratory was re-organized and a long-delayed understanding with the USA had been reached, the decision was made to build the NRX reactor. Subsequently it was decided that a small, simple reactor should be built first to confirm the physics predictions. ZEEP was built and when it started up September 5, 1945, it was the first reactor outside the USA (Fig. 2).

Safety was a consideration in the design of ZEEP but simplicity and flexibility were paramount. As Lawrence described it some years later (2),

"ZEEP was as simple as a reactor could be: an aluminum tank... filled with heavy water, in which was suspended several dozen uranium bars."

The basic safety was in the design - "zero" power meant there would be very few fission products produced and no heat to be removed. Nevertheless, there was a "safety system", a cadmium plate held up by a magnetic clutch connected to a radiation detector.

The NRX reactor, however, at 20 megawatts (thermal) power, presented major problems of heat removal, shielding, and containment of the fission products. With perhaps undue optimism about the reliability of the aluminum sheathing on the uranium fuel rods, once through cooling using river water was chosen. More consideration was given to the Argon 41 that would be produced in the air cooling, although dispersion was used as the means to keep doses small.

Numerous safety devices were installed on the original NRX reactor - to protect against loss of control, inadequate cooling and excessive radiation. Despite these precautions the worst reactor accident in Canadian history occurred on NRX on December 5, 1952, when a power excursion melted fuel and severely damaged the reactor structure.

Laurence was the chairman of the investigating committee, which concluded that there were too many precautions. To quote Laurence again (3),

"It was like piling every available object against a door to brace it against some menace from outside, when a well-designed lock or cross-bar would have served the purpose better."

There was need, he concluded, to identify more clearly the objective of the safety provisions.

Fig. 1

Fig. 2
In 1972, these criteria were modified to make the unavailability requirements more stringent (1 x 10^{-2}) and to combine "protective" and "containment" measures in one category of "safety systems" (2). The reference dose limits were retained even though the presumed frequency of "dual" failures was decreased (Table 2). No further, formal, modification of these criteria has been issued by the AECB and they are still applied, at least nominally.

### Table 2

1972 AECB "Siting Guide"

<table>
<thead>
<tr>
<th>Situation</th>
<th>Assumed Maximum Frequency</th>
<th>Metallurgy in Use in Calculation</th>
<th>Maximum Individual Dose Limits</th>
<th>Maximum Total Dose Limits</th>
</tr>
</thead>
<tbody>
<tr>
<td>Normal Operation</td>
<td>Weighted according to effect, i.e., frequency times dose for unit release</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td>0.5 mSv/yr in body</td>
<td>20 mSv/year</td>
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<td>1 mSv/yr in thyroid</td>
<td>50 mSv/year</td>
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<td>13500 mSv/year</td>
</tr>
</tbody>
</table>

*For other organs use 5 times ICRP annual occupational dose*

*For other organs use 1/10 ICRP occupational values*
would continue cooling for many hours. Consequently, the reliability of the safety system to function as designed became as important as the "availability", although no explicit requirement was expressed.

Designers continually argued that complete separation of systems was impractical. Yet, their designs, an analytical tool called "safety design matrices" (SDM) was introduced. This combined fault-tree/event-sequence approach has often contributed to a better understanding of system behavior and system interactions. Its effectiveness has been limited, however, by the relatively narrow application of the technique to only a few systems.

Over the next decade and a half, the design of nuclear power plants has become much more complex, leading to arguments that the earlier, relatively simple criteria were no longer satisfactory. More and more detailed analyses of postulated events were conducted but no new set of objectives or criteria emerged.

Criteria proposed

There were attempts to develop new or revised sets of principles, criteria, or general requirements in the late 1970's. Pioneered by strong arguments over the design of the Bruce A plant, an inter-organizational Working Group (IOWG) was established under the auspices of the AECB. That group proposed (5) a combination of deterministic requirements, such as separate safety systems, and probabilistic or risk criteria involving six categories of events based on their probabilities, with defined reference dose values for each (Table 3).

<table>
<thead>
<tr>
<th>Category</th>
<th>Individual Effective Dose Equivalent Interval, Event Sequence</th>
<th>Sun of the Probabilities of Occurrence of Failures within the Corresponding Effective Dose Equivalent Interval (Per Reactor Unit per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>$10^{-3}$ - $10^{-2}$</td>
<td>$3.33 \times 10^{-1}$</td>
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<tr>
<td>2</td>
<td>$10^{-2}$ - $10^{-1}$</td>
<td>$10^{-1}$</td>
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<tr>
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<td>$10^{-2}$</td>
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<tr>
<td>4</td>
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<td>5</td>
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<tr>
<td>6</td>
<td>$10^{2}$ - $10^{3}$</td>
<td>$10^{-5}$</td>
</tr>
</tbody>
</table>

The IOWG proposal was not adopted by the AECB, partially because the maximum reference dose was 5 Sv (100 Rem) and partially because AECB staff felt that the techniques and data necessary for probabilistic evaluations were not sufficiently advanced.

Two years later, (in 1980) the AECB issued a set of "consultative" documents on safety requirements for nuclear power plants, three of which specified the requirements for the special safety systems - shut-down, emergency cooling, containment - and the fourth for safety analyses. This last (9) adopted part of the IOWG proposal, but set five instead of six categories and assigned, arbitrarily, event sequences to each rather than define probabilities. This document was used on a trial basis, along with the earlier criteria, for the Darlington NGS.

In 1983, the Advisory Committee on Nuclear Safety (ACNS), which the AECB had established in 1980, issued its report ACNS-4, "Recommended General Safety Requirements for Nuclear Power Plants" (10). This report also proposed the retention of a number of the earlier principles and practices, such as the separation of safety systems, and, even more than the IOWG, proposed a probabilistic approach. ACNS laid out six "risk" categories, each defined by a band of calculated dose, and establishes limits on the sum of probabilities of all the event sequences leading to such a case (see Table 4).

Current situation

The early risk objective has largely been ignored. The principle of separate safety systems has been retained but only limited effort has been extended to ensure the maximum possible degree of independence. As a consequence, despite the introduction of SDM's, there is less assurance that the overall risk objective has been met than would have been achieved. The premise of testability of safety systems and the requirement to test, are only partially followed.

Ironically, other countries, which earlier pursued a very deterministic and arbitrary set of requirements, are now adopting basic risk goals and, more and more, turning to probabilistic risk assessments to demonstrate that the risk goals are achieved.

Probably the single fact of all is that most of the regulatory safety decisions made over the past two decades have not been recorded or filed in an easily retrievable way. Many of these decisions were made with the risk goal in mind and most were probably sound. Collectively they could provide a framework for the future, somewhat analogous to the role of recorded judicial decisions in our common law system, if a determined effort were applied to retrieve and organize these past decisions.

There are some bright spots. The ongoing work, reported elsewhere at this conference, to develop a set of goals, principles and requirements for small reactors, is a positive step towards a coherent approach (11).

In another direction, the concept of using "expert systems" to assist decision making in this area, also described in a paper at this conference (12), is a move towards enabling the experience of the past to be brought to bear on today's problems. Moreover, the knowledge base, which includes the total collection of past decisions and their rational, needs to be developed to make the expert system approach truly useful.

Table 4

Table 4

( from ACNS-4 )

<table>
<thead>
<tr>
<th>Category</th>
<th>Individual Effective Dose Equivalent Interval, Event Sequence</th>
<th>Sun of the Probabilities of Occurrence of Failures within the Corresponding Effective Dose Equivalent Interval (Per Reactor Unit per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>$10^{-3}$ - $10^{-2}$</td>
<td>$3.33 \times 10^{-1}$</td>
</tr>
<tr>
<td>2</td>
<td>$10^{-2}$ - $10^{-1}$</td>
<td>$10^{-1}$</td>
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<tr>
<td>3</td>
<td>$10^{-1}$ - $10^{0}$</td>
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</table>

Again, as for the IOWG proposal, the AECB did not accept the ACNS-4 recommendation, primarily on the argument of unproven probabilistic techniques.

Despite this stand, Ontario Hydro proceeded to conduct an extensive probabilistic analysis of its Darlington Plant. The project was to be very time consuming and expensive exercise but the many design improvements that resulted and the better understanding of the plant achieved reportedly justified the effort.

A different tack was taken in connection with the proposed Point Lepreau II Plant. Here, to avoid the open-endedness of the, "they propose - we dispose", approach of the AECB, the designers entered into a long dialogue with AECB staff in an attempt to define what analyses were required. This process was never completed because of the termination of the project. The progress achieved in the negotiations was reported at the CNS conference of 1985.

CNS 10th ANNUAL CONFERENCE, 1989, 14-3
Future directions

There is a real need to re-examine, and re-state, the objectives of nuclear safety — in design, analysis, operation, and especially in regulation.

The most logical and usable form for expressing the objective is risk, quantified to the extent feasible, with room for judgment where the risk goal and the risk presented cannot be reduced to numbers.

There is a fairly wide consensus about the numerical goal for nuclear safety — expression in different ways but generally being in the order of 10^-6 chance per reactor year of a member of the public being killed due to the operation of the facility. Since the major hazard is the release of radioactivity from a core melt due to partial failure of safety systems, this can be roughly translated into the likelihood of a major release of 10^-7 to 10^-8 per year for the reactor plant from a release that could lead to the premature death of 10 to 100 people.

These are formidable goals, but they are not significantly different from those that have been pursued, nominally at least, over the years. Their achievement poses a challenge. The approach recommended by the IOWG and ACNS appears rational, if augmented by return to some of the principles advocated by Laurence over three decades ago. The IOWG and the ACNS both propose increased sophistication of probabilistic analysis while retaining the concept of independent safety systems.

Laurence stressed independence not for its own sake but as a means of achieving the stringent risk goal with practical systems. With this objective in mind, a higher degree of independence is needed than has been achieved in the past. He also argued for safety systems to be testable and to be tested not only to demonstrate availability and reliability of the normally inactive safety systems but also to provide real data for probabilistic evaluations. There are problems and limitations to testing unless the tests can simulate the environment to which the systems may be exposed. They could lead to a false confidence if used so simplistically. More effective analysis techniques need to be applied in conjunction with appropriate testing. It is clear that Laurence and others working within the IOWG assumptions that components and systems would be designed and built to meet the conditions to which they could be exposed, i.e., be "environmentally qualified" in modern parlance. Unfortunately, this requirement, like other assumptions, was not explicitly stated and, consequently, was forgotten or ignored.

Even if sound safety principles and requirements are clearly expressed, the goal will not be achieved unless there is a consistent and thorough attention to all of the actions required as has been seen in recent papers, for large, complex systems, such as a nuclear power plant, there is a need for an organizational safety philosophy or corporate safety culture.

Epilogue

The nuclear program in this country is entering a period of restraint. The only nuclear developments over the next few years are likely to be in small reactors, a few uranium mines and, possibly, some accelerator applications.

This hiatus could be fortuitous for nuclear safety. It will offer a period when the objectives and prerequisites can be re-considered, and the many safety design, operational and regulatory decisions of the past can be reviewed and critically reviewed.

If this period is used effectively, Canada can again take a lead in the field of nuclear safety and will be well-equipped to meet the challenge of a renewed nuclear program.

References


(3) ibid

(4) Siddall, E. and Lewis, W.E., "Reactor Safety Standards and their Attainment" AECL-498 1957

(5) ----- ‘Reactor Siting and Design Guidelines’ unpublished...AECL file 26-1 O-0-0-0 Nov. 1964

(6) Hurst, D.G. and Bovio, F.C., "Reactor Licensing and Safety Requirements" AECL-1059 June 1972

(7) June 1974


(10) ----- ‘Recommended General Safety Requirements for Nuclear Power Plants’ by Advisory Committee on Nuclear Safety of the Atomic Energy Control Board ACNS - 4 (AECL - INF0 - 0116) June 1983


14-4 CNS 10th ANNUAL CONFERENCE, 1989
ABSTRACT

The questions of maintaining adequate safety and reliability of nuclear power plants and the questions of nuclear power plant life assurance and life extension are growing in importance as nuclear plants get older. Age-related degradation of plant components is complex and not fully understood. This paper provides an overview of the Canadian approach and the main activities and their results towards understanding and managing age-related degradation of nuclear power plant components, structures and systems. A number of proactive programs have been initiated to anticipate, detect and mitigate potential aging degradation at an early stage before any serious impact on plant safety and reliability. These programs include Operational Safety Management Program, Nuclear Plant Life Assurance Program, systematic plant condition assessment, refurbishment and upgrading, post-service examination and testing, equipment qualification, research and development, and participation in the IAEA programs on safety aspects of nuclear power plant aging and life extension. A regulatory policy on nuclear power plants is under development and will be based on the domestic as well as foreign and international studies and experience.

INTRODUCTION

Aging Population Nuclear Power Plants

The nuclear community is facing new challenges as the first generation of commercial nuclear power plants (NPPs) grow old. Some of these plants are approaching or have even gone beyond the end of their nominal design life, which, for most of the first commercial designs, was 25-30 years. Worldwide, a projection from the current NPP age profile shows that, in 1990, 37 plants will reach the age of 25 years, and 14 plants the age of 30 years. The respective figures will increase to 160 and 69 plants in the year 2000 [1].

The present age distribution of Canadian NPPs is shown in Table 1. Since there are no new plants on order after the Darlington NPP, the current average NPP unit age of 9 years will be increasing in the foreseeable future. At present, only two Pickering A units are 18 years old, but without new plants by the year 2000, the average Canadian unit age will be 18 years. Because of the strategic importance of nuclear power, the question of maintaining the safety and reliability of rather "old" plants, and questions of plant life assurance and life extension, is thus of growing importance.

Ontario Hydro Experience with Aging of Fossil Power Plants

The performance and reliability of power plants, their systems and components, may decline during the middle and later years of their life. These effects have been observed on Ontario Hydro's (OH) fossil plants and similar trends are expected for nuclear plants[2]. For example, OH's program of plant life extension at Lakeview Thermal Generating Station has revealed a number of age-related problems with plant equipment. A major cause of these problems was lack of preventive maintenance. To extend Lakeview's service life by 15 years, a $1.1 billion program of rehabilitation and replacements has been started at this station. A similar plant life assurance program has been initiated at OH's Lambton Thermal Generating Station. The reliability of this plant has decreased over the last few years - the forced outage rate went up from 2 to 7 percent in the last 6 years. It is expected that OH may have to spend over 300 M$ to ensure safe and reliable service for the 35-year nominal service life.

NPP Age-Related Issues

The term "aging", as used in NPP aging and life management* work, refers to the continuing, time-

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* NPP "life management" is used in a broad sense and includes plant life assurance as well as life extension.
dependent degradation of materials and plant components** due to plant service conditions. Aging phenomena include chemical and physical processes, such as corrosion, creep, fatigue, irradiation and thermal embrittlement, and service wear. These processes occur under design service conditions (which include environmental, loading, and power conditions) and may be accelerated by departures from the design specifications caused by improper storage, faulty installation, excessive testing, incorrect operation and poor maintenance. All NPP materials experience aging effects to a greater or lesser degree which leads to functional degradation of plant components. The impact of this age-related degradation has to be assessed in terms of answers to the following questions:

1. Is the plant safe to run?
2. Is the plant equipment in sufficiently good condition that the risk of catastrophic failure is reasonably low?
3. Is the plant equipment in sufficiently good condition to ensure high reliability?
4. Is the plant economic compared to other nuclear or fossil plants, or new generating stations?
5. Is the plant able to meet regulatory, political, and social requirements?

Because of the potential negative impact of the aging of NPP components on plant safety and reliability and the potential economic benefits of NPP life management, various organizations in many countries are devoting significant resources and effort to better understanding and managing the effects of aging. This paper provides an overview of Canadian programs designed to understand and effectively manage NPP aging. The paper emphasizes a systematic and an integrated approach to managing age-related degradation of NPP components.

**Canadian Programs on NPP Aging**

The goal of the Canadian programs on understanding and managing aging of NPP components is to maintain plant safety, reliability, and economics throughout the entire plant service life, that is, the nominal design criteria and any extended service life. Utilities have been using a number of programs such as preventive maintenance, inspection, monitoring, periodic review of NPP performance, and feedback of operating experience to detect, remedy, and mitigate the effects of failure of NPP components due to any cause including the effects of aging. These methods have been quite effective in dealing with aging up until now. However, because of the increasing age of plants, there is a need for integrated and systematic programs to understand and manage aging. The major Canadian programs in the area of NPP aging are listed below. All of the programs discussed below are pertinent to both safety and reliability aspects of NPP aging, although to a different extent.

1. Operational Safety Management Program
2. Nuclear Plant Life Assurance Program
3. Plant Refurbishment and Upgrading Programs
4. Aging Studies Using Components from Decommissioned NPPs
5. Equipment Qualification
6. Research and Development

(7) IAEA Aging Program Participation
(8) Regulatory Policy Development

The programs under items 1, 5, and 8 are aimed primarily at managing the aging of components in safety-related systems. The Nuclear Plant Assurance Program on the other hand, focusses on managing the aging of a relatively few "big ticket" plant components. Programs under items 3, 4, 6, and 7 are of a generic nature and the results are applicable to understanding and managing aging of all plant components. The highlights of these programs are presented in the following sections.

**Operational Safety Management Program**

The operational safety management program [3] consists of a preventive maintenance program; the Significant Event Report System; and a reliability-based assessment of the performance of safety related systems. These methods play an essential and complementary role in preventing, detecting, correcting, and mitigating failures of NPP safety-related systems and components from all possible causes, including aging, and in achieving a high level of plant safety and reliability.

Preventive maintenance programs include inspections, testing, surveillance, repair, and replacement activities designed to preserve functional capability of NPP components for operational and emergency use. These activities are used to a greater extent in maintaining the required level of performance of safety equipment compared to that of process equipment. They are also the primary means of detection and mitigation of aging effects in NPP components.

In Canada, at present, most of the preventive maintenance is in the form of scheduled maintenance which is based on a call-up system. In future, greater emphasis will be on predictive maintenance which is based on condition monitoring of selected components. Upgrading of present record keeping systems is planned to facilitate analysis of relevant data (condition monitoring data, operational and maintenance history data, baseline data and acceptance criteria) and reporting of trends so that the extent of component degradation can be determined and a decision on the type and timing of maintenance made.

The purpose of the Significant Event Reporting System is to control and limit the risk associated with NPP operation through systematic detection, analysis and correction of deficiencies, including those caused by aging degradation:

- a multi-level diverse screening process includes reviews of Significant Event Reports (SERs) by plant staff and management, different head office groups, a senior management review committee (when needed), and regulatory staff;
- trends of equipment and system deficiencies, identified by the SERs, and their causes are monitored; and
- lessons learned from SERs are communicated to other CANDU stations, owners, designers, and equipment manufacturers.

The effectiveness of an SER system to identify and evaluate component failures caused by aging effects is limited by lack of readily retrievable relevant data,
such as failure mode and cause, service life, service conditions and baseline data. The above upgrading of operational and maintenance record keeping systems related to preventive maintenance would also improve the efficiency and effectiveness of the SER system in identifying age-related component failures and most appropriate corrective actions.

Reliability-based assessment of performance of safety-related systems is an important part of the annual comprehensive and systematic review of nuclear power plant operation and maintenance. It allows identification of deteriorating performance and corrective actions based on analysis of such performance. Performance is measured against standards or targets which are established by the utilities on the basis of regulatory requirements, internal and external experience, international recommendations and other considerations.

For poised systems, that is, systems which are normally inactive and are triggered on demand, a test program is developed and implemented to provide confidence that the system will work when needed. Components or functions to be tested and the test frequency are determined using reliability analysis and adjusted on the basis of in-service experience.

Reliability surveillance groups at NPPs monitor daily test records, control room logs, work reports and deficiency reports to ensure that all reactor safety deficiencies are properly recorded, classified as to their severity, and corrected.

To assess performance of posed safety-related systems, several reliability indicators are evaluated and monitored, including system actual inoperability and system actual and expected unavailabilities. The component failure rate monitoring program annually evaluates component failure rates and compares them to component lifetime experience and generic failure rate data.

Monitoring and assessing performance of process systems involves recording and trending of both actual and near-miss serious process failures. Should either of these trends indicate a potential problems, a more detailed review of all component faults of a system is carried out to determine major contributors to undesirable performance and appropriate follow-up actions.

NUCLEAR PLANT LIFE ASSURANCE PROGRAM

A formal Nuclear Plant Life Assurance (NPLA) program [4,5] has been initiated for OH's nuclear power plants with the following main objectives:

1. To maintain the long-term reliability, availability, and safety of OH's nuclear plants during the nominal service life of 40 years (life assurance).
2. To preserve the option of extending the life of OH's nuclear plants beyond the nominal service life of 40 years (life extension).

The main emphasis of the program is to shift the balance in the mode of operation and maintenance from a reactive mode to one of prediction and prevention. The program focuses on a relatively few major components that are most critical to the long-term reliability, safety, and life of the plant since they cannot be easily and economically replaced. A list of such critical components has been prepared for the lead (oldest) commercial plant, Pickering NGS-A, and is shown in Table 2.

Table 2

<table>
<thead>
<tr>
<th>List of Critical Components for PNGS-A</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Fuel Channels</td>
</tr>
<tr>
<td>2 Steam Generator</td>
</tr>
<tr>
<td>3 Calandria Vessel</td>
</tr>
<tr>
<td>4 Reactor Heads, PHT Piping, Pressurizer,</td>
</tr>
<tr>
<td>5 Calandria Supports</td>
</tr>
<tr>
<td>6 Secondary Piping</td>
</tr>
<tr>
<td>7 Vacuum Building</td>
</tr>
<tr>
<td>8 Calandria Vault &amp; End-Shield Cooling Systems</td>
</tr>
<tr>
<td>9 Cables (power, control, and instrument cables)</td>
</tr>
<tr>
<td>10 Reactor Building</td>
</tr>
<tr>
<td>11 Turbines</td>
</tr>
<tr>
<td>12 Generator</td>
</tr>
<tr>
<td>13 Digital Computer System</td>
</tr>
<tr>
<td>14 Cooling Water Intake Structure</td>
</tr>
<tr>
<td>15 Spent Fuel Bay/Liner</td>
</tr>
<tr>
<td>16 Penetrations and Airlocks</td>
</tr>
</tbody>
</table>

The NPLA program for OH's nuclear power plants has been divided into the three phases of: methodology, scoping, and implementation. Significant input from international studies and various OH Divisions was received and used in the development of the program.

The NPLA methodology for an assessment of age-related degradation and for an engineering evaluation of critical components has been developed. Key elements of the methodology are presented below:

- Identify and understand age-related degradation mechanisms through an analysis of operating and maintenance history, and through research and development.
- Perform an assessment of the present condition of critical components to determine the extent of age-related degradation, and predict remaining service life.
- Initiate new programs of inspection, monitoring, and surveillance to ensure early detection of age-related degradation so that preventive measures can be taken before failure or loss of function.
- Identify changes in operating and maintenance procedures to prolong component service life.
- If required, identify programs of critical component repair, refurbishment, or replacement to restore its degraded functional capability.

The main objective of the scoping phase is to define new NPLA initiatives and long-term studies related to each critical component. The scoping studies are conducted in a systematic and comprehensive manner by following the step-by-step methodology outlined above. The basic data required for scoping studies includes items such as original design documentation, list of potential known and unknown degradation mechanisms, operating and maintenance history data for the component, commissioning and in-service inspection data. The output of the Scoping Phase is a list of new NPLA initiatives related to inspection, maintenance,
The preliminary scoping studies for Pickering NGS-A are now complete. The implementation phase (Phase 3) will begin soon after the acceptance of scoping phase recommendations by the station technical staff.

Canadian nuclear plants are fairly young and the costs associated with the plant life assurance activities have been small to date. The potential long-term benefits of the NPLA program are expected to be high as indicated by preliminary economic assessments of plant life extension, e.g. [6,7]. Thus it is felt that even though the actual costing for OH plants has not yet been done, the program is expected to have a high benefit to cost ratio.

PLANT REFURBISHMENT AND UPGRADING PROGRAMS

Pickering Units 1 and 2 Retubing and Upgrading Program

During the recent retubing outages (1983 - 1988) of Pickering Nuclear Generating Station Units 1 and 2, a major inspection and upgrading program was carried out with the following objectives:

(1) To determine the extent of age-related degradation and the condition of many of the plant critical and other safety-related components.

(2) To upgrade deteriorated components to current standards.

(3) To help formulate future inspection and maintenance programs consistent with the NPLA objectives.

In addition to the above objectives, an upgrading program was also performed to bring a number of plant systems to standards inherent in the more recently built CANDU units. In this sense, the retubing and upgrading program at Units 1 and 2 can be considered a major step in the life assurance activities.

Because of schedule and resource constraints, it was not possible to inspect all critical components included in Table 2. The results of these inspections, as applied to the Ontario Hydro nuclear plant life assurance criteria show that most of the critical components inspected are in good condition, and preventive maintenance programs are effective in maintaining a negligible degradation rate of component materials. However, based on the inspection results and the recommissioning performance tests, it is expected that the units will achieve the nominal service life of 40 years, and there is every reason to believe that the option of extending the life beyond 40 years is a practical proposition.

Pickering Units 3 and 4 Retubing and Upgrading Program

The pressures tubes at Pickering Units 3 and 4 and other nuclear units of OH are made from Zr-Nb material as compared to the Zircaloy-2 material in Pickering Units 1 and 2, and they are expected to be less susceptible to degradation due to a lower rate of deutrium pickup. Recent inspections at Pickering Unit 3, however, have shown one pressure tube to have higher than expected level of deutrium. More data are being collected to determine whether this is a generic problem with Zr-Nb pressure tubes. In the meantime, in spite of insufficient data but because of the manpower, man-rem, retubing outage schedules for all nuclear units, and system considerations, OH has decided to retube Pickering Units 3 and 4 starting in 1989 and 1991 respectively. During this retubing program, critical component condition assessment and upgrading, if appropriate, will be performed. These programs will take into account more fully the objectives of the NPLA program than was possible during similar programs for Pickering Units 1 and 2.

AGING STUDIES USING COMPONENTS FROM DECOMMISSIONED NPPs

An important part of the programs related to understanding the aging characteristics of NPP components and the development of effective detection and mitigation methods is the examination and testing of naturally aged plant components and materials. Thus, systematic NPP component aging studies have been initiated using components from decommissioned plants at NPD and Douglas Point.

NPD Station: Acquisition and Testing of Naturally Aged Components and Materials

After 25 years of operation, the NPD nuclear station was permanently shut down and decommissioned. This offered an excellent opportunity to obtain samples of naturally aged plant components and materials. Since plant components at NPD had been exposed to their operating environment longer than any other CANDU plant, it is believed that testing and examination of these components could help in identifying presently unknown aging mechanisms and thereby improve the long-term reliability and safety of OH's currently operating nuclear plants. This is in spite of some differences in design, materials, and operating conditions. Following the decision to decommission NPD two years ago, an R&D program utilizing selected NPD components was prepared by a working group composed of representatives of the owner - Atomic Energy of Canada Ltd. (AECL), the operator - Ontario Hydro, and the regulator - Atomic Energy Control Board (AECB). The program which is in two stages, is being carried out by Ontario Hydro and AECL staff, and is co-sponsored by OH, AECL, AECB and EPRI.

A methodology was developed to select from the hundreds of plant components and materials those that would have the greatest applicability to current CANDU nuclear plants. Selected components included pressure tubes, samples of large bore PHT piping, concrete samples, cable samples, motors, and in-situ tests on generator, motors, and electrical circuits. The objectives of the proposed programs related to the NPD components and materials are as follows:
(1) To provide information for OH's NPIA program by providing important general perspective on the aging characteristics of nuclear plant equipment and to improve our understanding of the long-term safety, reliability and maintainability of operating CANDU power stations through investigations of potential degradation mechanisms and failure modes.

(2) To identify presently unrecognized (unknown) synergistic aging mechanisms that could have a significant impact on the long-term safety and reliability of operating nuclear power plants.

(3) To help resolve several generic issues of concern to OH, and the regulatory agencies. Examples of such issues include the validation of accelerated aging and remaining life prediction models, and the development of condition monitoring techniques.

The Phase 1 work at NPD related to the retrieval of components and materials and the conduct of in-situ tests is now complete. Phase 2 work covers actual laboratory investigations of these naturally aged components and material samples. The scope of this work is presently being defined by project leaders with input from design, operations, research, and regulatory staff. Full details of the scope, benefits, and relevancy of work at NPD appear in reference [8].

Douglas Point Station: Acquisition and Testing of Naturally Aged Components and Materials

A program of acquiring naturally aged components and materials, and of conducting in-situ tests has recently been initiated at the Douglas Point station. Full details of the proposed program will be available shortly.

EQUIPMENT QUALIFICATION

The role of equipment qualification (EQ) in managing aging and maintaining operational readiness of essential safety equipment is well known. So is the evolution of EQ during the 70's and 80's which resulted in different degrees of EQ in plants of different ages. In Canada, new plants such as the Darlington nuclear station have a comprehensive EQ program which accounts for aging degradation through the concept of qualified life. The qualified life of selected equipment is established in the pre-operational stage and then monitored and maintained by appropriate operational and maintenance measures during the operational stage. Earlier plants have a program to maintain the initial degree of qualification, consisting of scheduled testing, inspection, maintenance, replacement and staff training activities. As well, status of EQ at these plants is being reviewed and upgraded on a selected basis. In particular, steps are being taken to protect existing safety-related equipment against harsh accident environment (e.g. moisture and submersion) where possible, or to replace it with equipment qualified for the expected service environment.

In addition to the above efforts by plant operators to maintain and improve EQ, there is a need to prepare and issue a planned CSA standard on EQ which would facilitate the implementation of EQ to generally accepted standards. A recently published AECC report [9] provides an overview of current EQ issues and practices, including recommendations on EQ standards and requirements.

RESEARCH AND DEVELOPMENT

Although progress has been made towards understanding and managing aging degradation of NPP components there are still many issues not fully understood which require appropriate R&D support. In Canada in recent years, both the nuclear industry and the AECB have approached NPP aging in a proactive way as illustrated by the above examples.

At present, R&D efforts are underway or are planned in the following areas:

- understanding of various material and component degradation mechanisms (e.g. for elastomers, different pressure boundary components and electrical cable insulation);
- development and validation of component remaining life prediction models (e.g. for pressure tubes and steam generator tubes);
- validation of accelerated aging techniques using naturally aged samples of NPP components (e.g. electrical cable insulation);
- development of in-situ repair technology for power plant components (e.g. for various pressure retaining components);
- development of improved non-destructive examination techniques;
- development of diagnostic and condition monitoring techniques for selected plant components (e.g. for electrical insulation of motor and generator windings and for various rotating equipment); and
- development of contingency plans for full scale replacement of major plant components (e.g. steam generators).

The results of this work are documented in numerous reports. We believe that a strong R&D program is a very important element of aging management. R&D can not only resolve specific problems, but can also discover new, emerging problems not yet observed in the operating plants. In addition, good research creates knowledge and competence which enables resolving unanticipated problems as they arise.

To continue receiving, and to maximize, these benefits of R&D, cooperative research projects and programs between Canadian nuclear industry, the AECB, foreign and international organizations should continue and be further developed.

IAEA NPP AGING PROGRAM PARTICIPATION

IAEA commenced activities concerned with safety aspects of NPP aging in 1985 when it convened its first working group on this subject. Canada joined these activities the following year at the Technical Committee Meeting which brought together representatives of eleven Member States and the Nuclear Energy Agency of OECD. Since then, Canada has been one of the more active participants.
In the early activities, which were oriented towards the safety aspects of NPP aging, Canadian contributions included, among others, participation in the Working Group which drafted a state-of-the-art report on safety aspects of NPP aging (to be published by the IAEA), and chairing of the Advisory group which in June 1988 prepared and recommended to the IAEA a multi-year program "Nuclear Power Plant Aging: Safety Aspects and Life Extension". The main objectives of this program are to establish and maintain, under the auspices of the IAEA, a program of international cooperation and information exchange with the purpose of understanding aging degradation processes and of developing methods and guidelines to manage aging. More details about the program can be found in reference [10].

More recently, Canada has provided the chairman for the two consultants meeting (December 1988 and April 1989) organized by the IAEA's Division of Nuclear Power on the general topic of "Nuclear Power Plant Aging and Life Management", and at the Agency's request, will be providing a cost-free expert to manage the 1989 activities sponsored by the Division of Nuclear Safety. As can be seen from the above, over the last three years, Canadian participation in and contributions to the IAEA program on NPP aging and life management have been substantial. It should be recognized that at the same time the relevant Canadian programs discussed in this paper have also benefited significantly from this participation. To stay abreast of new developments, we should continue to contribute to and benefit from this IAEA program.

REGULATORY POLICY DEVELOPMENT

The general safety concern related to the aging of NPP components is that plant safety could be impaired if the degradation of key components and structures is not detected and timely corrective action not taken before loss of safety function occurs. In particular, it is possible that degradation may not be revealed during routine operation and availability testing, but may lead to erosion of defence-in-depth and failure or even multiple failures of redundant components under transient conditions associated with an operational upset or accident when the safety function is suddenly demanded.

The AECB recognizes and acknowledges that the Canadian nuclear industry has initiated and continues to develop the programs to manage both the safety and reliability aspects of NPP aging, as described in the previous sections of the paper. At the same time, the AECB believes that, as the first CANDU commercial plant approaches the midpoint of its nominal service life, it is timely to document a regulatory policy on safety aspects of NPP aging. Consistent with the goal of the Canadian programs on NPP aging described in Section 2, the goal of the AECB regulatory policy will be to enhance the existing measures for maintaining plant safety during the entire and, especially, the later years of plant service life.

An internal AECB consultation process on development of the policy document is underway. It includes considerations of Canadian, foreign and international studies and experience on safety aspects of plant aging. According to standard AECB procedure, the policy proposal will be published as a Consultative Document to obtain comments both from the nuclear industry and the general public. These comments will be then considered by the AECB before releasing the policy in final form.

REFERENCES


CONCLUSIONS

The Canadian nuclear industry and the AECB recognize that aging is an important area of concern, and that reliability of NPP components, and consequently the overall plant safety and reliability may decline during the middle and later years of plant life. To control and manage the negative effects of aging, a number of proactive programs have been initiated. Much remains to be done to understand the known and potential new aging processes, and to develop effective methods for their timely detection and mitigation.

Continued improvement of the individual programs, taking better advantage of relevant experience from other industries, and strong commitment and cooperation of all involved organizations at the national as well as at the international levels will ensure the safe and reliable operation of NPPs through their entire service life.

14-10 CNS 10th ANNUAL CONFERENCE, 1989
Safety Considerations as a Determinant of Management Strategy - A Review of Models and Data

K.R. Weaver, Shaftesbury Scientific Limited

Paper not received.
RECENT TRENDS IN NUCLEAR EMERGENCY PLANNING
IN ONTARIO

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Ministry of the Solicitor General
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Department of Physics
University of Toronto

ABSTRACT

Following the Chernobyl nuclear accident, a Working Group (#8) was set up by the Province of Ontario to re-evaluate the technical basis for nuclear emergency planning. The report of this Working Group was completed in June of 1988. Among its recommendations was that detailed emergency planning be based on a two-tier approach, with two levels of accidents: the Maximum Planning Accident (MPA) and the Worst Credible Radiation Emission (WCRE). The latter concept is new for Ontario and provides for planning for those events whose frequency is very low or unquantifiable (e.g., sabotage) but which could lead to early morbidity or mortality in the general public. In consequence, an extension of the Primary Zone for some reactor sites from 10 km to 13 km was recommended. The possibility of early injury, and the necessity of treating acute radiation casualties, was considered by another Provincial Working Group (#9) and the recommendations included in their report will also be discussed.

INTRODUCTION

The Chernobyl nuclear accident in April 1986 had a direct impact on thinking about nuclear emergency planning in Ontario. With 21 large nuclear reactors within its borders and another 7 within 80 km, the province is seriously interested in nuclear safety and emergency planning to deal with a possible nuclear accident.

The effects of the Chernobyl accident, involving about 200 acute radiation casualties including 31 deaths, and the total evacuation of an area 30 km around the nuclear station, provided a strong incentive to review Ontario's arrangements for dealing with a nuclear accident. Detailed planning for measures such as evacuation has been in place for many years around nuclear facilities in Ontario, but for a 10 km radius. No plans existed for handling and treating acute radiation casualties. Accordingly, in late 1986 and early 1987 the Government of Ontario took several measures to have these issues examined:

- it set up the Ontario Nuclear Safety Review (Commissioner Dr. F. Kenneth Hare) to report on the safety of Ontario's nuclear reactors and on the adequacy of nuclear emergency plans.
- it established Provincial Working Group #8 (Chair: Dr. K.G. McNeill) to examine the adequacy of the technical bases of the Provincial Nuclear Emergency Plan.

...it also constituted Provincial Working Group #9 (Chair: Ms. P. McGee) to recommend the capability and organization needed in Ontario to handle and treat persons exposed to acute radiation doses as a result of a nuclear accident.

TECHNICAL BASES OF PLANS

The Ontario Nuclear Safety Review (ONSR) dealt mainly with the subject of the safety of Ontario Hydro's reactors. On the issue of the technical basis of nuclear emergency plans it worked closely with Working Group #8, and did not attempt to duplicate its work. The ONSR submitted its report before the Working Group had finished, and in it, noting the direction in which the Working Group was moving, made the following recommendation:

"That the Province of Ontario base its nuclear emergency planning on the maximum credible release of radioactive materials" (2).

The Working Group's Approach

The terms of reference given to Provincial Working Group (PWG) #8 required it to recommend an upper limit for detailed emergency planning and preparedness in Ontario based upon a consideration of the following factors:

(1) Scientific assessments of the risk of different types of accidents (in terms of estimated probabilities and effects), and the validity of such assessments.
(2) Risks which cannot be systematically assessed (e.g. those relating to hostile action).
(3) Public perceptions of the danger from nuclear accidents, and public expectations of consequence mitigation.
(4) The probable effectiveness of improvising emergency response measures in areas beyond those for which detailed plans and preparations have been made.
(5) A safety margin, where not already incorporated, to allow for the uncertainties involved.
(6) Any other factors the Working Group could justify as meriting consideration.
Having examined in depth the factors upon which it was required to base its recommendations, the Working Group came to the conclusion that it could not come up with one planning basis which could adequately reflect all these factors. Therefore, it decided to recommend two criteria:

(a) A Maximum Planning Accident (MPA), which would give the maximum consequence of any accident whose probability is above a predetermined value. This predetermined probability of occurrence, or cut-off probability, was to be set low enough to take into account public expectations regarding emergency planning for nuclear accidents. The MPA would then form the basis for the type of emergency planning and preparations currently being done.

(b) A Worst Credible Radiation Emission (WCRE), which would be defined as the maximum consequences possible from any nuclear disaster within the limits of physical and chemical realities. There would be no probability limits set. Emergency planning and preparations should be done for the WCRE only to the extent of mitigating its worst consequences - early morbidity or mortality.

Rationale for the Approach

Even if the probability of occurrence of the MPA is set at several orders of magnitude below the level for which preparations are made for other types of emergencies, this would still be within the band of probability considered in scientific assessments of risk such as PRAs/PSEs. The MPA concept thus would allow detailed emergency planning to be based upon both public expectations as well as on a scientific assessment of the likely hazard. The MPA would cater for the effects of engineering and human failures, assessed in a comprehensive and systematic manner, and thus would provide a rational, justifiable, and economical basis for the expenditure of resources on detailed emergency planning and preparation.

The WCRE indicates the very worst that could happen; the maximum effects possible from any event, however caused or however developed. Therefore, it makes possible accounting for the following:

- deliberate, hostile action
- events explicitly excluded from PRAs/PSEs
- combinations of events which may not have been predicted
- very low probability combinations of events
- the public's desire to be protected from the worst that could happen
- a safety margin

Comparing the WCRE to the MPA it can be seen that the former covers the following range of possible events:

(a) Accidents with extremely low probabilities of occurrence - that is, lower than the MPA's low frequency level.

(b) Events whose probability of occurrence cannot be estimated - that is, those due to hostile action, gross human error, excluded events (earthquakes, floods, fires etc.), and unpredictable events.

The Working Group's recommendation that, within an overall budget for emergency planning, planning and preparation should only be done for the worst consequences of the WCRE is based upon the following considerations:

(a) Society's resources are better spent in trying to remove the causes listed above (through such measures as ensuring effective security at nuclear stations, better training of operators, better operating procedures, effective fire prevention, etc.) rather than in preparing to deal with the effects of such preventable events.

(b) There does not appear to be any sound justification for expending scarce resources on elaborate measures to deal with all the effects of extremely improbable accidents. Such resources are much better spent in protecting people from the possible effects of the less unlikely scenarios covered by the MPA.

(c) However, the most severe consequences of the WCRE (early morbidity or mortality), however remote the likelihood of their occurrence, are extreme enough to warrant consideration in planning, and the preparation of plans to prevent their happening.

The Maximum Planning Accident (MPA)

PWG #8 decided to select the MPA at a probability of occurrence low enough to take into account public expectations that the risk from nuclear accidents be dealt with more comprehensively than that from other types of accidents. It noted that detailed emergency planning is normally impractical for accidents or disasters with a probability of occurrence of less than 10^-2 per year. For nuclear accidents, however, it adopted a probability a thousand times below this one, i.e., 10^-8 per reactor - year or 10^-8 years per multi-unit station of approximately 10 reactors.

A thorough examination of the available data led PWG #8 to the conclusion that calculations concerning severe accidents all give probabilities for severe core damage with intact containment which, within errors, are of the same general order of probability as 10^-8 per reactor-year. As a result the Working Group decided that calculations should be carried out on probable doses resulting from these types of severe accidents with major releases to containment, but taking into account the fact that, even with a prompt-critical power runaway, containment would still be intact. Severe core damage together with loss of containment is too improbable to be considered in the MPA category. Accordingly, PWG #8 came up with certain parameters for calculating the effects of an MPA of the type mentioned above. These are shown in Table 1 (see below).
Using the above parameters, PWG #8 carried out calculations of dose effects for the various nuclear facilities covered by the Ontario Nuclear Emergency Plan. Given in Table 2 (see below) are the results for the three Ontario Hydro stations.

**TABLE 1**

MAXIMUM PLANNING ACCIDENT (MPA) PARAMETERS

<table>
<thead>
<tr>
<th>Reactor Status</th>
<th>Equilibrium</th>
</tr>
</thead>
<tbody>
<tr>
<td>Releases to the Containment</td>
<td>100% of core inventory</td>
</tr>
<tr>
<td>Noble Gases</td>
<td>0.1% of core inventory</td>
</tr>
<tr>
<td>Iodines</td>
<td></td>
</tr>
<tr>
<td>Release Time to Environment</td>
<td>1 day</td>
</tr>
<tr>
<td>Hold-up Time in Containment</td>
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</tr>
<tr>
<td>Pickering</td>
<td>24 hrs</td>
</tr>
<tr>
<td>Bruce</td>
<td>52 hrs</td>
</tr>
<tr>
<td>Darlington</td>
<td>106 hrs</td>
</tr>
<tr>
<td>Plume Release Height</td>
<td>20 m</td>
</tr>
<tr>
<td>Weather (from end of hold up period)</td>
<td>Pasquill F weather for first 6 hours (wind speed of 1 m/s)</td>
</tr>
<tr>
<td></td>
<td>Pasquill D weather for next 15 hours (wind speed of 5 m/s)</td>
</tr>
<tr>
<td>Wind Direction</td>
<td>Steady for 24 hours</td>
</tr>
<tr>
<td>Wind Meander</td>
<td>22.5°</td>
</tr>
</tbody>
</table>

**TABLE 2**

EFFECTS OF THE MPA, 24 HOURS AFTER EMISSION

<table>
<thead>
<tr>
<th>Distance km</th>
<th>Dose Equivalent in rems (1 rem = 0.01 Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Pickering : Effective</td>
</tr>
<tr>
<td>1</td>
<td>400 : 30</td>
</tr>
<tr>
<td>3</td>
<td>120 : 9</td>
</tr>
<tr>
<td>5</td>
<td>50 : 5</td>
</tr>
<tr>
<td>10</td>
<td>15 : 1.5</td>
</tr>
<tr>
<td>15</td>
<td>6 : 0.6</td>
</tr>
</tbody>
</table>
Worst Credible Radiation Emission (WCRE)

The second basis for nuclear emergency planning recommended by PWG #8 was the WCRE. This was defined as the maximum release of radioactivity as a result of a nuclear disaster, however caused, bounded only by physical and chemical realities.

Table 3 (see below)

The parameters adopted by the Working Group for the WCRE are shown in Table 3.

Using the above parameters PWG #8 calculated the dose effects of such events. The results are shown in Table 4 (see below).

Table 3

WORST CREDIBLE RADIATION EMISSION PARAMETERS

<table>
<thead>
<tr>
<th>Reactor Status</th>
<th>Hold-up Time</th>
<th>Plume Release Height</th>
<th>Releaes to the Environment</th>
<th>Noble Gases</th>
<th>Iodines</th>
<th>Rates of Release</th>
<th>Noble Gases</th>
<th>Iodines</th>
<th>Weather Conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>100% of core inventory</td>
<td>1% of core inventory</td>
<td>10% per hour</td>
<td>All 1% in first hour</td>
<td>as in Table 1</td>
<td></td>
</tr>
</tbody>
</table>

Table 4

EFFECTS OF THE WCRE

<table>
<thead>
<tr>
<th>Distance (km)</th>
<th>Pickering (T* = 4 hrs)</th>
<th>Bruce &amp; Darlington (T* = 4 hrs)</th>
<th>Bruce &amp; Darlington (T* = 24 hrs)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Thryoid</td>
<td>Whole Body</td>
<td>Thryoid</td>
</tr>
<tr>
<td>1</td>
<td>1000</td>
<td>3x10^4</td>
<td>300</td>
</tr>
<tr>
<td>5</td>
<td>3.5x10^3</td>
<td>150</td>
<td>7x10^3</td>
</tr>
<tr>
<td>10</td>
<td>1.5x10^3</td>
<td>10</td>
<td>1.5x10^3</td>
</tr>
<tr>
<td>13</td>
<td>1.0x10^3</td>
<td>3</td>
<td>1.0x10^3</td>
</tr>
<tr>
<td>20</td>
<td>0</td>
<td>0</td>
<td>500</td>
</tr>
</tbody>
</table>

Working Group #8's Conclusions

In its report (3) PWG #8 has recognized the conservative nature of many of the assumptions made by it, as well as the sensitivity of some of its calculations to changes in input data. It also acknowledged that its mandate required it to limit itself mainly to technical considerations, and exclude such factors as cost-benefit, political considerations etc.

The Working Group recommended that Ontario adopt the dual basis it was proposing as the technical basis for its nuclear emergency plans. It also recommended that the detailed planning zone be extended from 10 km to 13 km for the Pickering, Bruce and Darlington nuclear sites, and that the province examine the implications of this revised technical basis for other issues such as early warning systems, the distribution and use of stable iodine, a review of Protective Action Levels, etc.

PWG #8 recommended that the planning zone for the Chalk River Nuclear Laboratories remain at 10 km, while that for the Enrico Fermi 2 reactor (in Michigan USA) be re-examined when more data became available (the current zone covers areas in Ontario 22-24 km from the reactor).
ACUTE RADIATION CASUALTIES

Provincial Working Group (PWG) #9 reported on the issue of the capability and organization required to handle and treat acute radiation casualties. It proceeded upon the assumption that the Government of Ontario would accept the recommendation of PWG #8 on the technical basis to be used for nuclear emergency planning.

The Approach Adopted

PWG #9 decided to base its approach to the problem on the following lines:

(a) As part of the Provincial Nuclear Emergency Plan, there should be a plan and adequate preparedness to handle and treat persons who may have been exposed to high radiation doses.

(b) This plan should provide for a basic organization and procedures, which should be capable of expansion and modification to conform to the requirements of the actual situation.

(c) This basic structure should have the capability of handling up to about 1000 persons for screening and diagnosis and up to about 50 persons for treatment of acute radiation syndrome.

(d) The offsite organization should handle both members of the public and nuclear facility staff personnel.

(e) The organization for dealing with this problem should be based upon and integrated with the organization set up under the Provincial Nuclear Emergency Plan to manage the overall response to the emergency.

(f) As far as possible, existing medical facilities should be used in the proposed organization.

(g) The Working Group could only provide the basis or outline of a plan. Its report should be used to develop a detailed implementation plan.

The Casualty Management System

The system recommended by the Working Group was based on a categorization scheme, dividing potential casualties into two groups:

(a) The Obviously Injured Casualty. The handling of such persons was recommended in the manner outlined in Figure 1.

(b) The Uninjured Potentially Exposed Person. Their handling was recommended in the manner outlined in Figure 2.

Working Group #9's Recommendation

PWG #9 recommended (4) that Ontario develop and implement an Acute Radiation Exposure Plan within the context of the Provincial Nuclear Emergency Plan. This plan should be based on the outline proposed by the Working Group, which is illustrated in Figures 1 and 2.

CURRENT STATUS

The recommendations of the Ontario Nuclear Safety Review, and Provincial Working Groups #8 and #9 are currently under review by the Government of Ontario, and decisions regarding them are expected in the near future.

As soon as these decisions are made, Ontario's Nuclear Emergency Preparedness Program will commence implementing them, as well as examining their implications for other areas affected, such as early warning for the public, the use of stable iodine etc.

CONCLUSION

With a large number of nuclear reactors within its boundaries and nearby, Ontario accords high priority to having in place effective emergency plans to deal with a possible accident at one of these sites. To be effective such planning must have a sound technical basis, as well as extended to cover all aspects of an emergency.

The Chernobyl accident caused Ontario to re-examine its nuclear emergency plans. It is expected that the result of this process will be to enhance the province's ability to ensure public health and safety in the admittedly remote eventuality of a nuclear accident.

REFERENCES


14-16 CNS 10th ANNUAL CONFERENCE, 1989
FIGURE 1 - HANDLING THE OBVIOUSLY INJURED CASUALTY

NUCLEAR FACILITY/ MUNICIPAL ORG.

INITIAL ACTIONS. SUMMON AMBULANCE

AMBULANCE SERVICE

STABILIZE

TRIAGE OF PHYSICAL INJURY

PRIMARY CARE HOSPITAL

TRIAGE OF PHYSICAL INJURY

MANAGEMENT OF PHYSICAL INJURY

*With assistance from Diagnostic and Holding Facility

DIAGNOSIS OF RADIATION INJURY*

Contamination Both Exposure

- Insignificant
- Significant
- Significant
- Insignificant

External Pulmonary, Ingested, Impaled

DECONTAMINATE

TERTIARY REFERRAL HOSPITAL

DIAGNOSIS & TREATMENT OF PHYSICAL AND RADIATION INJURY

#Records should be transferred to Diagnostic & Holding Facility, which will recall for follow-up.
FIGURE 2 - HANDLING OF UNINJURED EXPOSED PERSONS

NUCLEAR FACILITY/ MUNICIPAL ORG.

EXTERNAL DECONTAMINATION
ARRANGE TRANSPORTATION

DIAGNOSTIC & HOLDING FACILITY

MONITORING & EXTERNAL DECONTAMINATION
REGISTRATION
EXAMINATION
DIAGNOSIS

Exposure Both Internal Contamination

Significant Borderline/ Uncertain Insignificant

HOLD TILL DIAGNOSED

TERTIARY REFERRAL HOSPITAL

DIAGNOSIS & TREATMENT OF RADIATION INJURY

DISCHARGE, RECALL FOR FOLLOW-UP

14-18 CNS 10th ANNUAL CONFERENCE, 1989
Abstract: A prototype computer program, ARIES, has been developed to demonstrate the use of a knowledge base in support of safety and compliance functions associated with a CANDU nuclear power station. It was developed with an Expert Systems shell that includes a hypertext facility in addition to a rule based and topical representation of knowledge. ARIES includes knowledge bases for shutdown-depth requirements, safety system availability requirements and permitted operating configurations with links to other as yet undeveloped areas. A hypertext facility was used for explanation.

BACKGROUND

In the process of seeking approval either to startup a new reactor or for continuing the operation of existing ones, the AECB licensees submit a formal safety analysis demonstrating acceptable risk. Upon approval, these become formal documents providing the technical basis for licensing the reactor.

Safety submissions are reviewed by members of the AECB staff, with the appropriate expertise, to check for compliance with already established assumptions and methodologies. Also, because of the relatively flexible approach adopted in Canadian Reactor Licensing, they may contain novel methods of assessment which have to be given adequate consideration before approval can be granted. New approaches are often included to replace existing ones or additions or modifications to existing plants have to be represented. Both the scope of these technical Safety Reviews and their continually changing nature put large demands on the reviewers' comprehension as successive submissions are received and reviewed.

In recognition of the complexity of this process a study was initiated to determine whether an expert system could be of assistance by providing a more systematic means of performing the review process.

The primary objective of this study was to investigate the application of an expert system to a typical regulatory function met in Reactor Licensing. Figure 1 illustrates the regulatory domain of a reactor operating license covering design and operation of the plant.

Two areas of the regulatory process were investigated for potential application of an expert system:

- the development of a decision tool which could be used to formulate and express the decisions which are made as part of the approval process for an operating license.
- the development of a compliance monitor which would be used to assist the project officer in ensuring compliance with the requirements of the operating license.

From the outset it was recognized that the procedures for conducting the Safety Review process were not well documented. Furthermore, the knowledge base for these issues was found to be dispersed amongst the considerable correspondence which has transpired over several years of licensing. Thus although there are many factors which are audited systematically using known rules there are many others which are still in
the form of judgements made on specific facilities where compensating or extenuating circumstances are credited.

Because of this, it was decided to emulate the more operational (as opposed to the review) aspects of the regulatory functions where both the rules and the knowledge base are better documented. These are principally found in the area of licensing where AECB requirements have been translated into the operating policies and procedures which define the operating envelope for the reactor. Two topics were selected from the shutdown depth area of licensing. The first topic is the rules and procedures defining limits for safe operation based on the reactor configuration and the status of the shutdown systems. For the purposes of demonstration a rule-of-thumb target of -5 mk was incorporated for the acceptability of shutdown depth for any selected reactor and shutdown system configuration.

The second topic included in the knowledge base is the availability of the heat transport loop isolation function of the emergency core cooling system. This topic is linked to the shutdown depth evaluation because of the assumption of loop isolation when assessing the loss-of-coolant (LOCA) reactivity transient.

Two users of the expert system were identified:

**User 1** - An AECB Project Officer stationed at a nuclear generating station and auditing the operation of the station for compliance with the Reactor Operating License.

**User 2** - An AECB staff safety expert who would develop and maintain the expert system for the Project Officers. Once coded he would also use the knowledge base for reference purposes in safety evaluations.

The Project Officer is routinely confronted with situations where there is a change in equipment status or operating procedure. He must then address the following issues among others:

1) What is the safety significance of the change?
2) Is the plant still within the licensed operating envelope?
3) If the plant is outside the allowed envelope, is it possible to make an exception under certain conditions?
4) What direction, if any, must be given to the plant Operations Staff?

The expert system was designed to assist in resolving these issues when consulted by the Project Officer.

**THE KNOWLEDGE BASE**

The knowledge base for the expert system consists of facts and rules associated with the performance of the shutdown systems and the adequacy of shutdown depth. The 'facts-base' consists of all equipment and plant information necessary for applying the regulatory requirements. It includes among other things:

- configuration of all the reactivity devices
- reactor power level
- reactor power history
- moderator poison concentration
- primary coolant temperature
- D2O purity of the heat transport system

The 'rule-base' consists of all regulatory requirements that are relevant to the demonstration of shutdown system performance. These rules are given in Regulatory Documents and by precedent from past licensing actions. The result is a complex set of rules and interrelated safety issues. Examples of rules are the following:

**Design Requirements.** For events requiring prompt shutdown action, each shutdown system shall be designed such that, acting alone, it can ensure that:

i) the reactor is rendered subcritical and maintained subcritical;
ii) the relevant dose limits are not exceeded;
iii) loss of primary heat transport system integrity shall not result from any fuel failure mechanism.

**Safety Analysis Assumption.** The values of input parameters used in the analysis of each event shall ensure that the predictions of consequences is conservative and applicable at all times by taking account of:

(a) the different plant states for which continued operation will be permitted by the operating procedures;
(b) the uncertainties associated with each parameter.

**Operational Policies and Principles.** The following Operational Policies and Principles are typical of a CANDU station and relevant to shutdown system performance.

(i) The rate of fuelling shall be limited such that the concentration of boron in the moderator is restricted to the equivalent of 5 mk excess reactivity.
(ii) The void coefficient (void reactivity) shall not be modified by adding poison to the primary coolant without the authorization of the Station Manager. The isotopic concentration of the coolant shall at all times exceed 97.43 percent D2O by mass when the reactor is in operation (97.15 atom percent).
(iii) A shutdown system shall only be considered available if, on demand, it is capable of inserting negative reactivity at the rate and to the depth assumed in safety analysis documented in current licensing submissions and if it would be capable thereafter of maintaining an adequate shutdown margin.
Reactor Operating License. The following licence requirement is typical for a CANDU Station and restricts operation to approved states.

- Except with prior written approval of the Board no change which would render inaccurate the description or analyses in the Safety Report of documents listed in the application shall be made to the reactor Shutdown System No. 1, Shutdown System No. 2, containment system, emergency core cooling system or associated systems necessary for the proper operation of these systems.

ARIES DESCRIPTION

AECB Regulatory Information Expert System (ARIES) is a demonstration application. It is designed to test the feasibility of a "regulatory expert system". ARIES provides not only a program for storing and accessing AECB regulatory documents, but it also "applies" these requirements to an analysis of the safety of nuclear power plant operating conditions. It is an automated decision tool to assist AECB staff who must be familiar with the many regulations and requirements governing plant operations and who must understand the complex relations between operating conditions and plant safety.

Specifically, ARIES is a decision tool to assist AECB Project Officers who assess the safety of nuclear generating station operations. It would be used when the Project Officer analyzes proposed changes to operating conditions/procedures and decides whether such changes represent a safe state. To assist these decisions, ARIES must incorporate both an "understanding" of AECB requirements and a "knowledge" of the plant processes and systems. These facilities are built into the "knowledge-based" or expert system approach used in ARIES.

ARIES Development

The primary advantage of an expert system over conventional computer programs is the separation of the knowledge base from the reasoning mechanism or inference engine. This makes the rules and the objects about which they reason far easier to understand and to subsequently modify. As time goes on and the knowledge base is improved, the expert system becomes more and more valuable. Conventional computer programs suffer from the fact that as time goes on they become more and more difficult to maintain, and thereby lose value.

As with conventional computer programs, a most important part of any expert system is the user interface. A good user interface should provide a natural, easily understood environment which instills confidence in the user. Experience has shown that proper use of colour and graphical displays are very helpful in developing a good user interface. Expert system shells can assist considerably in developing an expert system. Shells normally provide the following things:

1. A convenient method for representing rules and objects which the rules reason about.
3. Facilities for building an effective user interface.

The expert system shell chosen for developing ARIES was KnowledgePro (KP). Besides having the above three characteristics, KnowledgePro also (a) integrates hypertext capabilities, (b) has the capability of displaying and interacting with graphics displays, (c) runs on an IBM PC or compatible (with EGA graphics and hard disk), and (d) is relatively inexpensive. Items (a) and (b) were especially useful for developing a good user interface.

KP is a logic programming language, similar to Prolog, which provides a backward-chaining "inference engine" for implementing knowledge-based programs, including expert systems. That is, the KP interpreter is a top-level inferencing procedure for any program defined by a knowledge base of facts and rules.

KP is compatible with other commonly-used DOS packages, such as Lotus 1-2-3, dBase III or PC Paint. For example, KP provides a graphics toolkit for incorporating PC Paint graphics into KP applications. As a result, graphics displays for menus or program output are easy to generate and to edit. KP also includes a 'run' command for calling external procedures or programs written in, for instance, Turbo Pascal. ARIES uses this facility to generate faster screen displays.

Example of Coding

An example of a rule used to decide if a valve is available in the heat transport loop isolation system is as follows:

English form:
If the instrument loop is unavailable or the odd(even) logic is unavailable or the odd(even) test handscrew position is test or the valve control handswcet position is test or open.
Then the odd(even) valve is unavailable.

KnowledgePro form:

```
topic valve_avail_p(valve).	if (inst_loop_avail_p is no)
or (logic_avail_p(valve) is no)
or (test_HS_position(valve) is test)
or (valve_HS_position(valve) is test)
or (valve_HS_position(valve) is open)
then valve_avail_p is no.
elseif valve_avail_p is yes.
end.
```

KnowledgePro stores rules and data objects as so-called "topics". Topics can also be hypertext windows which are displayed whenever the user selects a hypertext sensitive area. As an example of this, the explanation of the discovery that the withdrawal of absorber and adjuster rods is out of sequence contains a hypertext sensitive reference called "OPP 6.08". The topic for OPP 6.08 is defined as
**OPP 6.08**

**Sequence for Withdrawal of Absorber and Adjuster Rods.**

Rods shall normally be inserted and withdrawn in a sequence in accordance with the design intent.

This provides the user with an opportunity to see the exact description of the operational restriction if required.

**Model Scenarios**

The demonstration version of ARIES is not a complete representation of AECB regulations or AECB regulatory decision-making. Rather, it recognizes only two of the many possible decision scenarios. The two scenarios represent different operating conditions where the Project Officer is required to assess plant safety. The scenarios are described by the systems which are affected and the events which impinge on these systems.

**Scenario 1: Reactor Regulating System.**

A review of the Shift Log reveals that an adjustor rod is stuck halfway out of the core. The Project Officer must decide:

--- whether this fault has any safety significance,
--- if yes, then why is it significant, and
--- what is the limiting safety issue.

**Scenario 2: HTS Loop Isolation Valve Logic.**

A loop isolation valve is failed open. The Project Officer must decide:

--- whether the loop isolation system is impaired,
--- what actions are required by the operating staff,
--- what is the safety significance.

The demonstration version of ARIES includes routines to analyze these two events. The routines are designed to answer the questions posed above, but they are also capable of other decisions which follow from ARIES "knowledge base" about the two systems.

**Scenario 1.** ARIES analysis of the RRS fact base provides an assessment of shutdown depth. Shutdown depth is a measure of reactivity that is based on the current state (i.e., power level, fuel status and effective FP days). It also depends on the configuration of mechanical reactivity devices, including adjusters, MCAs and LZCs.

The user is presented with the screen shown in Figure 2. A mouse is used to select the device or configuration to be specified with input data. Once a specific device is selected the user is prompted to enter information about it. For example, after selecting one of the adjuster rods (Figure 3) the user is prompted to enter the percent withdrawn for the adjuster. A screen displaying the current status of different devices, as entered by the user or inferred by ARIES, can be shown at any time (see Figures 4 and 5).

![Diagram of Reactor Regulating System](image)

**Figure 2. RRS Graphics Screen. Reactor Status, HTS and Moderator System**
Figure 3. RRS Graphics Screen. Adjusters, MCAs and LZC Units.

Figure 4. RRS Display Screen. Adjusters, MCAs and LZC Units.
Reactor Regulating System: Equipment Status: Screen 2

<table>
<thead>
<tr>
<th>SDS1</th>
<th>SDS1</th>
<th>Reactor Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>out</td>
<td>out</td>
<td>Power Level</td>
</tr>
<tr>
<td>SA_1</td>
<td>SA_19</td>
<td>100 per cent</td>
</tr>
<tr>
<td>SA_2</td>
<td>SA_20</td>
<td>Fuel Status</td>
</tr>
<tr>
<td>SA_3</td>
<td>SA_21</td>
<td>equil</td>
</tr>
<tr>
<td>SA_4</td>
<td>SA_22</td>
<td>Eff FP Days</td>
</tr>
<tr>
<td>SA_5</td>
<td>SA_23</td>
<td>5 days</td>
</tr>
<tr>
<td>SA_6</td>
<td>SA_24</td>
<td>reactivity</td>
</tr>
<tr>
<td>SA_7</td>
<td>SA_25</td>
<td>-35 mk</td>
</tr>
<tr>
<td>SA_8</td>
<td>SA_26</td>
<td>net_reactivity</td>
</tr>
<tr>
<td>SA_9</td>
<td>SA_27</td>
<td>0 mk</td>
</tr>
<tr>
<td>SA_10</td>
<td>SA_28</td>
<td>Moderator Status</td>
</tr>
<tr>
<td>SA_12</td>
<td>SA_29</td>
<td>Boron Conc</td>
</tr>
<tr>
<td>SA_13</td>
<td>SA_30</td>
<td>0 ppm</td>
</tr>
<tr>
<td>SA_14</td>
<td>SA_31</td>
<td>D20 Purity</td>
</tr>
<tr>
<td>SA_15</td>
<td>SA_32</td>
<td>98 per cent</td>
</tr>
<tr>
<td>SA_16</td>
<td>SA_33</td>
<td>Moderator pH</td>
</tr>
<tr>
<td>SA_17</td>
<td>SA_34</td>
<td>7.8 pH</td>
</tr>
<tr>
<td>SA_18</td>
<td>SA_35</td>
<td>HTS System</td>
</tr>
<tr>
<td>out</td>
<td>out</td>
<td>Temperature</td>
</tr>
<tr>
<td>SA_19</td>
<td>SA_36</td>
<td>&gt;= 100 degrees C</td>
</tr>
</tbody>
</table>

Color key: user inferred default

Press any key to continue...

Figure 5. RRS Display Screen. SDS1, SDS2 and Reactor Status.

The ARIES knowledge base includes a set of rules which, for each reactor state, specifies a reactivity amount. For example, ARIES rules return different values for the following test data:

1. Equilibrium Fuel, Power Level is > 60% FP, Coolant Temperature > 100 degrees (Hot) and Effective FP Days > 5.
2. Equilibrium Fuel, 0 - 100% FP, Hot, Effective FP Days < 5.
3. Equilibrium Fuel, 0% FP, Coolant Temperature is < 100 degrees (Cold) and any Effective FP Days.
4. Fresh Fuel, 0% FP, Cold and any Effective FP Days.

Another set of rules returns a reactivity amount for each configuration of each reactivity device.

Using these values, ARIES estimates the reactor's shutdown depth, then determines whether this depth is sufficient for continued operation.

RRS analysis also tests the availability of shutdown systems SDS1 and SDS2. Specifically, it recognizes when SDS1 and SDS2 configurations are inconsistent with Operating Policies and Principles (OPP).

5. The number of available shutoff rods (SDS1) is less than 26.
6. The number of available poison tanks (SDS2) is less than 5.

RRS rules also check for other contradictions between RRS facts and the OPP. That is, ARIES reports an action for the following data:

7. The concentration of moderator Boron exceeds 10 ppm.
8. Banks of adjusters are withdrawn out of sequence.
9. For a bank of adjusters, a rod is stuck in.

Scenario 2. ARIES knowledge base for Loop Isolation analysis tests the availability of loop isolation valves. Valve availability depends on the status of the paired valves and other logic devices (e.g., instrument loop, 2/3 logic, test handswitches and valve control handswitches). Since ARIES understands the connections among these devices, there are numerous combinations of data that are inconsistent with the OPP requirement.

The primary input screen is shown in Figure 6. A mouse is used to select a component for input to alter its status. As for the RRS system, a display screen can be called up at any time to show the current status of the HT loop isolation system.

ARIES' reliability was tested using the following sets of inconsistent data:

1. Instrument loops P201K and P201M are failed.
2. Both ODD and EVEN 2/3 logic devices are failed or unavailable.
3. The EVEN 2/3 logic device is failed or unavailable, and test handswitch 63432-HS115K is set to 'test'.
4. The EVEN 2/3 logic device is failed or unavailable, and valve control handswitch 63331-HS22 is set to 'stop'.

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HTS Loop Isolation System: Equipment Status

<table>
<thead>
<tr>
<th>Inst Loops</th>
<th>Status</th>
<th>Position</th>
<th>Logic</th>
<th>Status</th>
<th>Position</th>
</tr>
</thead>
<tbody>
<tr>
<td>P201K</td>
<td>avail</td>
<td>n/a</td>
<td>odd</td>
<td>avail</td>
<td>n/a</td>
</tr>
<tr>
<td>P202K</td>
<td>avail</td>
<td>n/a</td>
<td>even</td>
<td>avail</td>
<td>n/a</td>
</tr>
<tr>
<td>P201L</td>
<td>avail</td>
<td>n/a</td>
<td>Test HS</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>P202L</td>
<td>avail</td>
<td>n/a</td>
<td>63432_HS115K</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>P201M</td>
<td>avail</td>
<td>n/a</td>
<td>63432_HS115M</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>P202M</td>
<td>avail</td>
<td>n/a</td>
<td>Valve HS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Valves</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3331_MV22</td>
<td>avail</td>
<td>open</td>
<td>63331_HS22</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>3332_MV2</td>
<td>avail</td>
<td>open</td>
<td>63332_HS2</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>3335_MV1</td>
<td>avail</td>
<td>open</td>
<td>63335_HS1</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>3335_MV3</td>
<td>avail</td>
<td>open</td>
<td>63335_HS3</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>3331_MV13</td>
<td>avail</td>
<td>open</td>
<td>63332_HS13</td>
<td>avail</td>
<td>auto</td>
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<td>63335_HS2</td>
<td>avail</td>
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<tr>
<td>3335_MV2</td>
<td>avail</td>
<td>open</td>
<td>63335_HS4</td>
<td>avail</td>
<td>auto</td>
</tr>
<tr>
<td>3335_MV4</td>
<td>avail</td>
<td>open</td>
<td>HTSystem</td>
<td>D20 Purity</td>
<td>99.6 per cent</td>
</tr>
</tbody>
</table>

Color key: user inferred default

Press any key to continue...

Figure 6. Loop Isolation Graphics Screen.

Figure 7. Loop Isolation Screen.
ARIES was also tested using the following consistent data:

5. Isolation valve 3311-MV13 is failed and closed. valve 3311-MV22 is failed and open, and valve control handswitch 63311-HS13 is set to 'open'.
6. Isolation valve 3311-MV22 is failed and open, and test handswitch 63432-HS115M is set to 'test'.
7. One of the above, and HTS coolant temperature is less than 100 degrees C.

ARIES was also tested using the following consistent data:

8. Instrument loop P201L is failed and test handswitch 63432-HS115K is set to 'test'.

USING ARIES

ARIES is designed to provide decision support to a Project Officer who is monitoring compliance with the conditions of a Reactor Operating License.

ARIES is consulted as follows:

1) Update the current operating state of the reactor and configuration of equipment and components.
2) Query ARIES as to the safety significance of the proposed operating state and configuration.
3) ARIES returns an assessment of the proposed operating state as being within or outside the licensed operating envelope.
4) ARIES will advise the Project Officer of constraints that must be placed on operation in the proposed configuration.

In addition to the knowledge base of facts and rules, ARIES uses hypertext extensively as an explanation facility. Hypertext is used in two ways:

1) to make the link between interrelated licensing requirements/rules.
2) to elaborate the significance of specific rules or conditions.

ARIES can also be used by the Safety Analyst who is concerned with the evaluation of the margin for shutdown depth. The knowledge base brings together in a coherent fashion the many inter-related aspects of reactor regulation that apply to a specific safety issue. It also is a powerful way to access the requirements and regulations that are given in regulatory guides, memos, letters and other documents associated with rule making.

DISCUSSION AND CONCLUSIONS

In the technical review function both the knowledge of generic issues, together with the rules of procedure and criteria, are not easily extracted from the extensive process of consideration and judgement which is continually applied to safety assessments. In the more operational fields of licensing, where both rules and knowledge are more firmly established, their transcription into and their use in an Expert System is relatively straightforward.

ARIES demonstrates that an expert system could be used by a Project Officer as a powerful tool to support the more rule-based compliance functions. It is feasible to implement such a system on a desk top personal computer so that it can be accessed directly at a site office.

Both the complexity of many regulations and their interpretation make a hypertext facility an essential part of any system.

As is common with most companies and organizations, the AECB's relevant corporate knowledge is mostly in the form of correspondence with its licensees. This has a tendency to disperse information rather than collect and entrench it. A knowledge base which could link all similar or related issues in the ongoing correspondence and at the same time permit it to be continually summarized into a "generic issue" file would appear to be the solution to this problem. These files would be subsequently accessed by the reviewer to provide a means of judging and considering acceptance of the licensee's proposals.

While KnowledgePro was adequate for the ARIES prototype, some difficulties were encountered with this expert system shell. First, display of the status screens (see Figures 5, 6 and 8) took far too long (> 30 seconds) when performed directly from KnowledgePro. This necessitated writing separate Pascal programs to read the current status data from a file and to display it appropriately. Secondly, the topic structure of KnowledgePro does not provide the frame-based automated inheritance mechanism that some more powerful expert system shells do. Expanding ARIES's knowledge base beyond its current size would be much easier and more consistent with such a facility.

The ARIES prototype demonstrates the complexity of coding required for development of an expert system. There are 60 pages of source code for the two scenarios coded into ARIES. It consists of 117 topics for rules and procedures, 49 topics for hypertext and prompt windows, and 15 lists representing groups of objects.

Given the complexity of the coding for ARIES, working expert systems should be developed for specific tasks. For example, an expert system could be developed from the ARIES model to monitor the availability of the special safety systems. This, in turn, could be developed as modules for each of the four systems.

The work done for ARIES has shown the potential for applying expert systems as a tool in the regulatory process in cases where there is a well defined knowledge base of rules and regulations. In its current form the ARIES prototype does not have sufficient knowledge to be used by the AECB staff. It does provide a basis upon which a working system could be developed.

ACKNOWLEDGEMENT

C.T. Downie (Safety Evaluation Division, AECB) initiated and was instrumental in the development of this project which was funded under the R and D Program of the Atomic Energy Control Board.
CONCEPTUAL PROBABILISTIC SAFETY ASSESSMENT
FOR CANDU 3

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ABSTRACT

This paper discusses probabilistic safety assessment (PSA) input to the CANDU 3 standard product design. The most recent effort in this area has been the "Conceptual Probabilistic Safety Assessment (CPSA)" for CANDU 3. Explanation is given of the initiating events studied in the CPSA, the methodology used, the acceptance criteria, and a summary of the results.

INTRODUCTION

Atomic Energy of Canada Limited (AECL) is currently completing the detailed design of an advanced plant. This new design, the CANDU 3, will expand the market for CANDU nuclear generating stations by offering the proven economic advantages of CANDU plants in a smaller size (450 MW net).

The ability to deliver the generating cost of current larger nuclear units in a smaller size will require the use of novel design, construction, manufacturing, licensing and management techniques. One of the most important goals is to minimize the possibilities for design change and back fits during the construction phase of the project. Achieving this goal means that the design work will need to be almost complete before construction starts and that the licensing authorities have reviewed the design and siting aspects of the project and are satisfied that no further major design changes will be required during construction.

Thus, there is a need for a thorough assessment of the safety aspects of the plant before construction starts. Safety assessments in Canada have for many years included the use of Probabilistic Safety Assessment (PSA) techniques. A logical extension of a PSA prior to construction is to incorporate PSA techniques into the design process, particularly in the concept definition phase, when these techniques can be used to define the minimum required component redundancies and other safety-related system design requirements.

Specifically, the objectives of the CPSA were:

a. To provide safety design assistance in the early stages of the project to ensure that the conceptual design includes adequate redundancy.
b. To establish reliability targets for safety related systems.
c. To identify accident sequences that could be large contributors to public risk.
d. To provide input to the environmental qualification program and control centre/emergency operating procedure design.

In this paper, we will discuss the methodology, initiating events considered, the acceptance criteria, and results of the CANDU 3 CPSA.

ACCETANCE CRITERIA

All PSA work typically involves postulating certain initiating events, and examining the possible scenarios which could ensue. Full scope PSAs present results which estimate:

- the frequencies of releases to the public; and
- the radiological consequences which could result.

A step-curve acceptance criterion (frequency/consequence) for such full scope PSA work has been proposed for the CANDU 3 (Reference 1). However, the CPSA is not a full scope PSA. It was conducted during the early stages of the CANDU 3 program. We have not had the benefit of consequence analysis specific to the CANDU 3 that has been extended to the evaluation of public dose. In addition, the CPSA does not include containment performance probabilities, so it is somewhat difficult to estimate the frequencies of releases outside containment. In place of the frequency/consequence step curve, we adopted the following design objectives:

1. All individual event sequences leading to a reactor core with no heat sink shall be less than $1 \times 10^{-6}$ events per year predicted average frequency.
2. All individual event sequences leading to the moderator having to act as a heat sink with the fuel channel integrity likely to remain intact shall be less than $1 \times 10^{-6}$ events per year predicted average frequency.
3. For individual event sequences where the moderator acts as a heat sink and there is some uncertainty with regard to fuel channel integrity, the predicted average frequency shall also be less than $1 \times 10^{-6}$ events per year.
For the purposes of the CPSA, we assume that for events where the heat transport system pressure is less than 5.5 MPa, the fuel channel integrity will be maintained when the moderator is required to act as a heat sink. Analysis is currently ongoing in this area and if this assumption is invalidated by the analysis, then the assumptions will be changed appropriately for the next stage of PSA work.

Based on past work (e.g. CANDU 6 analysis) other events which would lead to radioactivity release include: fuel handling failures, loss of inventory and failures in the moderator system. A check of CANDU 6 analysis in the Safety Report shows that given the expected frequency of these events, there is no difficulty in meeting the frequency/consequence criterion mentioned in Reference 1. The same conclusion is true when examining these events combined with containment impairments.

EVENTS ANALYZED

The list of CPSA initiating events is given in Table 1. Efforts were concentrated on initiating events which, based on past experience, have the potential to cause a loss of all heat sinks or loss of all heat sinks except for the moderator. For some typically postulated events, there was insufficient design information to do a meaningful assessment, and so these events were not addressed in this phase of PSA work. Note that external events (e.g., earthquakes, tornadoes, fires, and flooding from external sources) are not considered but are left for other studies.

METHODOLOGY

The motivation of the CPSA was to some extent "opposite" to the motivation of other PSA studies. Most PSA studies represent a detailed assessment of existing systems, the results of which can be used as a judgement of plant safety. The CPSA, on the other hand, starts with requirements for plant safety and develops a series of individual system reliability targets which, if met, will combine to meet the original overall plant safety targets.

<table>
<thead>
<tr>
<th>TABLE 1 CPSA INITIATING EVENTS</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Loss of Class IV Power</td>
<td>12. Spurious Opening of a UTS Liquid Relief Valve</td>
</tr>
<tr>
<td>2. Failure of Service Water Supplies</td>
<td>13. Spurious Opening of a Pressurizer Relief Valve</td>
</tr>
<tr>
<td>4. Steam Line Break in the Turbine Building</td>
<td>15. End Fitting Failure</td>
</tr>
<tr>
<td>5. Feedwater/Condensate Failures in the Turbine Building</td>
<td></td>
</tr>
<tr>
<td>6. Feedwater Break Downstream of the Steam Generator's Check Valve</td>
<td></td>
</tr>
<tr>
<td>7. Failure of Boiler Pressure Control</td>
<td></td>
</tr>
<tr>
<td>8. Pressure Tube Rupture</td>
<td></td>
</tr>
<tr>
<td>9. Pressure Tube Leak into the Annulus Gas System</td>
<td></td>
</tr>
<tr>
<td>10. Large Loss of Coolant</td>
<td></td>
</tr>
<tr>
<td>11. Heat Transport System Leak</td>
<td></td>
</tr>
<tr>
<td>12. Spurious Opening of a HTS Liquid Relief Valve</td>
<td></td>
</tr>
<tr>
<td>13. Spurious Opening of a Pressurizer Relief Valve</td>
<td></td>
</tr>
<tr>
<td>14. Feeder Break</td>
<td></td>
</tr>
<tr>
<td>15. End Fitting Failure</td>
<td></td>
</tr>
</tbody>
</table>

Obvious difficulties present themselves, when one attempts to assess complex hardware which is not built or completely designed. Still, the concept of establishing reliability targets for safety related systems, before the systems are completely designed, is seen as a very worthwhile objective. For this reason, the CPSA was undertaken using methods which matched the constraints.

One prominent example of how these constraints have impacted on the methods, is fault tree analysis. The major effort behind most PSA studies is the assessment of system reliabilities using fault tree methods. In the CPSA, fault trees are not extensively employed. This is because system reliabilities are for the most part established as requirements, derived from the event tree analysis. It is hoped for future work, to use state-of-the-art fault tree methods to testing the compliance of the completed design with the CPSA derived requirements.

Most of the CPSA work was accomplished by means of event tree analysis. The most significant aspect of the event tree construction process is the decision of which mitigating or support systems to model, and their position within the trees. In an effort to account for system interdependencies as much as possible, the support systems (e.g., electrical power, service water, etc.) required by front line mitigating functions are all modelled as separate event tree branch points. Similarly, to account for interdependencies between different types of operator actions, all actions are modelled as one composite event tree branch point.

Consideration of environmental effects (e.g., steam) was a significant factor in the decision of which systems to credit in the event tree construction. If, for example, a steam environment exists in the area where a system is not steam qualified, then no credit is taken for that system.

In quantitative terms, the event tree ties together the reliability of different systems, so that they can be judged within the context of overall plant safety. An IBM-PC based code, called "ETA" (Reference 2) was used to draw the trees and carry out simple multiplication tasks.

The number at the end of each sequence path represents the frequency of the plant end state. Each sequence end point frequency is arrived at in the CPSA by multiplying the initiating event frequency by the probabilities along the sequence path. Mathematically, this multiplication process is valid only if all sequences along the sequence path are completely independent. The simple multiplication process was used in the CPSA because the lack of detailed design information made alternative methods (i.e., event tree merging) not possible. While it is unlikely that no dependencies exist, event trees were constructed to represent the known cross links between systems as much as possible. Furthermore, many CANDU systems, and in particular the four special safety systems, are required by licensing and design standards, to be independent from each other and from the initiating failures which they are designed to mitigate. This will be confirmed once the design allows the development of detailed fault trees.

To a large extent, the frequencies of initiating events have been estimated by extrapolating the results of CANDU 6 assessments. For pipe breaks, the CANDU 3 piping lengths are estimated and then expressed in terms of "diameter lengths". Failure rates per unit length, which were derived in past studies, are applied here in the CPSA for each "diameter of length". In cases where CANDU 6 initiating event frequencies were not available from fault tree analysis or operations experience, the CANDU 3 values were extrapolated after considering any major design differences.

The CANDU 3 will have four "special safety systems". These systems are shutdown system one (SDS-1), shutdown system two (SDS-2), emergency core cooling (ECC), and containment. Each of these systems is required by AECL licensing criteria, to meet a target dormant unreliability of 1E-3. The CPSA used this 1E-3 value for these systems, in the event tree analysis.

The success/failure probability targets of systems, other than special safety systems, are arrived at by a variety of methods. The CPSA gives a brief description of each system, and shows the reliability target and how it was arrived at. In some cases, some
coarse fault tree analysis was employed. Here, an IBM-PC based package of codes called "CAFTA" (Reference 3) was used to draw and analyze the fault trees. In other cases, particularly for simple systems, the targets were estimated from engineering judgement and operations data from Ontario Hydro.

Pre-accident human errors were not specifically modelled, since the CPSA did not incorporate detailed judgement and operations data from Ontario Hydro. Post-accident human errors were modelled in the event tree for each initiating event. Probabilities of error were time-based, as follows:

- 0-15 min. available, probability of failure = 1
- 15-30 min. available, probability of failure = 10^-1
- 30-60 min. available, probability of failure = 10^-2
- >60 min. available, probability of failure = 10^-3

As mentioned earlier, each sequence path in the event tree was limited to only one operator branch point. This practice has the effect of assuming total dependence between different actions, while at the same time taking no credit for longer term recovery actions.

EVENT TREE ENDPOINTS

The public dose consequences of each event sequence endpoint depends on the heat sink that is cooling the fuel, the timing of the change from one mode of cooling to another and the heat transport pressure at which the change in cooling mode occurs. In examining the potential combinations of these parameters, we developed a list which describes all the potential event tree endpoints for this study. This list is given and the potential endpoints described in Table 2.

One should note that event trees have been terminated once the event frequency becomes less than 1E-9 events per year. This is well below the cutoff frequency used in the frequency/consequence step curve mentioned in Reference 1. This approach was used to make sure that significant contributors just below the Reference 1 cutoff (1E-6 events per year) were not missed and to allow easier investigation of design alternatives which could imply the deletion of systems from a particular event tree.

Table 2

<table>
<thead>
<tr>
<th>CATEGORIES OF PLANT END STATES</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. FF indicates a full heat transport system with forced circulation using the heat transport pumps. Heat is transferred to one or two steam generators (FF indicates Forced flow and Full inventory)</td>
</tr>
<tr>
<td>2. TF indicates a full heat transport system with flow by thermosyphoning forces only. Both steam generators are being maintained full by main or auxiliary feedwater pumps. (Thermosyphoning Flow with Full inventory)</td>
</tr>
<tr>
<td>3. TP Thermosyphoning with Partial inventory</td>
</tr>
<tr>
<td>4. S indicates a heat transport system full to at least header level with forced flow using the shutdown cooling pumps and cooling by the shutdown cooler and Group 1 service water. (Shutdown cooling is the heat sink)</td>
</tr>
<tr>
<td>5. FS indicates that reactor shutdown by the regulating system and shutdown system has failed to occur when required. Consequences could include a mismatch between power production and the heat sink resulting in severe fuel overheating and core disassembly (Failure to Shutdown)</td>
</tr>
<tr>
<td>6. E indicates a loss of coolant from the heat transport system with successful makeup from D_2O storage tank, demineralized water or the ECC system. Heat is removed either by the ECC heat exchanger for large breaks or the steam generators for small breaks (ECCS or D_2O feed supply makeup)</td>
</tr>
<tr>
<td>7. MHL Indicates that a loss of coolant has occurred and inventory makeup has failed. For large breaks, the heat transport pressure is low because of the break. For small breaks, crash cooling of the steam generators combined with continued feedwater maintains the heat transport system pressure at a low value. Under these circumstances, when moderator cooling is available, heat is transferred from the fuel to the moderator and fuel melting is prevented and core geometry is maintained. For this class of events the timing of the loss of inventory makeup is shortly after the event so that the heat transfer to the moderator starts within the first hour (&quot;MHL&quot; indicates that the moderator is the heat sink after an Hour with Low heat transport pressure)</td>
</tr>
<tr>
<td>8. MDL indicates the same situation as in MHL except that inventory makeup fails in the long term (beyond one day). (Moderator is the heat sink beyond one Day with the heat transport system at Low pressure)</td>
</tr>
<tr>
<td>9. MMM indicates that there has been a small break loss of coolant with a failure of heat transport cooldown using steam generator crash cool. (The Moderator in the heat sink within Minutes with the heat transport pressure at Medium value (about 5 MPa). Under these circumstances makeup by the ECC system becomes ineffective because the accumulator stops once the accumulator has depressurized to about 5 MPa). If continued feedwater is available, the heat transport pressure will remain at about 5 MPa. With continued moderator cooling, the decay heat will be transferred to the moderator</td>
</tr>
<tr>
<td>10. AHH all feedwater to the steam generators is lost and shutdown cooling fails or is not initiated by the operator. Under these circumstances, Loss LOCA or small LOCA situations, the heat transport system will pressurize to the liquid relief valve setting once the steam generator inventory is boiled away. The ECCS and moderator system are available but the effectiveness of these systems in mitigating the event consequences is uncertain and needs to be assessed. (&quot;A&quot; indicates the ECCS and the moderator are available. The heat sink is no longer effective after about one Hour and the system pressure in High)</td>
</tr>
<tr>
<td>11. ADH indicates the same situation as in AHH except that the loss of steam generator is in the long term after one Day)</td>
</tr>
</tbody>
</table>

CNS 10th ANNUAL CONFERENCE, 1989, 14-29
12. BHH Indicates that same situation as for AHH except that the moderator system is also impaired in terms of flow or cooling. B indicates that only the ECC system is available to mitigate the event. Under these circumstances channel failures will occur. Once some channels fail, the ECC system will fire as the heat transport system depressurizes. The number of channel failures and their consequences has not yet been assessed. As for AHH, the loss of steam generator inventory occurs in about one hour and system pressure becomes high after the loss of cooling.

13. BDH indicates the same situation as EHH except that the loss of steam generator heat sink occur in the long term after a day.

14. BMM indicates the same situation as MMM except that the moderator is not cooled. Again under these circumstances, some channels will fail and this will lead to heat transport system depressurization. The decrease in system pressure will permit ECC flow but the number of channel failures and their consequences has not been assessed yet.

15. NHJ No heat sink available at about one hour with system at high pressure.

16. NHL same as NHJ but system is at low pressure.

17. NDL same as NHL but heat sink is lost in the long term after one day.

18. NDF Not Developed Further because of low frequency.

RESULTS

Here, the highlights of the results are presented in terms of the derived acceptance criteria mentioned earlier.

1. All individual event sequences leading to a reactor core with no heat sink shall be less than 1E^-6 events per year frequency. If we consider the end states BHH, BDH, BMM, NHJ, NHL and NDL, the two highest frequency endpoints are:
   a) Feeder Break
      and Group 1 service water systems are unavailable
      and no Group 2 service water flow.
   b) End Fitting Failure
      and Group 1 service water systems are unavailable
      and no Group 2 service water flow.
   Each of these two sequences are predicted to occur at a frequency of 3.53E^-7 events per year. Therefore, the first criterion is met.

2. All individual event sequences leading to the moderator as a heat sink at 5.5 MPa or lower heat transport pressure shall be less than 1E^-4 events per year frequency. If we examine the end states AHH and ADH, the largest contribution is:
   a) Complete Loss of Class IV Electrical Power
      and Failure of all back-up feedwater to the steam generators
      and Failure of shutdown cooling system.
   The sequence frequency is predicted to be 6.64E^-7 events per year so the criterion is met.

CONCLUSIONS AND FUTURE WORK

No CPSA sequence endpoint had a frequency which exceeded the three derived acceptance criteria. This gives some assurance that the reliability targets assigned for individual systems were appropriate. It remains as future fault tree work, to show that each system can attain the reliability targets assigned.

While conducting the CPSA work, items of potential concern were fed back to designers and problems were rectified with minimal perturbation in the design process. Thus, the "on-line" PSA program has already proven its usefulness to the CANDU 3 project in minimizing the impact of necessary changes, through its "early warning" system.

Specifically with regard to control centre design input, we have already produced a preliminary list of initiating events and corresponding corrective operator actions. This list was made possible as a direct spin-off of our CPSA event tree work. The information will be used by control centre designers to prioritize and configure alarms and controls.

The next step in the CANDU 3 PSA program is to study more initiating events, and develop the systems fault tree analysis.

REFERENCES


SUMMARY

In 1988, AECL obtained government support for the design of the CANDU 3 over a three-year period starting 1988 April 1. The objective of this three-year program is to complete the detailed design to the greatest extent possible, excluding site specific design. This standard plant design is intended to satisfy domestic as well as export needs, hence, the design envelope has been chosen to accommodate the characteristics of a large number of sites around the world. Part of this objective is to ensure that the standard plant design is licensable. The licensing basis for the standard plant is "licensable in Canada".

Completion of the CANDU 3 standard plant design will position AECL to react quickly to marketing opportunities, both within Canada and abroad. Availability of a licensable product will provide confidence that specific projects can be completed on schedule and should thus help in obtaining project financing for both domestic and off-shore projects. Approval of the standard plant design by the Atomic Energy Control Board (AECB) is also necessary to achieve the 35 month construction schedule established for the CANDU-3.

The focus of the first year of the program was to fully document the conceptual design and the design requirements so that detailed design could proceed in years two and three with reduced risk of changes.

The licensing objectives for the first year were the establishment of a detailed licensing basis, which includes the specification of all licensing requirements and safety design requirements; and the preparation of a Conceptual Safety Report, which summarizes the results of design assist analysis for safety systems, as well as results of a preliminary Probabilistic Safety Assessment (PSA).

The AECB has been aware of the CANDU-3 plans and developments since the beginning of 1987 and has maintained a keen interest in this work. On the basis of a very limited review, AECB staff offered comments on certain aspects of the design. These comments were reflected in the current design and analysis work. The AECB is to assume a formal licensing role for the next stage, which is a detailed review of the conceptual design and of the proposed safety and licensing requirements.

The key licensing milestone at the end of the program is the standard plant design approval. During the third year, the standard plant PSA and Safety Report will be issued to the AECB for review and comment. On the basis of these documents and other detailed design documents, the AECB will identify the conditions, if any, for acceptance of the standard plant design. As proposed, a standard plant design approval will mean that if the design is not modified in a significant way, it can be referenced in a construction licence application for a specific site and only the site specific characteristics of the design will need to be discussed, at that time.

INTRODUCTION

Industry Developments and Trends

A resurgence of the nuclear industry in the 1990s appears to be likely. For this reason the industry world-wide is poising itself for the second phase of nuclear power commercialization. If the first phase of commercialization is any indication of future activity, the next few decades will provide many challenges and opportunities to the industry.

In preparation for the next wave of nuclear orders, suppliers of nuclear power plants have been updating the designs used for currently operating plants. The updating generally includes improvements in the following areas:

- safety
- cost
- schedule
- constructibility
- maintainability and operability

The concept of standard plant designs, while not new, is being revived with a vengeance. The reason is simple economics. For nuclear power to be competitive with coal fired plants at a coal price of $40/tonne, the construction time for a nuclear power plant must be reduced to three to four years - from start of construction to in-service. A pre-approved standard plant design is essential to meet such a schedule.

Developments in the U.S.

The plant standardization efforts in the U.S. are being co-ordinated by EPRI under the label of the Advanced Light Water Reactor Programme (ALWR). EPRI has produced a design requirement document that all
future plants must comply with. The intent is to get approval of the requirements by the Nuclear Regulatory Commission (NRC), while individual suppliers update their designs to bring them in line with the new requirements. The second step is the submission of a standard plant safety report by each supplier to the NRC for certification of the design.

The recent U.S. NRC action approving a one-step licensing process gives the U.S. nuclear industry a much needed boost. One-step licensing means that a combined construction licence and conditional operating licence will be issued by the NRC prior to the start of construction. This action also means that the NRC is prepared to license a standard plant design before construction start; and that once licensed, the licence will be valid for a period of ten years.

The NRC move is expected to bring predictability to the U.S. licensing process. This is an essential step to rebuild the confidence of the utilities in the nuclear industry - a prerequisite to future orders for nuclear power plants.

The Japanese have already used a one-step licensing process which allows the granting of a combined construction and operating licence prior to the start of construction. This process has allowed utilities to construct plants in much less time than four years. For example, Takahama units 3 and 4, owned by the Kansai Electric Power Company, have been built in 39 months from first concrete to fuel loading.

Developments in Canada

In Canada, the CANDU 3 is the latest and most advanced reactor design, capable of producing 450 MW of electrical power. The goal of AECL is to have a licensed standard plant design prior to the start of construction of the first plant.

The licensing process in Canada has evolved in parallel to the development of the nuclear power program. The basic three steps - site acceptance, construction approval and operating licence - have been applied to all commercial projects (Reference 1).

The discussions held between AECL and AECB on the licensing of a repeat plant at Point Lepreau represent the first steps taken to introduce the concept of one-step licensing in Canada under the label of up-front licensing (Reference 2, 3). The concept of "repeat" plant c: "standard" plant licensing was first discussed, between AECB and AECL in 1982. Detailed discussions commenced in April, 1983, and terminated in April, 1986, with the submission by AECL of a proposed Licensing Basis Document (Reference 4). The discussions were terminated earlier than anticipated when it became clear that the construction of a repeat plant at Point Lepreau was becoming less likely. Despite the early termination, a significant milestone was achieved - the concept of issuing a combined construction approval and conditional operating licence prior to the start of construction was established.

After the submission of the Licensing Basis Document, staff of the Atomic Energy Control Board completed their review of documents submitted in support of a licensing basis agreement and submitted a staff report to their Board (Reference 5). The report contained the recommendation that, "the up-front licensing approach has merit," and that "the Board confirm its endorsement of the acceptability of this licensing approach." In March, 1987, the Atomic Energy Control Board accepted the up-front licensing approach (Reference 6).

This paper describes the approach and plan proposed to the Atomic Energy Control Board for the licensing of the CANDU 3.

STANDARD PLANT DESIGN PROGRAM

In 1988, AECL obtained government support for the design of the CANDU 3 over a three-year period starting 1988 April 1. The objective of this three-year program is to complete the detailed design, to the greatest extent possible, excluding site specific design. This standard plant design is intended to satisfy domestic as well as export needs, hence, the design envelope has been chosen to accommodate the characteristics of a large number of sites around the world.

Design Principles

The following standard plant design principles and objectives were established at the inception of the design:

a. To enhance or improve traditional CANDU advantages including safety, low radiation exposure, high capacity factor, ease of maintenance, and low operating cost.
b. To reduce specific capital cost, construction schedule, and unit energy cost.
c. To standardize the plant design such that it is suitable for any reasonable site, worldwide, without significant changes to the design.
d. To accommodate division among the plant structures and systems to facilitate a variety of shared financing, contractual arrangements, or partners with one or more organizations, without significant design or documentation changes.
e. To employ state-of-the-art technologies, including design, construction, operation and project management technologies, consistent with construction in the 1990s.
f. From a component design viewpoint, to:
   1. maximize component life,
   2. provide easy replacement at end of component life; "easy" replacement means quick and simple - without complex tooling or an extended outage - thereby minimizing radiation exposure.
3. minimize component cost, and
4. minimize component installation time and cost.

From a maintenance/in-service inspection viewpoint, to:

1. achieve a minimum of two years of station operation between scheduled maintenance/in-service inspection outages, which will not exceed 21 days, and
2. accommodate major equipment replacement (fuel channels, steam generators, etc.), major systems modification or modernization (control, computers, etc.), or major equipment refurbishing (reblading turbine, etc.) in a major maintenance outage, lasting up to 90 days. Such a major maintenance outage is expected to be required no more frequently than every 15 years.

Design Approach

The design approach is to establish a comprehensive set of design requirements, including safety and licensing requirements, prior to the start of detailed design work. These requirements are developed during the evolution of the conceptual design to ensure that requirements identified during the conceptual design phase are included in the design requirement documents. The process is an iterative one and is terminated when the conceptual design is "frozen".

At concept design freeze, the requirements have already been well established and the key plant systems will have gone through a formal design review process. The formal design reviews include participation from utility staff and other experts outside of CANDU Operations. Feedback from independent AECB staff reviews will also be incorporated prior to concept design freeze. These third party reviews will ensure that the detailed design will proceed without major surprises and costly design changes.

In addition, New Brunswick Electric Power Commission staff, at Point Lepreau, have done a thorough and systematic review of the design, and support the basic design concepts.

Program Schedule

The standard plant design program stretches over a three-year period and is integrated with a project schedule that calls for the first unit to be in-service by the fall of 1995. On this basis, construction for the first unit would start in the fall of 1992.

Design Activities

The main design activities during the first year of the program are the establishment of design requirements and the development of design concepts. The focus in the second year is the completion of conceptual design documents, and the start of detailed design. The third year is devoted completely to detailed design and the production of documents for the detailed design.

It is anticipated that by the end of the third year about 90% of all non-site specific engineering will have been completed. This would amount to about 70% of the total project engineering.

Licensing Activities

The main licensing activity during the first year of the program was the compiling and submission to the AECB of the Conceptual Safety Report. The standard plant Conceptual Safety Report includes the following key documents:

- Standard Plant Licensing Basis
- Regulatory Compliance Document
- Systematic Review of Plant Design
- Safety Design Guides

The key licensing activities during the second year of the program are:

- preparation of assessment reports on currently outstanding safety issues
- preparation of assessment reports on CANDU 3 specific features (eg: novel design features such as directed ECCS)
- interaction with AECB staff to ensure discussion and resolution of conceptual design issues

The major licensing activities during the third year will be the production of the Standard Plant Safety Report and interaction with AECB staff to discuss and resolve remaining issues.

Licensing Plan

Standard Plant Licensing

Up-front licensing in the form of standard plant design approval is an essential step towards making construction costs of nuclear plants competitive with those of coal-fired stations. This benefit can be fully realized, however, only by exercising a great deal of self-discipline on the part of all the players involved in the process. A standard plant design approval requires precise commitments by both the designer and the licensing authority.

The commitment by the designer is to generate a high-quality product that can maintain its commercial validity over a sufficiently long period of time, so as to warrant concentration of resources on the engineering effort for the generic design development. The commitment by the licensing authority is to review the product to ensure that it meets all safety and licensing requirements that make it suitable for being referenced in the licence application.

Paramount to meeting the above commitments is a mutual understanding between the designer and the licensing authority on the licensing ground rules to be applied to the design, and to stick to them as far as possible. Any exceptions to the ground rules...
should have a justifiable basis. Hence, inherent in the standard plant design approval process is a thrust to improve the quality of the design process, whereby all requirements are set and agreed upon up-front, and the design implementation then follows from a clear understanding of those requirements.

Milestones

The first major licensing milestone is AECB acceptance of the standard plant licensing basis. This must be accomplished well in advance of the conceptual design freeze, to ensure that the design will comply with all licensing requirements.

The second major licensing milestone is AECB acceptance of the standard plant conceptual design prior to design freeze to ensure that detailed design can proceed with a low risk of changes resulting from the licensing process.

The third key licensing milestone is AECB approval of the standard plant design at the end of the three-year program. Such an approval would be the equivalent of a "conditional construction licence." It would allow a prospective licensee to use the standard plant design, or to reference it, in a construction licence application, so that only site specific issues need to be discussed at the construction licence time.

A final standard plant design approval is the last major licensing milestone. Such an approval would be the equivalent of a conditional operating licence. This approval would be obtained "de facto" when a construction licence is granted for the first project. Hence, subsequent plants at the same site or different sites could use the final design approval for a combined construction licence/operating licence application, so that only site specific issues need to be discussed during the review process for each application.

The final design approval provides the prospective licensee the added assurance that the plant will be licensable subject to:
- demonstration that plant specific features, including site conditions, are safely bounded by those incorporated in, or used as input to, the standard design,
- demonstration of compliance with all licensing requirements concerning the conduct of plant operations,
- demonstration that construction of the plant is in accordance with the approved Construction QA program, and
- satisfactory reassessment of the design based on any new significant safety information that may be generated in the time period between the design approval and the application.

STANDARD PLANT DOCUMENTATION

A standard plant design approval represents a major task for both AECL and the AECB as this requires a large volume of documents to be produced and reviewed. Many of the documents required are those that traditionally have been submitted at the operating licence stage.

The major licensing documents proposed to be submitted for a standard plant design approval are listed in Table 1. These licensing documents would be supported by detailed design and analysis documents as listed in Table 2.

| TABLE 1: LICENSING DOCUMENTS PROPOSED TO BE SUBMITTED FOR A STANDARD PLANT DESIGN APPROVAL |
|---|---|
| 1) Standard Plant Licensing Basis |
| 2) Safety Design Guides |
| 3) Standard Plant Safety Report |
| 4) Standard Plant Probabilistic Safety Assessment |
| 5) Assessment of Common Cause Events |
| 6) Assessment of Low Probability Events |
| 7) Topical Reports |
| 8) Regulatory Compliance Document |
| 9) Overpressure Protection Report |
| 10) System Classification List |
| 11) Safety Analysis Data List |
| 12) Documentation for computer codes |
| 13) Register of Licensing Documentation |

| TABLE 2: DESIGN DOCUMENTS PROPOSED TO BE SUBMITTED FOR A STANDARD PLANT DESIGN APPROVAL |
|---|---|
| 1) Design Requirement documents for specified systems |
| 2) Design Guides |
| 3) Design Descriptions for specified systems |
| 5) CANDU Operations Engineering QA Manual |
| 6) CANDU Operations Procurement QA Manual |
| 7) Design specifications for specified components |
| 8) Decommissioning Plan |
| 9) Reliability Targets for specified systems |
| 10) Human Factors Plan |
| 11) Post-LOCA Radiation Management Study |
| 12) Environmental Qualification Program |
| 13) Radiation Exposure Control Program |
| 14) Periodic Inspection Program |
| 15) SDS No. 1 & 2 Trip Computer Software QA Plan |

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Design

To date, about 30% of the conceptual design documentation has been completed with good progress being made in the process systems (50% complete). Systems which are new or unique to the CANDU 3 are still being finalized. These include, for example, the Distributed Control System which uses data highway technology, the Plant Display System using exclusively CRTs, the Emergency Core Cooling System with directed injection capability and the fuel channel design to accommodate single-ended refuelling.

Safety Analysis

The Conceptual Safety Report was completed and issued to the AECB in March, 1989. This report contains preliminary assessments of safety systems performance as well as preliminary assessments of common cause events (e.g. earthquakes, tornadoes, fires, etc.). A Conceptual Probabilistic Safety Assessment was also completed and issued to the AECB along with the Conceptual Safety Report.

The following topical reports have also been completed and issued to the AECB:
- Plant Layout and Grouping Assessment
- Containment System Assessment
- Safety System Pressure Transmitter Testing
- Main Steam Line Isolation Capability
- Testability of Electrical Penetrations

Other reports are nearing completion.

AECB Review

Staff of the Atomic Energy Control Board have been informed of plans and developments on the standard plant design since the beginning of 1987 and have maintained a keen interest in this work. On the basis of a very limited review of preliminary documents, AECB staff have made comments on certain aspects of the design. These comments were useful and have been reflected in the current design.

With the submission of the Conceptual Safety Report and supporting documentation to the AECB, in March, 1989, the AECB has been asked to perform a formal design review and to indicate its acceptance of the approach proposed by AECL. To our knowledge, AECB staff have not yet started the formal review process. However, the AECB recognizes the benefit of carrying out this review and has initiated actions to make it possible.

EXPERIENCE TO DATE

As stated above, a standard plant design approval process demands a great deal of self-discipline on the part of all the players involved in the process.

A disciplined approach to designing and licensing a nuclear power plant may be the quality way and the only economical way, but it is not the natural or instinctive way for designers: designers want and need flexibility during the design process to produce the ultimate design. The application of this approach to the CANDU 3 is therefore necessitating a cultural change in the design team. Whereas requirements may have previously been considered flexible and subject to change during the course of the design, they are now defined as items that have to be complied with. Hence, the specification of requirements is now a more rigorous process that has to be done earlier in the design.

The need to specify requirements early has also necessitated changes in the organization structure of the design team. From the inception of the program, the safety team was integrated with the design team to ensure that a complete set of requirements would be specified and to resolve any conflicting requirements early. The integration of these teams has worked extremely well.

The use of PSA tools at the beginning of the program helped to identify a number of design changes early and, thus, eliminated the need for costly changes later in the design process.

CONCLUSION

The standard plant licensing approach proposed for the CANDU 3 is a natural evolution of the up-front licensing approach that was developed, in co-operation with the Atomic Energy Control Board, for the licensing of a repeat plant at Point Lepreau. Furthermore, the standard plant approach is consistent with the approaches used in Japan and those being adopted in the U.S.

Such an approach is workable, as has already been demonstrated in Japan, but requires strict discipline and co-operation on the part of the designer, the licensee and the regulator. This will be the challenge of the 1990s.
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RADIATION-INDUCED AIR/WATER PARTITIONING AND GAS PHASE SPECIATION OF IODINE FROM IRRADIATED IODINE SOLUTIONS

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ABSTRACT

The effect of gamma radiation on partitioning of iodine between air and cesium iodide solutions ranging in concentration from $10^{-3}$ to $10^{-7}$ M has been investigated using radiochemical methods. In this paper, the dependence of partition coefficient on the total dose delivered at dose-rates ranging from 0.03 to 20 kGy/h in CsI solutions of various concentrations ($10^{-3}$ to $10^{-7}$ M) and pH (5 to 9), with or without the presence of a reducing agent, hydrazine, is discussed. It is found that radiation reduces the partition coefficient, liquid to gas iodine concentration ratio, under all conditions with the results being more pronounced at lower pH and higher iodide concentrations. The dose-rate at which a given total dose is delivered is found to be an important factor in determining the partition coefficient. Irradiation of the solutions containing $10^{-3}$ M hydrazine resulted in higher partition coefficients, indicating that the presence of the reducing agent negated some of the effects of radiation.

1. INTRODUCTION

Accidents resulting in the release of significant amounts of radioactivity to the environment from nuclear power reactors are unlikely yet possible events. One such accident resulting in the release of fission products is the loss-of-coolant accident (LOCA) and loss of shutdown, occurring when a failure in the cooling system employed to transport heat from the reactor fuel is encountered. Uranium oxide fuel is a ceramic material which contains the majority of fission products in the fuel grain (grain inventory) as well as some in the open cavities referred to as gap inventory (1). If the reactor overheats, for example, to a loss of coolant some of the fission products could be released from the fuel matrix and be swept via the cooling system into the containment structure surrounding the core resulting in an aqueous pool containing fission products.

Of all fission products, radiiodine is especially important because of its high fission yield, long biological half-life, high volatility, and ease of ingestion and inhalation. Radiiodine released following an accident in the primary heat transport system is considered to be in the form of water soluble cesium iodide (2). The cesium iodide rapidly dissolves to form non-volatile $I^-(aq)$ ions. However, $I^-(aq)$ can be oxidised to form volatile species which through leakage or filtered venting inherent to containment design can become airborne. The extent to which radiiodine becomes airborne is described in terms of water/air volumetric partition coefficient ($H$), defined as the ratio of aqueous to airborne radiiodine concentration.

The oxidation of $I^-(aq)$ ion and subsequent reactions of the various oxidation products, such as molecular iodine ($I_2$) and hypoiodous acid (HOI), with each other and organic impurities present in the aqueous pools formed throughout the containment structure result in the production of a multitude of radiiodine species of different oxidation states. The situation is further complicated by the presence of radiation fields produced from the decay of various fission products. Therefore, in order to predict the behaviour of radiiodine following a reactor accident an understanding of the chemical nature of the aqueous pool, such as pH, composition, presence of redox agents, as well as the effects of variable radiation fields is required.

In this paper some of the findings of an ongoing investigation of the air/water partitioning and gas phase speciation of radiiodine in the presence of variable gamma radiation fields is discussed. It has been estimated that the maximum possible total iodine concentration that could be produced following any accident in a 600 MW CANDU reactor would be less than $10^{-2}$ M (3). The low concentrations of iodine solutions suggest the difficulty involved in making meaningful quantitative assessments of iodine behaviour. In order to overcome this problem, radio-analytical methods were used to determine the partition coefficient and gas phase speciation of CsI solutions ranging in concentration from $10^{-2}$ to $10^{-7}$ M at pH values of 5 and 9. The effect of gamma irradiation was studied by radiating solutions at dose-rates ranging from 0.03 to 20 kGy/h. Finally, the effect of hydrazine, a strong reducing agent, on
partitioning and gas phase speciation of irradiated radioiodine solutions was investigated.

2. EXPERIMENTAL METHOD

The behaviour of iodine was studied in Pyrex thiemeyer flasks containing air and Csl solutions of known concentration and pH. These flasks were equipped with inlet/outlet lines, each of which could be sealed with ground glass stop-cocks. The inlet line delivered air at the surface of the liquid while the outlet line withdrew air from the top of the flask. The flasks were constructed of three different total volumes of approximately 300, 180 and 60 mL.

The reagents used throughout the research were of high grade. The water used was purified by reverse osmosis with further purification through filtration and deionization using a Barnstead nano-pure system. This system, including a charcoal bed for removal of organic substances, claimed to be able to reduce the concentration of organics to less than 10 parts per billion. Cesium iodide solutions of concentration 10⁻⁵ and 10⁻⁶ M were prepared by successive dilution of 1-131 labelled stock solutions. The L-131 tracer was purchased from Amersham in 10 mCi allotments and was reported to contain less than 1 microgram of stable 1-127 per 20 mCi of L-131. The pH of Csl solutions was adjusted using only drops of dilute LiOH and/or H₂SO₄ solutions.

After the preparation of Csl solutions of known concentration and pH, each solution was shaken mechanically using a wrist-action shaker for at least 2 hours. The solutions were then irradiated using Co-60 gamma cells, capable of delivering 0.4 and 20 kGy/h dose-rates. In some experiments the dose-rate was reduced to 0.03 kGy/h using lead shielding. Control solutions were shaken for up to 10 days with no irradiation to elucidate the effects of irradiation and provide a basis for comparison. The partitioning and gas phase speciation were determined by pulling a known volume of the air within a flask, usually less than 70 percent, through the use of a set of species-selective adsorbents developed by Keller (4). The adsorbents used in order of contact with the airborne iodine were: cadmium iodide for retention of molecular iodine, p-iodophenol for retention of hypoiodous acid, and triethylenediamine (TEDA) impregnated charcoal for retention of any remaining species such as organic iodides. In addition, two pieces of teflon filter, having pore diameters of 2-5 micrometers were placed at the entrance of the sampler for monitoring the entrance of any liquid aerosols. The viability of this gas sampling apparatus is extensively discussed elsewhere (5). It suffices here to say that cadmium iodide is highly efficient in the retention of molecular iodine (>90%) and iodoform (>85%), iodoform is efficient in adsorption of hypoiodous acid (>90%) and high-molecular-weight organic iodides such as butyl iodide (>65%), while charcoal adsorbs any remaining low-molecular-weight organic iodides.

3. RESULTS AND DISCUSSION

3.1 Effect of Dose/Dose-Rate on Partition Coefficient

The primary interaction of radiation with aqueous Csl systems occurs with water. Radiation decomposes water into a mixture of radicals according to the reaction [6]:

\[
4.25 \text{H}_2\text{O} \rightarrow 2.7\text{e}^- + 2.8\text{H}^+ + 0.55\text{H}_2 + 0.45\text{H}_2\text{O}_2
\]

where the coefficients are G values representing average number of molecules formed (or destroyed) as a result of absorption of 100 eV of radiation energy.

The primary radicals formed by water radiolysis can react with themselves or the solutes in water to produce more radicals and stable molecules. The reactions between water radiolysis products and iodine species are very complex and involve many intermediates; however, the most important reactions are those which involve the oxidation of non-volatile I⁻ to volatile I₂ or HIO. Iodide oxidation predominantly occurs as a result of the reaction of iodide with the oxidising hydroxyl radical, OH:

\[
\text{I}^- + \text{OH} \rightarrow \text{I} + \text{OH}^-
\]

while atomic hydrogen (H) and solvated e⁻ (or its oxygenated form O₂⁻) act as the main reducing agents:

\[
\text{I}^- + \text{e}^- \rightarrow \text{I}^-
\]

\[
\text{H}^+ + \text{H} \rightarrow \text{H}_2
\]

\[
\text{H}_2\text{O}_2 \rightarrow \text{H}_2\text{O} + \text{O}_2
\]

Under acidic conditions solvated e⁻ or its oxygenated form O₂⁻ react with hydrogen ions to form HO₂, thus no longer being able to reduce oxidised iodide back to I⁻(aq):

\[
\text{e}^- + \text{O}_2 \rightarrow \text{O}_2^-
\]
\[ \text{O}_2^- + \text{H}^+ \rightarrow \text{H}_2^\text{O} \]

Furthermore, under acidic conditions atomic hydrogen reacts with hydrogen ions to produce an oxidising species \( \text{H}_2^+ \):

\[ \text{H} + \text{H}^+ \rightarrow \text{H}_2^+ \]

The net effect of the above reactions is to produce an oxidising environment under acidic conditions, while a reducing environment is favoured under neutral or basic conditions.

The graphs of partition coefficient vs dose (figures 1 and 2) for \( 10^{-7} \) M CsI show that the partition coefficient decreases with increasing dose until a final steady state value is reached. However, it can be seen that at both pH values of 5 and 9 the decrease in the partition coefficient is more significant at higher dose-rates. It should be noted that the partition coefficients of irradiated \( 10^{-5} \) M CsI solutions are at least 10 times (at pH=9) and 1000 times (at pH=5) less than those of the non-irradiated solutions (i.e. radiation greatly increases the volatility of the solutions).

In case of non-irradiated \( 10^{-7} \) M solutions of pH 5 and 9, the partition coefficient decreases to a value of less than \( 10^{-3} \) after an initial irradiation of the solution with less than 2 kGy of gamma energy and stays relatively constant thereafter. At the more acidic pH of 5 the decrease in the partition coefficient is more pronounced and partition coefficients as low as \( 10^{-3} \) are observed. It should be noted that as in the case of \( 10^{-5} \) M solutions partition coefficient decreases with an increase in the total dose delivered. However, the trend of higher dose-rates causing higher decreases is not observed. This to some extent can be attributed to the uncertainty in dealing with the very low concentrations of these solutions. In case of non-irradiated \( 10^{-7} \) M solutions of pH 9 shaken for up to 9 days (figure 5), the partition coefficient never dropped below 5x10^{-6}, while under similar conditions at pH 5 partition coefficient decreased to as low as \( 10^{-5} \) (i.e. in both cases radiation did result in production of more volatile species).

3.2) Effect of Concentration on the Partition Coefficient

In case of non-irradiated CsI solutions of \( 10^{-3} \) M and \( 10^{-5} \) M concentration, the partition coefficient decreases with time over a 9 day period, with the decrease being more pronounced at the lower concentration.
In other words, for non-irradiated solutions, low concentrations correspond to lower partition coefficients at both of the investigated pH values (Figure 5). This observation is consistent with the findings of other investigators (7,8).

For irradiated $10^{-7}$ M CsI solutions of pH=5, the dominant species in the gas phase is still $I_2$ (albeit at a lower concentration of $10^{-10}$ M). At pH=9 $I_2$ is dominant over HOI by a factor of 10 (a maximum $I_2$ concentration of $10^{-11}$ M). For both dose-rates of 0.4 kGy/h and 20 kGy/h, while for the lower dose-rate of 0.03 kGy/h both $I_2$ and HOI are present in approximately equal concentrations (ca. $5 \times 10^{-19}$ M), while organic iodides are initially present at 3-4 times lower concentrations and slowly build up.

It should be noted that at the lower concentration of $10^{-7}$ M CsI a lower concentration of $I_2$ is observed in the gas phase which translates into a higher partition coefficient.

3.3) Effect of Radiation on Gas Phase Speciation

For irradiated $10^{-5}$ M CsI solutions of pH=5 the only species observed in the gas phase is molecular iodine ($10^{-6}$ M). In case of non-irradiated $10^{-5}$ M CsI (pH=5) solutions organic iodide and HOI are the dominant species in the gas phase and their combined concentration reaches values of up to $6 \times 10^{-12}$ M after 2-3 days. At the more basic pH of 9 ($10^{-5}$ M CsI) and for dose-rates of 0.4 and 20 kGy/h the dominant species in the gas phase is still $I_2$ (up to $2 \times 10^{-15}$ M $I_2$); however, some HOI (almost 10 times less) is also observed. In case of the lower dose-rate of 0.03 kGy/h almost equal concentrations of $I_2$ and HOI are observed in the gas phase with little or no organic iodides present. It should be mentioned that in case of non-irradiated $10^{-5}$ M CsI solutions of pH=9, the gas phase concentration of organic iodides is greater than that of HOI which is in turn greater than $I_2$.

For irradiated $10^{-7}$ M CsI solutions of pH=5, the dominant species in the gas phase is still $I_2$ (albeit at a lower concentration of $10^{-10}$ M). At pH=9 $I_2$ is dominant over HOI by a factor of 10 (a maximum $I_2$ concentration of $10^{-11}$ M). For both dose-rates of 0.4 kGy/h and 20 kGy/h, while for the lower dose-rate of 0.03 kGy/h both $I_2$ and HOI are present in approximately equal concentrations (ca. $5 \times 10^{-19}$ M), while organic iodides are initially present at 3-4 times lower concentrations and slowly build up.

It should be noted that at the lower concentration of $10^{-7}$ M CsI a lower concentration of $I_2$ is observed in the gas phase which translates into a higher partition coefficient.

3.4) Effect of Hydrazine on Partition Coefficient

Hydrazine is a strong reducing agent which, if added to dousing sprays and the emergency core coolant, would be able to produce reducing conditions in the water pool after a LOCA. The main reactions of hydrazine involve the reduction of $I_2$ to $I^-$ (aq) and depletion of $O_2$ and $H_2O_2$ from the aqueous solution (3).

$$2I_2 + N_2H_4 \rightarrow 2I^- + 4H^+$$

$$2O_2 + N_2H_4 \rightarrow 2H_2O_2 + N_2$$

$$2H_2O_2 \rightarrow 4H_2O + N_2$$

The effect of radiation on partitioning of CsI solutions of pH=9 in the presence of $10^{-5}$ M hydrazine (20 ppm) was investigated. Figures 6 and 7 show some of the findings for dose-rates of 0.4 and 20 kGy/h. At both concentrations the decrease in the partition coefficient observed under irradiated conditions has been minimized and higher partition coefficients are observed. Similarly, the presence of hydrazine in non-irradiated control solutions resulted in higher partition coefficients (Figure 5).
4.) CONCLUSIONS

The partition coefficient of iodine between air and dilute cesium iodide solutions irradiated at various gamma dose-rates ranging from 0.03 to 20 kGy/h decreases with increasing total dose delivered to the solutions. In case of irradiated 10^{-2} M CsI solutions of pHS9 (most relevant to accident conditions) a conservative lower limit of 10^{4} can be placed on the partition coefficient, while this number becomes even more conservative at lower CsI concentrations. The presence of 10^{-4} M hydrazine in CsI solutions increases the partition coefficient even in the presence of the rather strong 20 kGy/h radiation field.

5.) ACKNOWLEDGEMENTS

The authors would like to gratefully acknowledge the financial support provided by the Department of Nuclear Materials Management at Ontario Hydro and URIF.

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Some Implications of Assumed Tellurium Behaviour in Containment in the Long Term Following a LOCA

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Paper not received.
FISSION PRODUCT GRAIN-BOUNDARY INVENTORY


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SUMMARY

During irradiation, the fission products generated inside the UO₂ matrix migrate to the grain boundaries by a process controlled by the irradiation temperature. In the unlikely event of an accident, those fission products which have reached the grain boundaries will be the first to be released from the bulk UO₂. They may be released by different mechanisms depending on the accident sequence. The inventory available on the grain boundaries thus is an important component of the source term and its release history.

Previous work has shown that oxidation in air at low temperatures causes separation of the UO₂ grains by a process of grain-boundary oxidation. On-line measurements of fission-product evolution during this process gave a direct determination of the amounts of Xe and I located at the grain boundaries.

Since there is a radial temperature gradient across the fuel pellet during irradiation, the grain-boundary inventory also is expected to depend on the radial position. In this work, small fragments of irradiated UO₂ were selected from different radial positions and annealed at low temperatures in air. The grain-boundary inventories were measured and the results compared with metallurgical measurements of the grain size as a function of the pellet radius.

Introduction

The fission-product inventory available for release during a nuclear reactor accident can be divided into three components: the inventory in the fuel-to-sheath gap, the inventory in the grain boundaries, and the inventory still within the grains (as atoms in solution or as bubbles). In any given accident scenario, each of these components may be released by different mechanisms at different times. Bubble migration, atomic diffusion, grain growth, and phase boundary sweeping are mechanisms which can transport fission-product atoms from within the grains to the grain boundaries.

In the past, the grain-boundary inventory of fission gases has been measured by crushing the UO₂ and collecting the released gas [1]. In this work, a new technique to open the grain boundaries was used. This technique is based on a chemical reaction which separates the grain boundaries. At 400 to 650°C, UO₂ oxidation is characterized by rapid conversion to U₃O₈ powder in a process which is best described as grain-boundary attack. Preferential oxygen penetration along grain boundaries produces oxidation to U₃O₈ and separation of the grains, as sketched in Figure 1, followed by fragmentation caused by the large volume expansion upon conversion to U₃O₈. This process has been described previously [2,3].

The purpose of this paper is to describe measurements of the fraction of the local inventory of iodine and xenon stored within the grain boundaries at different locations across the radius of a fuel pellet.

Experimental Details

Figure 2 shows a schematic of the hot-cell experimental arrangement. The irradiated fuel specimen is located in the bottom of a reaction tube where the atmosphere is controlled by the composition of a carrier gas. The temperature of the sample was controlled by raising and lowering the furnace. The carrier gas flows over the sample transporting any released volatile fission products through a series of deposition devices and filters that trap the condensable vapours (i.e., iodines). Inert gases, such as the xenons, continue and are monitored as they pass through the delay chamber. For this work, Xe-133 with a low energy (81 keV) γ-ray and I-131, with a 364 keV γ-ray were monitored. Because of its higher energy, the I-131 γ-ray can be observed through the walls of the furnace and reaction tube. Using a γ-ray spectrometer and collimator, I-131 activity in the sample can be measured directly through a port in the cell wall.
The fuel used in this experiment was irradiated to a burnup of 144 MW·h/kgU at a maximum linear power of 54 kW/m and hence a maximum central temperature below 1600°C. When the results were analyzed, it became apparent that the radial location of the sample within the pellet during irradiation was an important parameter. Because of the heavily fractured nature of the fuel, an indirect method based on cesium and zirconium content was used to determine the radial location for each sample. It is well-known that cesium migrates down the temperature gradient towards the outside of the pellet during irradiation. Zirconium, on the other hand, remains near the location where it was produced. Figure 3 shows a post-test scan of the cesium and zirconium concentrations across the fuel element taken at 0.8 mm intervals. The ratio Cs/Zr, therefore, defines the radial location from which any given sample was obtained.

Finally, metallography was done on a cross-section of the fuel element to measure the grain size. Figure 4 shows the grain structure at five radial locations.

Results and Discussion

Figure 5 shows a typical thermal cycle used for these tests. The sample was held in air at 500°C for about 3000 s. Oxidation of UO₂ in air at this temperature results in an oxidation product, U₃O₈, composed of very fine particles, typically of the order of a fraction of the grain size of the starting material [2,3]. Thus, all of the grain boundaries are fractured and xenon and iodine located at the boundaries would be released. After the period at 500°C, the samples were reheated to 850°C for about 1000 s to obtain an estimate of the initial inventory of xenon. Unlike a fractional release of iodine, which was measured directly, that of xenon has to be obtained indirectly, since the initial Xe-133 activity of the sample is unknown. Previous work [4] has shown that xenon is released in amounts dependent only on the temperature when UO₂ or U₃O₈ are heated in air. An example is shown in Figure 6, where UO₂ was oxidized to U₃O₈ at 500°C and then heated successively to 800°, 1000° and 1100°C. At each temperature ramp, there was an additional xenon release but there was no additional release during the time the sample was held at temperature. Figure 7 shows a compilation of these results for both isothermal and temperature transient experiments.

It was assumed that the fission products released by the 500°C oxidation process were released from grain-boundary inventory. Based on the results shown on Figures 6 and 7, the initial xenon inventory can be calculated by taking the ratio of the release measured at 500°C to the total release measured at 500°C and 850°C and assuming that the latter total was equal to the fraction shown for 850°C on Figure 7. Figure 8 shows the xenon activity measured at the delay chamber. The area under the curve between 48 500 s and 53 000 s represents the xenon released from the grain boundaries. The total area under the curve represents the 35% of the inventory to be expected at 850°C according to Figure 7.
Cs-137 ACTIVITY, ELEMENT AC24

Zr-95 ACTIVITY, ELEMENT AC24

FIGURE 3: CESIUM AND ZIRCONIUM CONCENTRATIONS ACROSS THE UO₂ FUEL PELLET

Figure 9 shows the percent of xenon total sample inventory located in the grain boundaries released as a function of radial location. Superimposed on these results are the grain size measurements. It is apparent that there are two populations of grain-boundary inventory results. In the outer region of the UO₂ pellet, the average inventory is 8.2%, while in the central region the average is 14%. The boundary between these two regions corresponds with the boundary between significant and insignificant grain growth which occurred at r/R = 0.4, confirming grain growth as one of the important mechanisms for the release of fission products from the grains.

Figure 10 shows the fraction of iodine released during the oxidation of the UO₂ at 500°C by direct

Pellet center. Marker is 100 μm.

Pellet mid-radius. Marker is 50 μm.

Pellet periphery. Marker is 50 μm.

FIGURE 4: METALLOGRAPHIC SECTIONS OF AS-IRRADIATED FUEL

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FIGURE 5: TYPICAL TEMPERATURE HISTORY

FIGURE 6: XE AND I RELEASE FROM \( \text{U}_2\text{O}_8 \) AS A FUNCTION OF TEMPERATURE AND TIME

FIGURE 7: RELEASE OF XENON VS. TEMPERATURE FROM \( \text{UO}_2 \) OXIDIZED IN AIR

FIGURE 8: MEASURED XENON RELEASE FOR A TYPICAL TEMPERATURE HISTORY

FIGURE 9: XENON GRAIN BOUNDARY INVENTORY AND GRAIN SIZE AS A FUNCTION OF PELLET RADIUS

FIGURE 10: IODINE GRAIN BOUNDARY INVENTORY AND GRAIN SIZE AS A FUNCTION OF PELLET RADIUS
measurements. Again, there are two groups of data separated by the degree of grain growth. In the outer region of the fuel, the average iodine grain-boundary inventory was 8.36%, in good agreement with the 8.2% measured for xenon. This result is expected since xenon and iodine are generally believed to behave similarly in UO₂ (for example, references 5 and 6). However, in the central region of the pellet, the average iodine grain-boundary inventory was 21%, half as large again as the measured xenon inventory. A plausible explanation of these results can be made by referring to Figure A. Metallography of the central region of the pellet, from about 0 to 0.4 r/R, shows that most of the grain-boundary porosity is in the form of isolated bubbles. This suggests that as pressure built up in the grain-boundary pores, they could interconnect and the resulting tunnels vent gaseous fission products to the fuel-to-sheath gap. But, the decrease in pressure after venting would allow the tunnels to reseal. Noble gases would easily escape. For the iodine not to have escaped, it must have been bound in some way. It is possible that it could be chemically bonded to another fission product; for example, with Cs as CsI. Although CsI has a boiling point of 1280°C, with the pressure to be expected in small bubbles, boiling may be suppressed. Relieving the pressure by venting the noble gases would permit boiling of CsI but the resulting time delay could well prevent the escape of significant quantities during the time that the tunnels remain open. In this way, after the first venting of the grain-boundary bubbles, there will always be an excess of iodine over xenon in the grain boundaries.

Conclusions

(1) In the peripheral region of the fuel pellet, where no grain growth has occurred, the grain-boundary inventory of xenon and iodine is similar, at about 8% of the local inventory.

(2) In the central region of the fuel pellet, where grain growth has occurred, the grain-boundary inventory of iodine is about 21% of the local inventory, while the grain-boundary inventory of xenon is about 14%.

References


EXAMINATION OF TMI-2 CORE SAMPLES IN CANADA

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ABSTRACT

Results are presented from the Canadian examination of samples from the Three Mile Island-2 (TMI-2) reactor core. This investigation is part of an international effort sponsored by the Organisation for Economic Co-operation and Development (OECD), and has provided useful information about material interactions and fission-product behaviour during severe-accident conditions. Although the data is most directly applicable to Pressurized-Water-Reactor (PWR)-specific conditions, many of the results are applicable to the generic behaviour of core materials at high temperatures. Canadian participation in this analysis exercise has contributed to the development of interpretive and analytical expertise in a range of disciplines, as well as providing access to results and data of other international studies relating to severe-fuel-damage accidents.

INTRODUCTION

The accident at TMI-2, in 1979 March, resulted in severe fuel damage, including melting, interaction and relocation of reactor core materials. Canada has participated as an OECD-Nuclear Energy Agency member country in the TMI-2 Accident Evaluation Program conducted by the United States Department of Energy, to assist in understanding the mechanisms and extent of fuel damage and fission-product release. As part of this effort, samples from various regions of the damaged core have been examined in Canada at Chalk River Nuclear Laboratories and the Whiteshell Nuclear Research Establishment, with funding from the CANDU Owners Group (COG). Individual samples were subjected to detailed examination by metallography and SEM, including EDX and WDX analyses, auto-radiography, gamma-scanning and radiocchemical analysis.

The samples examined in this project range from fuel-rod segments to ceramic rocks of previously molten material from core locations, as outlined in Figure 1. The salient features of the end-state core configuration are the large cavity in the upper grid region, the large central zone of previously molten material surrounded by a crust, the loose core debris above the previously molten material, standing rods surrounded by solidified core material, and lower plenum debris. It is now estimated that as much as 62 tonnes (45 weight%) of the original core melted and 20 tonnes relocated to the lower plenum region. A total of four fuel-rod segments were examined in this project. They originated from the undamaged core perimeter, from immediately below the molten zone, and from the upper grid assembly. Other samples were examined from the upper crust region, the central molten ceramic zone, the lower crust and from the lower plenum region.

RESULTS AND DISCUSSION

Fuel-rod segments

The fuel-rod segment M2-4, from the outside edge adjacent to the former wall of the core, showed no evidence of damage to the fuel cladding, which appeared to be ductile with a moderate amount of hydriding (57 μg/g H). The estimated temperature in this region was less than 500°C.

In contrast to M2-4, the fuel-rod segments H11-3 and G7-3-61, which had remained attached to the upper grid plate, showed a significant amount of clad oxidation with outer layers of α-Zr(O) and ZrO2 up to 40 and 80 μm thick, respectively. Both of these samples showed extensive hydriding, and analysis indicated hydrogen levels in the Zircaloy.
cladding in excess of 500 μg/g. The probable maximum clad temperatures were above 1050°C for H1-11-3 and 1600°C for C7-3-61. There was evidence of Zr-UO₂ interaction between the fuel and the clad, and desintering of UO₂ into grain-size fragments (Figure 2). This is a generic observation which would be expected to occur in any Zircaloy-clad UO₂ fuel during severe-accident conditions. Rod C7-3-61 showed a previously molten layer of variable thickness around the complete circumference, located on the inside of the clad between two layers of α-Zr(O) (Figure 3). EDX analysis showed the previously molten layer to be a Zr-U alloy (84.5 wt% Zr, 13.2 U, 1.7 Fe, 0.5 Cr) with dendrites rich in Cr and Fe (47.5 wt% Zr, 5.5 U, 26.5 Fe, 20.5 Cr). Oxygen was not analysed, but was probably a minor constituent. These compositions suggest molten stainless steel or Inconel interacting with the inner surface of the Zircaloy clad, with access through the jagged opening when the rod failed, or by melt interaction of the Inconel grid spacers with standing fuel rods. A large circumferential temperature gradient is apparent by the variation in melt interaction thickness around the rod, as shown in Figure 4. The thickness ratio of external oxide/α Zr(O) is indicative of adequate steam supply to the reacting surface, and thus there is no evidence for steam starvation during the late phase of the accident. The observed hydriding probably occurred earlier in the accident sequence.
Chemical analysis of UO₂ samples for La-139 by high-performance liquid chromatography indicated burnups of 750, 1250, and 1700 MWD/MTU for rods M2-4, Hl-11-3, and C7-3-61, respectively. These values are in agreement with burnup values derived from gamma spectroscopic measurements of Cs-137/Cs-134 ratios (Table 1). Core elevations based on the known axial power distribution were calculated using the average of the burnups from chemical and gamma analyses. These resulting estimated elevations agree with the core positions during sample retrieval operations.

The cladding of fuel-rod segment G12-R8-8, from below the crust region, showed a thin (about 10 µm) ZrO₂ layer, a fully prior beta structure, but no evidence of Zr-U-O₂ interaction, indicating a maximum cladding temperature of about 1100°C at this location.

The axial Cs-137 distribution in all the fuel rods showed the characteristic neutron-flux profiles and locations of individual fuel pellets and gaps, or damage caused due to fuel-pellet movement during shipment/handling. Significant levels of Co-60 were observed in Hl-11-3, with the Co-60 concentrated at the lower end of the rod. In sample G12-R8-8, the Cs-137 distribution was very uniform, indicating that this rod segment is from the central region of the core.

Gamma spectroscopy was used to determine the number of atoms of long half-life species in fuel rods M2-4, C7-3-61, Hl-11-3 and core bore sections K9-P1-H and K9-P4-G. The detecting efficiency was determined using a whole pellet from rod M2-4 as a calibration source of Cs-137 (662 keV), since initial U-235 enrichment and burnup (by chemical analysis) were known and allowed more reliable calculation of the Cs-137 inventory. An inverse energy dependence for the detector efficiency was used to apply the calibration to other gamma energies. The dominant activities in the fuel were Cs-137, Cs-134 and Eu-154, while cladding and melts additionally contained Co-60, Ru/Rh-106 and Sb-125 (Table 2). Significant concentrations of Co-60 were found in the metallographic sample of G7-3-61. This sample included the cladding and Inconel melt, the latter material likely containing the Co activity. Because of inhomogeneous distributions of these isotopes, concentrations could only be calculated from data in Table 2 if assumptions were made about partitioning of species between fuel and clad, crust or melt. The results for core bore samples in Table 2 are referred to in the subsequent sections.

Mapping of radial Cs-137 distributions was carried out on samples containing sufficient activity. Gamma spectra were collected over 200 seconds at discrete positions using an intrinsic Ge detector with an EG&G Ortec 575 amplifier and a Nucleus PCA card. These two-dimensional scans were performed by moving the cross section of the sample in 0.5 mm steps with an X-Y stepping device past a 1.0 mm square collimator installed between the detector and the sample. The principal activity after 10 years of decay was from Cs-137, the distribution of which is shown for rod M2-4 in Figure 5. This radial distribution was similar to ones for the other fuel rods. The flatness of the distribution indicates that there was no significant Cs redistribution during irradiation or during the accident, and that behaviour was typical of low-power irradiation to low burnup.

Gamma spectroscopy was used to determine the number of atoms of long half-life species in fuel rods M2-4, C7-3-61, Hl-11-3 and core bore sections K9-P1-H and K9-P4-G. The detecting efficiency was determined using a whole pellet from rod M2-4 as a calibration source of Cs-137 (662 keV), since initial U-235 enrichment and burnup (by chemical analysis) were known and allowed more reliable calculation of the Cs-137 inventory. An inverse energy dependence for the detector efficiency was used to apply the calibration to other gamma energies. The dominant activities in the fuel were Cs-137, Cs-134 and Eu-154, while cladding and melts additionally contained Co-60, Ru/Rh-106 and Sb-125 (Table 2). Significant concentrations of Co-60 were found in the metallographic sample of G7-3-61. This sample included the cladding and Inconel melt, the latter material likely containing the Co activity. Because of inhomogeneous distributions of these isotopes, concentrations could only be calculated from data in Table 2 if assumptions were made about partitioning of species between fuel and clad, crust or melt. The results for core bore samples in Table 2 are referred to in the subsequent sections.

### Table 1: Summary of Burnup Analyses and Sample Elevation Estimates

<table>
<thead>
<tr>
<th>Sample</th>
<th>Initial Enrichment</th>
<th>Burnup (MWD/MTU)</th>
<th>Chemical Analysis</th>
<th>Calculated Elevation (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>M2-4</td>
<td>2.94%</td>
<td>NA</td>
<td>1075</td>
<td>900</td>
</tr>
<tr>
<td>Hl-11-3</td>
<td>2.94%</td>
<td>1050</td>
<td>1200</td>
<td>31±3</td>
</tr>
<tr>
<td>C7-3-61</td>
<td>2.94%</td>
<td>2110</td>
<td>1910</td>
<td>31±3</td>
</tr>
</tbody>
</table>

### Table 2: Gamma Spectroscopic Measurements

<table>
<thead>
<tr>
<th>Sample</th>
<th>CS-137 Activity (662 keV) ROD M2-4</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
</tr>
</tbody>
</table>

![Figure 5: 137-Cs activity distribution map of rod M2-4. No overheating is evident.](image)
### TABLE 2
Calculated Quantity of Long Half-Life Gamma Emitters
(Decay Corrected to 1979/Mar/29)

<table>
<thead>
<tr>
<th>SAMPLE</th>
<th>137-Cs (+10%)</th>
<th>134-Cs (+25%)</th>
<th>154-Eu (+25%)</th>
<th>60-Co (+25%)</th>
<th>125-Sb (+25%)</th>
<th>106-Ru/Rh (+25%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>K9-P1-H/A</td>
<td>3.2X10^18</td>
<td>3.5X10^16</td>
<td>3.6X10^15</td>
<td>5.7X10^14</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(5.9 g UO_2, 7.3 g ** crust)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>K9-P1-H/B</td>
<td>1.8X10^18</td>
<td>3.6X10^16</td>
<td>3.5X10^15</td>
<td>2.6X10^17</td>
<td>1.5X10^16</td>
<td>1.1X10^18</td>
</tr>
<tr>
<td>(2.0 g UO_2, 6.2 g ** crust)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Rock 7-1</td>
<td>2.7X10^17</td>
<td>4.5X10^15</td>
<td>6.0X10^15</td>
<td>7.1X10^14</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(est. 7.9 g)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>K9-P4-G</td>
<td>1.4X10^17</td>
<td>ND</td>
<td>2.4X10^17</td>
<td>8.0X10^16</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(est. 34 g)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C7-3-61</td>
<td>2.7X10^18</td>
<td>3.0X10^16</td>
<td>4.6X10^15</td>
<td>1.4X10^15</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(5.6 g UO_2, with clad)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C7-3-61</td>
<td>1.4X10^18</td>
<td>1.0X10^16</td>
<td>ND</td>
<td>ND</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(2.7 g UO_2, no clad)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>H1-11-3</td>
<td>1.8X10^18</td>
<td>6.8X10^15</td>
<td>ND</td>
<td>ND</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(5.4 g UO_2, no clad)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>M2-4</td>
<td>1.4X10^18</td>
<td>4.4X10^15</td>
<td>ND</td>
<td>ND</td>
<td>ND</td>
<td>ND</td>
</tr>
<tr>
<td>(10.8 g UO_2, no clad)</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<td></td>
</tr>
</tbody>
</table>

* calculated from dimensions of visible UO_2 (density 10.6 g/cm^3)
** total-UO_2 mass
ND - not detected

### Upper Crust

The rock sample D8-P4-B, from the upper crust region, appeared to be an agglomerate of the previously molten material and metallic melts. Autoradiography of a section from this rock showed that most of the activity was concentrated in the one half of the section which contained voids and metallic inclusions. The other half was composed of large ceramic grains (500 μm) with some grain-boundary porosity and showed much less activity. Gamma scanning of the two different regions through a 0.635 mm by 18 mm slit collimator showed that Cs-137, Cs-134 and Eu-154 activities were approximately the same, while Sb-125 and Ru/Rh-106 were concentrated in the porous region. Figure 6 shows a large metallic inclusion, the interior of which was mainly Sn, In, Ag and Cd (control-rod material) showing no activity, whereas the surface region showed a two-phase structure with a high activity. One phase contained Ni, Sn and a small amount of Fe, and the other phase Ni, Sn and a significant Fe content. The matrix phase surrounding the inclusions was composed mainly of U, Zr and some Fe. Oxygen is also likely to be present but was not analyzed. Inclusions inside the pores of this resolidified material also contained activity. Analysis by EDX of the inclusions inside pores of a similar sample (Bottomley and Coquerelle) identified the inclusions as crystals containing Fe, Ni and Cr, and thus, the activity is likely associated with Co-60.

Chemical analysis of a slice of sample D8-P4-B (1.71 g, density 7.4 g/cm^3) was performed. The sample was prepared by fusing in KHCO_3 above 700°C with collection and scrubbing of off-gases, and dissolution of the fused mixture in HCl. Chemical and radiochemical analyses were performed using various techniques to measure U, Te, Te-99, T-129, Sr-80, Pu-239/240, Am-241/Pu-238, Cd-114, Sb-125, Eu-154/155, Co-60 and Cs-137/134 concentrations (Table 3). The sample consisted of 63 weight% U. A burnup of 2175 MWd/MTU was calculated from the Cs-137/Cs-134 ratio, assuming 1.98% enrichment of U-235, and using calculated fission-product inventories. At position D8, this burnup corresponds to an elevation of about 328 cm. Since the hardstop elevation for the upper crust is about 175 cm, it is apparent that the UO_2 in the analyzed sample originated from upper regions of the core. All of the fission-product species detected in this sample were found to be depleted compared to inventory calculations (relative to U in the sample).

Results from this sample (D8-P4-B) show that the upper crust was composed of a mixture of metallic alloys and a U/Zr oxide ceramic. Some Fe was found in both the ceramic phase and in a Ni/Sn/Fe metallic alloy. This finding sets the oxygen potential of the upper crust in the region between Ni/NiO and Fe/FeO since the alloy rich in Ni was metallic (unoxidized), and because Fe must have been oxidized in order to flux the U/Zr ceramic phase.

Fission-product Sb and Ru were found to be concentrated in the region containing metallic alloys. These isotopes were released from the fuel during the accident, but appear to be sequestered by some metallic phases. This emphasizes the role of chemical effects in influencing fission-product behaviour.
Core bore section K9-P1-H, from the lower crust region, consisted of UO₂ pellets in a matrix of partially and totally oxidized Zircaloy cladding and a number of unidentified previously liquid alloys. The sample was cut into two pieces, labelled as A and B (Figure 7a).

The heavily oxidized fuel sheath in these sections appeared as layered bands of ZrC₂ surrounding the UO₂ pellets. In some instances, ZrC₂ was located in wider radial cracks of the pellets. Oxidation of the UO₂ was observed near the pellet peripheries. In some cases, oxidation to phases resembling U₂O₈ and U₂O₇ was noted. The formation of U₂O₇ in 15 MPa steam is possible at temperatures of about 1350°C and lower. In 3 MPa steam, U₂O₇ can form at 1175°C and lower. Autoradiography of the section (Figure 7b,c) showed that most of the activity was concentrated in the pellets; however, isolated small regions of alpha-active material were in the melt region. These are probably UO₂ grains which were produced by desintering upon interaction with molten Zircaloy. Some of the open cracks in the fuel also appeared to contain higher levels of gamma activity, possibly Cs-137.

Gamma scanning of section K9-P1-H showed that the Cs-137, Cs-134 and Eu-154 were concentrated primarily in the fuel pellets. The Sb-125 activity was concentrated in the melt region between the fuel pellets, although some activity was still in the pellets. A similar distribution was seen for the Co-60 activity. The Ru-106/Rh distribution was similar to Cs and Eu, although significant levels were also found in the melt region. Comparison of isotope activities between pieces A and B (Table 2) showed that B contained 500 times more Co, and all of the detected Sb and Ru. The contents of Cs and Eu were similar for the two pieces. This indicates that the fission-product distributions in the melt were very inhomogeneous.

Burnup was calculated from the Cs-137/Cs-134 ratio. Calculated values for piece A were 2110 and 1990 MWd/MTU using the 605 and 796 keV Cs-134 peaks respectively. This corresponds to an elevation of about 15 cm up from the bottom of the fuel stack. According to the 796 keV peak, piece B had a higher burnup of 3650 MWd/MTU. It was not possible to use the 605 keV peak from piece B because of interference with a peak in the Sb-125 spectrum. The corresponding elevation for piece B would be 51 cm from the bottom of the fuel stack. Both of these elevations are consistent with core measurements of the upper and lower limits of the lower crust in plane L of the reactor grid. However, the two burnup estimates would be expected to show closer agreement with each other, as was observed for multiple samples of rod C7-3-61. Figure 8 shows the L section of the TMI-2 (adjacent to the K plane) core with estimated limits of upper and lower crust boundaries. The core positions for the other samples examined in this study, as determined from burnup, are also shown in the diagram.

The core section D8-P1-F from the lower crust contained several fuel pellets embedded with the
FIGURE 7a: Metallographic cross section of core sample K9-P1-H showing pellets embedded in a metallic melt.

FIGURE 7b: Alpha autoradiograph of core sample K9-P1-H showing activity in pellets and UO₂ fragments in the melt.
same orientation into the surrounding frozen melt. The embedded pellets were clearly marked by autoradiography. The Cs and Eu activity was concentrated in the pellet region while Sb, Co, and Rh/Rh activity existed in both pellet and surrounding matrix. In this sample, the cladding had melted and fully interacted with the UO₂ on one side, and with the surrounding material on the other side, producing an intermediate interaction layer. Figures 9 and 10 show SEM images of the UO₂/melt interaction zones. A ceramic phase of primarily U, Zr and O surround the UO₂. The metallic melt consists of three phases containing Ag, In, Cd, Sn (control-rod materials), Fe, Ni, Cr, Al (structural materials) and U, Zr, O (fuel materials). Some U was detected in the control rod phase, and this indicates either dissolution of UO₂ by this metallic phase, or interaction with metallic U produced by UO₂/Zircaloy interaction.

Ceramic Melt
Core sample K9-P4-C consisted of a ceramic rock from the central core region which contained previously molten material. It was cut into two thin sections (pieces A and B). It contained gamma activity from Cs, Eu, and Co. Piece B contained about 100 times more Co-60, 40 times more Eu-154 and half the Cs-137 detected in piece A (Table 2). This again indicates an inhomogeneous distribution of the fission-products even in the ceramic melt. If it is assumed that this ceramic is primarily composed of UO₂ with 2% U-235 enrichment, and burnup of about 4000 MWd/MTU, the measured Cs activity in pieces A and B represent 249.05% of the Cs-137 and Cs-134 inventories. The same assumptions indicate 1143% of the Eu-154 inventory in piece A, and 72044% in piece B. These results indicate some retention of both Cs and Eu in the melt, with Eu being...
The results of this investigation have provided useful information about maximum temperatures, fission-product distributions and material interactions at core positions for the samples examined. Some of the most important findings are summarized below.

Fuel-rod segments: Temperature limits during the accident were established at four core positions based on examination of intact fuel-rod segments. These limits ranged from below 500°C at the core periphery to above 1600°C at an upper/core grid position. Metallographic evidence showed that the upper/core grid fuel rods failed by molten Inconel interaction with the Zircaloy cladding. The same samples also showed moderate to heavy hydriding of cladding with extensive oxidation of Zircaloy, UO\(_2\)/Zircaloy interaction and desintering of UO\(_2\). The core elevations estimated from burnup determinations were found to be consistent with reported sampling positions. There was no significant release or redistribution of Cs from the intact fuel-rod segments. Cladding microstructures indicated that the late phase of the accident was not steam starved, although earlier phases produced hydrogen-rich gases in the upper core regions.

Upper crust: The upper crust was composed of an agglomerate of metallic inclusions in a porous ceramic matrix. Alloys of control-rod materials (Ag, In, Cd) were segregated as low activity metallic inclusions, and were intimately mixed with structural material alloys. The oxygen potential at the upper crust was set somewhere below the equilibrium for Ni/NiO and above Fe/FeO. There was significant release of volatile, semi-volatile and low-volatile fission-products from UO\(_2\). Burnup calculations indicated that the UO\(_2\) in the upper crust had relocated from upper core elevations.

Lower crust: The lower crust consisted of a multi-phase mixture of previously molten control rod, structural and fuel rod materials surrounding standing fuel pins with partially and completely oxidized cladding. A localized observation of UO\(_2\) oxidation to U\(_3\)O\(_8\) indicated an oxygen potential

FIGURE 9: UO\(_2\) / melt interaction zone in lower crust sample D8-P1-F. SEM image with elemental analysis by WDX. Magnified image shows eutectic between control-rod material and structural materials, with U, Zr segregation to grain boundaries.

1 : Ag, In, Cd, Sn, U
2 : Ni, Fe, Cr, Al
3 : U, Zr, (O)
close to pure steam, and temperatures below 1300°C. Small alpha active particles in the melt were consistent UO₂ dissolution by Zircaloy at higher elevations. Burnup of the fuel rods set the elevation of the lower crust between 15 and 51 cm from the bottom of the fuel stack. Gamma activity distributions in the lower crust were highly inhomogeneous.

Ceramic melt: The ceramic melt contained inhomogeneous distributions of gamma activity. Most of the fission-product activity had been released from the fuel, although significant amounts of Ca, Eu and Co (from structural materials) were detected.

A comparison with results from other international investigations will be made, with the objective of contributing to a consensus on material behaviour during the TMI-2 accident. Although the data is most directly applicable to PWR specific conditions, many of the results are applicable to the generic behaviour core materials at high temperatures. Canadian participation in this analysis exercise has contributed to the development of interpretive and analytical expertise in a wide range of disciplines, as well as providing access to results of other international studies and data relating to severe-fuel-damage accidents.

REFERENCES


FIGURE 10: UO₂ / melt interaction zone in lower crust sample D8-Pl-F. SEM image with elemental analysis by WDX. Magnified image shows remnants of cladding in melt interaction zone.
INTERACTION OF ZIRCALOY WITH URANIUM DIOXIDE AND STEAM AT TEMPERATURES BETWEEN 1000 AND 1700°C. VERIFICATION OF THE HITO CODE

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ABSTRACT

In a hypothetical severe fuel damage accident the zircaloy cladding may react, on its internal face, with the UO₂ fuel and, on the external face, with steam. These reactions result in the formation of (α, U₅Zr) alloy. The HITO code describes both internal and external oxidation of the Zry cladding. The diffusion equations of oxygen in the various reaction layers are solved by the finite differences method in the implicit version. Experimental results of isothermal experiments are used to check the code. Comparison between the experimental data and the results calculated with the HITO code shows a good agreement.

1. INTRODUCTION

During the normal operation of a nuclear power reactor the fuel cladding tubes remain in a temperature range such that the zircaloy (Zry) exists in thehcp structure (α-phase).

If a power cooling mismatch accident occurred, the temperature of the cladding would locally increase and a severe fuel damage (SFD) accident would take place. Under SFD conditions a combined external and internal oxidation of the fuel cladding tubes occurs.

At any temperature the reaction on the external wall of the cladding with the cooling water gives origin to a superficial oxide (ZrO₂) layer. Beneath this layer oxygen-stabilized α-Zr is produced. If the temperature is high enough, in the core of the tube wall β-Zry is also formed. This process, which is controlled by oxygen diffusion, has already been extensively studied and described in models [1-5].

On the internal surface of the cladding, if good contact between the UO₂ fuel and Zry is established, Zry reduces UO₂ to form oxygen-stabilized α-ZrO₂ and uranium metal. The uranium reacts with Zry, low in oxygen to form a (U, Zr) alloy rich in uranium. The (U, Zr) alloy lies between two α-ZrO₂ layers. The affinity of zirconium for oxygen is the driving force for the reaction.

The experimental results [6] indicate that initially the sequence of reaction layers formed radially from the center of the fuel rod to the cladding outside surface is, either in isothermal or transient experiments:

\[
\text{UO}_2 + \text{U} \rightarrow (\alpha-\text{ZrO}_2) + (\text{U,Zr}) \rightarrow (\text{U,Zr}) \text{ alloy}
\]

and will be referred to in the following as phases I to VII, respectively.

Photomicrographs reveal that phase II consists of radially elongated grains containing small amounts of uranium separated from each other by continuous channels at the grain boundaries filled with (U,Zr). This alloy, which is liquid above about 1150°C, also constitutes the phase III which is initially a smooth, closed reaction layer. For longer annealing times the phase III takes the shape of disconnected (U,Zr) globules and at certain places both α-Zr(O) phases (α and β) come into contact. This fact turns difficult the experimental determination of the interfaces position.

No uranium has been detected in phases IV and V and therefore they have the same chemical composition as the α and B phases formed during the oxidation of Zry in steam.

Oxygen is continuously removed from phase I (UO₂) until the oxide reaches the lower phase boundary given by the U-O binary phase diagram. Metallic uranium is formed which must necessarily flow across the I/II interface to react with zirconium and form the (U,Zr) alloy. The uranium most probably diffuses in the channels of phase II to reach phase III. At the same time, zirconium diffuses in the opposite direction and phase II increases its width, while, from the external surface, oxygen continues to flow into the cladding.

The multiplicity of the phenomena involved would result in a great complexity of the calculation code. Instead, a simplified analysis designed to give to the code a greater applicability was performed. This simplification is based on the fact that the two α-Zr(O) layers formed at the inside and the α-Zr(O) layer formed at the outside of the cladding grow at roughly the same rate. This is the reason why the model we develop here considers, as in [7], that only oxygen diffusion governs the formation and growth of the different phases and doesn't take into account uranium and zirconium diffusion.

Accordingly, the oxygen concentration profile is chosen to represent the state of the system. Two partial differential equations must be solved to calculate the oxygen concentration profiles in the various reaction zones and the movement of the interfaces between the zones. They are the Fick's second law and the Stefan's equation.

The interactions of the cladding on both the internal and external surfaces are initially independent of each other. The cladding behaves for each reaction as if it were a semi-infinite medium. As a result, the interfaces movement obey during the first period parabolic rate laws. Moreover, the pellet and cladding radii are sufficiently large with reference to the layers thicknesses that the cylindrical geometry can be approximated with small error by a plane geometry. Figure I.a. represents the oxygen concentration profile at an arbitrarily chosen instant during the initial stages of the interaction.

As oxidation progresses, if the annealing is long enough, the concentration at the center of the tube wall is no more negligible but it takes higher and higher values. The oxygen concentration gradient in the B-phase tends to vanish while the B-phase width
diminishes until it disappears. As a result, the phases $\alpha$-$\text{ZrO}_2$, and $\alpha$-$\text{Zr}(0)$ transform into a single phase and the number of phases diminishes from seven to five. Figure I.b. represents the concentration profile in this situation.

Later on, the $\alpha$-$\text{Zr}(0)$, the (U,Zr) alloy and the $\alpha$-$\text{Zr}(0)$ also disappear, one by one, until both oxides $\text{ZrO}_2$ and $\text{UO}_2$ become into contact. This will happen if, obviously, the experiment is sufficiently long. Figures I.c., I.d. and I.e represent these successive stages of oxidation.

The oxygen diffusion problem (Fick and Stefan's equations) can be analytically solved only during the first period, i.e., when the interfaces move according to a parabolic rate law and the cladding may be considered as two semi-infinite media. When one of these conditions fail, the problem has to be solved by numerical methods.

A. THE MODEL

2. Fundamental equations

In a previous work [7] we have studied the interaction $\text{UO}_2/\text{ZrY}$ and solved analytically the diffusion and Stefan problem under the assumptions mentioned above. We have further assumed that the geometry, although cylindrical, could be approximated for simplicity by a plane geometry. However, in the present work we have preferred to take into account the cylindrical geometry of the cladding and to employ cylindrical coordinates to solve Fick's second law and the Stefan's equation. The diffusion equation

$$\frac{\partial C}{\partial t} = D \nabla^2 C - \nabla \cdot (\mathbf{v} C)$$

is written as:

$$\frac{\partial C}{\partial t} = D \frac{\partial^2 C}{\partial r^2} + \frac{1}{r} \frac{\partial C}{\partial r} + \frac{\partial C}{\partial r}$$

(1)

The condition of conservation of volume

$$\nabla \cdot \mathbf{v} = \frac{1}{p} \frac{\partial}{\partial r} (r v_r) = 0$$

has been included. From here we obtain $v_r = \frac{\partial r}{\partial t}$. The velocity $v$ employed here is $v = v_r r$. It represents the rigid displacement of the (zirconium) lattice (which is observed in experiments like those carried out in [8]) over which the diffusion process is mounted.

As for the Stefan problem, we must look at a given interface at $\xi$, as in Fig.2, and consider that the difference between the incoming and outgoing fluxes gives origin to the interface movement

$$\frac{\partial C}{\partial t} = J_{\text{in}} - J_{\text{out}}$$

(2)

where the flux $J$ takes the general form:

$$J = -D \nabla C + v C$$

In a problem with cylindrical symmetry it is expressed as

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\[ J_r = -D \frac{\partial C}{\partial r} + v_r C \]  

(3)

where \( J_r \) and \( v_r \) are the radial components of \( J \) and \( \bar{V} \), respectively.

In addition, the conservation of Zr and U mass has to be considered. It is expressed as:

\[ \frac{dM_r}{dt} = \int C_{Zr}(r, t) \, dV = 0 \]

and a similar expression for uranium. \( C_{Zr} \) and \( C_U \) represent the Zr and U concentrations measured in weight. The integrals for Zr and U have to be divided into a certain number of integrals for each one of the phases involved. Consequently, the resulting equations take another form according to the number of phases present in the system, i.e., according to the time elapsed since the beginning of the experiment.

In general, the condition of conservation of the Zr mass yields the value of \( v_r \) which is related with the translation of the zirconium lattice.

An interesting problem arises at the interface \((U,Zr)/(ZrO_2)\). It may be seen in Fig. 2 that the oxygen concentration jump is inverse to those observed at the other interfaces. This fact is responsible for the instability of this interface. This problem is analyzed in [9]. The condition of conservation of the uranium mass is employed to fix the position of this interface.

At the phase boundaries, where the chemical reactions or the phase changes take place, a volume expansion or contraction occurs due to the density change of the lattice. However, when the 3-phase is formed, the addition of oxygen to the Zr lattice of the 3-phase modifies very little the volume of the unitary cell [9] and consequently, we have assumed that no volume change is associated to this phase change. On the contrary, when ZrO2 is formed, a significant volume change is produced. The Pilling-Bedworth factor (\( \alpha = \frac{1.66}{1} \)) relates the volume associated to one Zr atom in the oxide (ZrO2) to the corresponding in the Zr-phase. This expansion is produced at the \((U,Zr)/(ZrO_2)\) interface and "pushes" the whole oxide layer outwards with a velocity \( v_r \).

The study of the system \( UO_2/ZrY/Ar \) by Heimann et al. [6] reveals that a contraction of the cladding diameter takes place. This is due to the "disappearance" of Zr at the interface \((U,Zr)/(ZrO_2)\) and the consequent "appearance" of Zr in the growing phase III. This phenomenon is also present in the system \( UO_2/ZrY/H_2O \) vapour and is responsible for the rigid inward movement of the lattice. In the outermost phase, \( ZrO_2 \), this movement is superimposed to that due to the oxide expansion.

### 2.2. The Interfaces Velocity

#### 2.2.1. The System of seven Phases

Let us consider first the system in its first stage, i.e., when it is constituted by seven phases. With reference to Fig. 1.a., no translation exists \((v_r = 0)\) in phases I \((UO_2)\), II \((U,Zr)\), and III \((U,Zr)\) so that \( v_r \) is given by:

\[ v_r = \frac{-D_1 \frac{\partial C_1}{\partial r} + \frac{\partial C_2}{\partial r} \left( \frac{1}{\xi_1} - \frac{1}{\xi_2} \right)}{D_1 \frac{\partial C_1}{\partial r} + \frac{\partial C_2}{\partial r} \left( \frac{1}{\xi_1} - \frac{1}{\xi_2} \right)} \]

For the sake of simplicity in the computation of results, we define:

\[ v_j = \frac{-D_1 \frac{\partial C_1}{\partial r} + \frac{\partial C_2}{\partial r} \left( \frac{1}{\xi_1} - \frac{1}{\xi_2} \right)}{D_1 \frac{\partial C_1}{\partial r} + \frac{\partial C_2}{\partial r} \left( \frac{1}{\xi_1} - \frac{1}{\xi_2} \right)} \]

where the index \( j \) indicates the interface to be considered. The concentration jumps \( \Delta C_i \) are determined by subtracting the concentration at the right boundary of phase \( i \) and that at the left boundary of phase \( j \), we have then \( v_r = v_i \).

Experimental evidence [6] indicates that the interface \( \xi_2 \) remains immobile at the same location as the initial contact surface between \( UO_2 \) and \( ZrY \), which is referred to as \( \xi_p \) (pellet radius).

\[ \xi_2 = \xi_p \text{ or } v_2 = 0 \text{ at } t \]

As it was said beforehand, the uranium mass conservation condition is used to determine \( v_3 \). In fact the total uranium mass is:

\[ m_U = \pi L \left[ C_{UO_2} \xi_1^2 + C_{U}^{(U,Zr)} \xi_3 (\xi_3 - \xi_1^2) \right] \]

where \( L \) is the tube length, \( C_{UO_2} \) and \( C_{U}^{(U,Zr)} \) are the uranium concentrations (mass per unit volume) in the phases \( UO_2 \) and \( (U,Zr)\), respectively and \( g \) is the volume fraction of phase II occupied by \( (U,Zr) \). In order that \( dM_U/dt = 0 \), the following condition must hold:

\[ v_3 = \frac{-C_{UO_2} \xi_1 + (1-g) C_{U}^{(U,Zr)} \xi_3}{h \xi_3} \]

where \( h = C_{UO_2}^{(U,Zr)} / C_{UO_2} \) and \( v_2 = 0 \).

Velocities \( v_3 \) and \( v_5 \) are influenced by the translation of the Zr lattice and are given by:

\[ v_3 = v_3^0 + 1/\xi_1 \text{ and } v_5 = v_5^0 + 1/\xi_3 \]

Velocities \( v_6 \) and \( v_7 \) are also influenced by the oxide expansion. Let us look at the \((ZrO_2)/(ZrO_2)\) interface \( \xi_9 \) where the oxide is generated. If, due to diffusion only, the interface moves inwards \( d\xi_9^0 \) during \( dt \), an oxide thickness \( d\xi_9^0 \) is added to the oxide layer. These two effects cause the
interface to displace outwards by \(-t\cdot d\xi'_b\) at where \(v_b\) represents the velocity due to diffusion only, so that the term \(t/\xi_b\) has to be subtracted from the total velocity at \(\xi_b\). In summary, we have for the expansion velocity at \(\xi_b\):

\[ v_{\text{exp}}(\xi_b) = -(t-1)(v_b-1/\xi_b) \]

At \(\xi_b\) it is

\[ v_{\text{exp}}(\xi_b) = v_{\text{exp}}(\xi_b) \cdot \xi_b/\xi_b \]

The Stefan's equation applied to \(\xi_f\) gives:

\[ \frac{d\xi_f}{dt} = -v_{\text{exp}}(\xi_f) \cdot \xi_f \]

\[ \frac{d\xi_f}{dt} \right|_{\xi_f} = \frac{\xi_f}{\xi_f} + \frac{\xi_f}{\xi_f} \cdot v_{\text{exp}}(\xi_f) \cdot \xi_f \]

from where

\[ v_b = \alpha v_0^U + \xi_f/\xi_f \]

with

\[ \alpha = (C_{10}-C_3)/(C_{10}-C_3) \]

Then,

\[ v_{\text{exp}}(\xi_b) = -(t-1)C_{10}^0 \]

The velocity \(v_f\) results from the superposition of two translations: one due to the oxide expansion \(v_{\text{exp}}(\xi_f)\) and the other due to the diameter contraction \(t/\xi_f\).

\[ v_f = [1 - (t-1)\alpha v_0^U] / \xi_f \]

The condition of conservation of the Zr mass is employed to determine the parameter \(\xi_f\). The total mass of Zr is

\[ m_{2f} = \pi L \left[ (C_{10}^2(1-g) + C_{24}^2(1_{12}^{\text{Zr}} + C_{24}^2(1_{12}^{\text{Zr}}^2 - C_{12}^2) + \right. \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

where the various \(C\)'s represent the zirconium concentrations in the phases \(\alpha, \beta, (U,\text{Zr})\) and \(\text{ZrO}_2\). By using the condition \(d\xi_f/\xi_f = 0\) and defining

\[ a = C_{24}^{(1_{12}^{\text{Zr}})} / \xi_f, \quad b = C_{24}^{(1_{12}^{\text{Zr}})} / \xi_f, \quad c = 1 - b/a, \quad d = C_{24}^{(1_{12}^{\text{Zr}})} / \xi_f \]

the following expression is obtained:

\[ \xi_f = \frac{(1-c)\xi_f - (1-g)C_{12}v_{\text{exp}}C_{12}y_{\text{exp}}}{(1-1/a)(a_0v_0^U - (d/a-1)c_0v_0^U)} \]

2.2.2. The system of five phases. When the \(\beta\)-phase has disappeared, the system has for the oxygen concentration profile the appearance shown in Fig.1.b. The phases are numbered 1 \((UO_2)\), 11 \((\beta,\text{ZrO}_2)\), 111 \((U,\text{Zr})\), IV \((\alpha,\text{Zr})\) and V \((\text{ZrO}_2)\). The calculation of velocities \(v_1, v_2, v_3\) proceeds as in 2.2.1., given that the condition of conservation of the uranium mass is formulated as before. Instead, for the conservation of \(\alpha\) we write:

\[ m_c = \pi L \left[ (C_{24}^{(1_{12}^{\text{Zr}})} + C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \right. \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

\[ C_{24}^{(1_{12}^{\text{Zr}})} (C_{12}^2-C_{12}^2) + \]

from where:

\[ \xi_f = \frac{(1-g)\xi_f - (1-g)C_{12}v_{\text{exp}}C_{12}y_{\text{exp}}}{(d/a-1)c_0v_0^U} \]

For the velocities \(v_1\) and \(v_2\) we find expressions similar to those of \(v_0\) and \(v_1\) in 2.2.1.

\[ v_1 = \alpha v_0^U + 1/\xi_f \]

\[ v_2 = [1 - (t-1)\alpha v_0^U] / \xi_f \]

2.2.3. The system of four phases. Fig.1.c. represents this situation; phases I, II, and III are as before and phase IV is \(\text{ZrO}_2\).

Once the \(\alpha\)-phase has disappeared, the interface \(\xi_f\) with \(\text{ZrO}_2\) begins to advance on the \((U,\text{Zr})\) alloy, in addition to the already existing oxide \((\text{ZrO}_2)\), a new oxide \((UO_2)\) begins to develop at \(\xi_f\), proceeding from the consumed \((U,\text{Zr})\) alloy. The conditions of conservation of the \(U\) and \(\text{Zr}\) mass employed in 2.2.1. and 2. no longer hold.

The transversal movement of the \(\text{Zr}\) lattice ceases because no more free \(\text{Zr}\) is available to be transported to the \(\alpha - \text{ZrO}_2\) phase, we take \(\xi_f = 0\) from now on.

The Stefan equation applied to the interface \(\xi_f\) gives:

\[ v_4 = v_3^0 - v_{\text{exp}}(\xi_f) C_{10} / (C_0-C_{10}) \]

where \(v_{\text{exp}}(\xi_f) = -(t-1)\nu_3^0\) we then obtain:

\[ v_4 = \alpha v_3^0 \text{ with } \alpha = (C_{10}-c)/((1-C_{10}-c) \]

\[ v_3 = v_3^0 \text{ with } v_3 = v_3^0 \text{ where } v_3^0 \]

Velocities \(v_1, v_2\) are, as before, \(v_1 = v_1^0\) and \(v_2 = 0\).

2.2.4. The system of three phases. The situation is represented in Fig.1.d.; phases I and II are as before and phase III is \(\text{ZrO}_2\).
When the (U,Zr) alloy has disappeared, the immobile interface \( \alpha-Zr(\text{U})_3/\alpha-\text{U} \) has also disappeared. Both interfaces \( \xi_1 \) and \( \xi_2 \) move according to the diffusion and Stefan equations; no conservation conditions are applied. We have then:

\[
v_1 = v_1^0
\]

\[
v_2 = -v_2^0 \quad \text{and} \quad v_1 = v_1^0
\]

where \( v_2^0 = -(1-i)v_2^0 \). We then obtain:

\[
\frac{\partial C_1}{\partial t} = \frac{\partial}{\partial x} \left( D \frac{\partial C_1}{\partial x} \right) = (1-i) \frac{F}{(C_1-C_0)}
\]

The system evolves until the \( \alpha \)-phase disappears and both oxides come into contact (Fig. 1) and the single interface reaches a stationary position.

## The Numerical Method

We have chosen to solve the diffusion equation in each phase by the finite differences method in the implicit version.

The space and time domains are divided into intervals of amplitudes \( \Delta x \) and \( \Delta t \), respectively. The partial derivatives in eq.(1) are approximated as:

\[
\frac{\partial^2}{\partial x^2} \approx \frac{C_{i+1}^{K+1} - 2C_i^{K+1} + C_{i-1}^{K+1}}{\Delta x^2}
\]

\[
\frac{\partial}{\partial x} \approx \frac{C_{i+1}^{K} - 2C_i^{K} + C_{i-1}^{K}}{\Delta x}
\]

where the subscripts correspond to the space level and the superscripts to the time level. A centered scheme, which use is made evident by the expression of the first radial derivative, is adopted because it improves the convergence of the method.

Substitution in eq.(1) gives:

\[
C_i^{K+1} = (R-2z_1)C_i^{K+1} + (1+2z_1)C_i^{K+1} - (Rz_1)C_{i+1}^{K+1} - (Rz_1)C_{i-1}^{K+1} \quad \text{(4)}
\]

with \( R \) and \( z_1 \) defined as \( R = \frac{\Delta t}{\Delta x^2} \) and \( z_1 = \frac{(1-z_1)}{\rho_1 \Delta x^2} \);

\( \rho_1 \) represents the radial coordinate of the \( i \)-th node of the grid in the phase under consideration.

The expression (4) shows that every three neighboring unknowns are related by a linear algebraic equation.

Given that this is an evolutionary problem, the solution at the nodes of the net at the time \( t^K \) is calculated from the solution at the time \( t^{K+1} \), when eq.(4) is applied to every net node between the domain edges of the partial differential equation, a system of linear equations is obtained. Its solution yields the values of the unknowns \( C_i^{K} \) at any given time and space level. It may be noted that the concentrations in the different nodes are not independent of each other.

Since we are dealing with a multiphase problem, a different system of equations has to be constructed for each phase (which differ in the values of the diffusion coefficient of oxygen, \( D \), and of the grid spacing, \( \Delta x \)), according to the number of phases present at the instant under consideration. Those systems of equations are, however, connected by a common value of \( \Delta t \). It is calculated in such a way that during the time interval \( \Delta t \) any one of the interfaces reaches the nearest space node.

Each phase is divided into regular space intervals \( \Delta x \), but they differ from one to another phase because of the markedly different thicknesses of the various phases. In this way, a good accuracy can be achieved with a reasonably low number of points in each phase.

The procedure to calculate \( \Delta t \) is as follows: starting from the concentration profile at a given instant, the Stefan equation (2) is solved with the aid of eq.(3), and the velocity \( \rho_1 \) of each interface is found. The time necessary for each interface to reach the next net node is calculated and the minimum among them is chosen as the value of \( \Delta t \).

Let us consider a phase where the spatial subscript \( i \) runs from 0 to \( H+1 \) nodes. The concentrations at the left and right boundaries of the phase, \( C_0 \) and \( C_H \), respectively, are time independent (they are functions of temperature only). These boundaries may or not coincide with spatial nodes. The situation is represented in Fig. 1 where \( P \) and \( q \) measure the fractions of the grid by which each boundary is separated from the nearest node.

![Fig. 1](image)

Equation (4) requires of equally spaced points, so that the subscript \( i \) is constrained to run from 1 to \( H+1 \). In order to determine the concentrations at the nodes 0 and \( H \), a parabolic interpolation equation is employed, it means, \( C_0 \) is interpolated between \( C_1 \) and \( C_2 \) while \( C_H \) is interpolated between \( C_H-1 \) and \( C_H+1 \).

The system of \( H+1 \) linear equations with \( H+1 \) unknowns can be written in the matrix form:

\[
\begin{bmatrix}
C_1 \\
C_2 \\
C_3 \\
\vdots \\
C_H \\
C_{H+1}
\end{bmatrix}^{K+1} = \begin{bmatrix}
D_1 & C_1 & 0 & \cdots & 0 \\
C_2 & C_3 & C_3 & \cdots & 0 \\
C_3 & C_3 & C_3 & \cdots & 0 \\
\vdots & \vdots & \vdots & \ddots & \vdots \\
D_H & 0 & 0 & \cdots & C_{H+1}
\end{bmatrix}^{K+1} \begin{bmatrix}
C_1 \\
C_2 \\
C_3 \\
\vdots \\
C_H \\
C_{H+1}
\end{bmatrix}^K
\]

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where the matrix elements $a$, $b$, and $c$ are given by:

\[
\begin{align*}
    a_1 &= 0 \\
    a_1 &= 1 + 2R \left(1 - \frac{Z_1}{R}\right) / (1+p) \\
    a_1 &= -2R \left(1 + \frac{Z_1}{R}\right) / (2+p) \\
    a_1 &= -BZ_1 \\
    b_1 &= 1 + 2R \\
    c_1 &= -KZ_1 \\
    a_{H-1} &= -2R \left(1 + \frac{Z_{H-1}}{R}\right) / (2+p) \\
    b_{H-1} &= 1 + 2R \left(1 - \frac{Z_{H-1}}{2}\right) / (1+p) \\
    c_{H-1} &= 0
\end{align*}
\]

$lb$ (left boundary) and $rb$ (right boundary) are:

\[
\begin{align*}
    lb &= 2K \left(1 - \frac{Z_1}{R}\right) / (1+p) \\
    rb &= 2K \left(1 - \frac{Z_{H-1}}{R}\right) / (1+p)
\end{align*}
\]

A tridiagonal system like this one is stable if the matrix is diagonal dominant, i.e., if $|b_i| > |a_i| + |a_i|$. This condition is fulfilled in our problem.

Starting from the concentration profile at the time level $K$, the solution at the time level $K+1$ is attained by inversion of the matrix (one for each phase). This procedure is quite simple due to the sparsity of the matrix elements.

4. THE DIFFUSION CONSTANTS

In a previous work [10] the interaction $\text{UO}_2/\text{Zr}$ was analyzed. The interface positions were experimentally determined for different times and temperatures in the range 1000 to 1700°C, during the period of parabolic kinetics (the interface positions are proportional to $t^\frac{1}{2}$). A model similar to the one presented here, but restricted to the parabolic rate period, was presented in that work, in order that the equations of the model had solution, the oxygen concentrations at the boundaries and the diffusion coefficients of oxygen were fitted to the experimental values, for each temperature, in those phases where the composition is such that no reliable information could be obtained from the phase diagrams. Once adequate solutions were found for each temperature, analytical expressions were fitted for the oxygen concentrations at the interfaces and the diffusion coefficients, as functions of temperature. The values used in [10] together with those (well known) corresponding to the interaction $\text{Zr}/\text{H}_2\text{O}$ are listed below; the calculation method and the reference are also indicated.

\[
\begin{align*}
    C_1 &= 1290.6 \text{ mg/cm}^3 \quad [10] \\
    C_2 &= \begin{cases} 
        1291.1 \exp[5.036x10^{-6} T(K) - 1064.41] \text{ mg/cm}^3 & \text{for } 1273 K < T < 1793 K \\
        1277.5 + 0.0584x10^{-5} T(K) \text{ mg/cm}^3 & \text{for } 1793 K < T < 2023 K \end{cases} \quad [10] \\
    C_3 &= \frac{X_0}{1 - X_0} \quad 1.1186 \text{ mg/cm}^3 \quad [10] \\
    C_4 &= \frac{X_0}{1 - X_0} \quad 1.1186 \text{ mg/cm}^3 \quad [10]
\end{align*}
\]

where the atomic fraction of oxygen in the saturated $\alpha$-$\text{Zr}(\text{O})$, $X_0$, is calculated from the binary phase diagram $\text{Zr}/\text{O}$ [11].

\[
\begin{align*}
    C_5 &= 35 \text{ mg/cm}^3 \quad \text{estimated value, lower than the AES detection limit [6, 10].} \\
    C_6 &= \begin{cases} 
        39.8 \text{ mg/cm}^3 & \text{for } 1273 K < T < 1523 K \\
        44.0 - 5.933x10^{-5} T(K) \text{ mg/cm}^3 & \text{for } 1551 K < T < 1793 K \end{cases} \quad [10] \\
    C_7 &= \begin{cases} 
        1291.1 \exp[5.036x10^{-6} T(K) - 1064.41] \text{ mg/cm}^3 & \text{for } 1273 K < T < 1523 K \\
        1277.5 + 0.0584x10^{-5} T(K) \text{ mg/cm}^3 & \text{for } 1793 K < T < 2023 K \end{cases} \quad [10] \\
    C_8 &= \begin{cases} 
        1.1186 \text{ mg/cm}^3 & \text{for } i = 8, 9 \quad [10]
\end{cases}
\end{align*}
\]

where the weight per cent in the $\alpha$- and $\delta$-phase boundaries, $w_\alpha$ and $w_\delta$, were taken from the pseudo-binary diagram for $\text{Zr}^2/\text{O}$ [12].

\[
\begin{align*}
    w_\alpha &= \exp(-79400 / R T(K)) \quad \text{for } 1273 K < T < 1523 K \\
    w_\delta &= \exp(-12670 / R T(K)) \quad \text{for } 1523 K < T < 2023 K \\
    w_\alpha &= \exp(-79400 / R T(K)) \quad \text{for } 1273 K < T < 1523 K \\
    w_\delta &= \exp(-12670 / R T(K)) \quad \text{for } 1523 K < T < 2023 K \\
    w_\alpha &= \exp(-79400 / R T(K)) \quad \text{for } 1273 K < T < 1523 K \\
    w_\delta &= \exp(-12670 / R T(K)) \quad \text{for } 1523 K < T < 2023 K \\
\end{align*}
\]

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The diffusion coefficients are measured in cm²/s, the activation energies in cal/mol and the constant K is 5.4 cal/K mol.

With reference to the calculation of the interfacial velocities, it should be noted that in all those cases in which a velocity is obliged to take a value given, for instance, by a conservation condition, which is violated in order to restabilize this equation, one of the interfacial concentrations, that at the right boundary, Cₘ of the phase at the left of the interface, has to be modified. This means, for example, that the modification of Vₚ due to uranium conservation obliges to modify Cₘ in the phase III.

II. THE CALCULATION PROCEDURE

The isothermal experiments we want to simulate exist in reality of an initial heating-up transient, during which temperature rises at a rate of 10 K/s, followed by an isothermal period of length and temperature variables from one experiment to another, and a cool-down transient of -5 K/s.

The calculation starts at 1000°C with the determination of the oxygen concentrations at the phase boundaries and the diffusion coefficients as indicated in I. (Subroutine L7).

The finite difference method requires an initial solution to start the calculation. It is analytically obtained under the assumptions mentioned in I. of: i) the cladding behaves either for the internal or the external reaction as if it were a semi-infinite medium; ii) the interfacial movements obey parabolic rate laws; iii) the cylindrical geometry can be approximated by a plane geometry. The oxygen concentration profile at the first five and last three phases are determined independently (Subroutines ANSb and ANSy) and then matched at the center of the cladding wall.

The interfacial positions at the initial instant are also obtained. With them a grid is defined in each phase (Subroutine GD1) in such a way that each phase has between 8 and 20 nodes. It is worth noting that after a few seconds oxidation certain phases are very narrow, especially (KZ) which may have a few angstroms while others are very thick, especially (Z) which may have several hundreds angstroms.

Given that there are several assumptions implied in the initial solution which may be responsible of certain error, it seems reasonable to choose the initial instant well near t₀. However, the thinness of the IVth phase gives origin to a marked increase in the calculation time. We have found that t₀ = 5 s is a good choice for the initial instant.

With the first concentration profile already built, we are in conditions to start with the calculation loop that consists of:

1. Calculation of the interfacial velocities according to the equations given in 2.2.
2. Calculation of the time step Δt in the way described in 1. (Subroutine M11).

During the temperature transients, a large value of Δt may be responsible for a significant error, so that we have found it necessary to limit the temperature variation in one iteration by limiting the time step. As the interfaces move and considering that the special grids of neighbouring phases are in general different, it may happen during a time step that new points have to be added either at the left or right edges of a phase or that the extreme points disappear. All of these possibilities, i.e., the changes of m, n and q (Fig.4) in each phase are included in this subroutine.

-- Once the value of Δt is determined, we have the time t₀ for the next iteration. This means that we calculate the new temperature (function TEPH) and diffusion constants (Subroutine W7).

-- In order to carry out the time evolution the matrix defined in 3. has to be solved. Subroutine M111111 generates the matrix elements and subroutine THI performs the matrix inversion. With this operation the calculation loop is completed.

Other aspects are being checked in each iteration as for example the number of points of each phase. If this number is very high (about 50) it is reduced to its half by taking one point every two points and a new grid of double width is built.

As it was said beforehand the p-phase is the first to disappear, but its grid is the thicker of all the grids, so that when it is described by a few nodes, its thickness is still sufficiently large in comparison with the other phases. In this situation, a point is interpolated every two points (Subroutine M1111111), the grid points are renumbered, and new values for n, p and q are calculated.

When one of the phases disappears (t₁0) the remaining phases are renumbered.

VI. RESULTS AND CONCLUSIONS

A variety of isothermal experiments were carried out [15] at different maximum temperatures and durations. During the test conduct the temperature is raised up to the maximum temperature and after 60 seconds, during which thermal equilibrium is established, the external pressure is applied to produce a good fuel cladding solid contact. It is important to underline that the external oxidation starts when temperature begins to rise while the internal reaction starts later, when the external pressure is applied.

Figures 5, 6, 7, 8 and 9 show the experimental results together with the calculated HEITO results for nominal annealing temperatures of 1000, 1200, 1300 and 1400°C, respectively. The abscissa represents the time at maximum temperature (it does not include the heat-up and cool-down transient times) and the ordinate represents the position of the interface measured from the interface with the fuel (Fig.1).

The agreement is in general very good although a certain scatter is observed at the lower temperatures in the curves representing the internal interaction. This small disagreement seems due to the nature of the experimental conduct. In fact, the lower curves were displaced to the right to take into account the delay in the application of the external pressure, the agreement would be improved. The disagreement between experimental points and calculated curves is as less visible for the higher temperatures since a longer time is implied in the temperature transients.

The HEITO code gives an accurate representation...
of the disappearance of the phases as the oxidation progresses, confirming that the hypothesis proposed for the validity of the conservation equations are correct.

HTO is a good code to simulate the experiments of interaction of UO\(_2\)/Zry/water vapor. It could be improved by:

--- the addition of the equations necessary to include a time dependent thermal gradient.

--- its convoluation in transient experiments.

--- a more detailed study of the disappearance of the d-phase where the major discrepancies are found. The code TRANSiX \([10]\) which describes the precipitation of α-grains in the μ-matrix during cooling should be incorporated to HTO.

REFERENCES


A major experimental program has been established at the Chalk River Nuclear Laboratories (CRNL) that will provide essential data on the thermal and mechanical behaviour of nuclear fuel under abnormal reactor operating conditions and on the transient release, transport and deposition of fission product activity from severely degraded fuel. A number of severe fuel damage (SFD) experiments will be conducted within the Blowdown Test Facility (BTF) at CRNL. A series of experiments are being conducted to commission this new facility prior to the SFD program.

This paper describes the features and the commissioning program for the BTF. A development and testing program is described for critical components used on the reactor test section. In-reactor commissioning with a fuel assembly simulator commenced in 1989 June and preliminary results are given. The paper also outlines plans for future all-effects, in-reactor tests of CANDU-designed fuel.

INTRODUCTION

One objective of the Canadian research program into the behaviour of CANDU fuel under abnormal reactor operating conditions, such as Loss-Of-Coolant Accidents (LOCA) and Severe Fuel Damage (SFD) accidents, is to develop physically-based computer models and to verify them against experimental data (1-4). During the last 15 years, concurrent modelling and experimental programs have developed an understanding of (a) the mechanical strength and rupture behaviour of the Zircaloy fuel sheath (5-8), (b) fission product release from the UO₂ fuel (9,10) and (c) the oxidation of UO₂ and its effects on activity release during transient conditions where sheath temperatures remain below about 1200 °C.

The in-reactor, experimental, LOCA program at Chalk River started with low temperature conditions in the X-2 loop of the NRX reactor (9) and has now progressed to the investigation of fuel behaviour and activity release from a defective fuel element during elevated temperature transients (1200-2500 °C). Under this program, the release, transport and deposition of fission products - in vapour, aqueous and solid phases - will be measured on-line during the transient. To meet these objectives, the Blowdown Test Facility (BTF) was constructed at Chalk River.

A series of experiments is being conducted to commission this new facility prior to LOCA and SFD testing. This commissioning program includes:

1. Non-nuclear, out-reactor commissioning: Testing and calibration of process systems and instrumentation; Software development and quality assurance checks for both the data acquisition systems and the Programmable Logic Controller; Laboratory testing of the reactor test section seals.

2. Non-nuclear, in-reactor testing: (a) Steady-state measurements of the heat loss from the test section to the reactor moderator, of the fission heat produced from test section components during reactor operation and of the thermal neutron flux distribution, and (b) Coolant depressurization transients (blowdowns) with an instrumented stringer, which simulates an experimental fuel assembly, installed in the test section. This experiment is called Exp-BTF-10006.

3. Nuclear, in-reactor testing: An instrumented, stainless-steel-clad, UO₂, 3-fuel element assembly in the test section will be subjected to blowdowns after the BTF-10006 test. This test is called Exp-BTF-102.

This paper describes the principal features of the Blowdown Test Facility and of its commissioning program. The performance of the seals, which are used on the reactor test section of the BTF, was evaluated through laboratory testing and results are summarized. A description of each in-reactor commissioning experiment is provided. Preliminary results from the BTF-10006 in-reactor test are presented. This is followed with an outline for future in-reactor SFD and LOCA experiments.
DESCRIPTION OF THE BLOWDOWN TEST FACILITY

The vertical test section of the BTF is located in the E-12 lattice position of the NRU experimental reactor (Figure 1). Coolant for the test section and the experimental fuel is supplied from the U-1 loop. The test section is of a re-entrant design. The coolant enters the bottom of the test section and flows up and into the annulus between the pressure tube and a flow divider called the re-entry tube (Figure 2). Just above the reactor core, the coolant is directed into the centre of the re-entry tube and flows downward and over the test fuel. The coolant exits the test section again at the bottom of the vessel.

The active core length of the NRU reactor is about 3 m. Depending on the objectives of the experiment, the BTF test section may accommodate fuel bundles with up to seven CANDU fuel elements or a 3x3 array of full-length Pressurized Light Water Reactor (PLWR) fuel rods.

The fuel may be cooled with either pressurized water or steam while the reactor is operating at full power. At decay power levels of the test fuel, cooling may be controlled with either a steam purge (2-40 g/s), a helium purge (0-1 L/s), or a combination of these two cooling modes. A reservoir of cold water is also available, if required, to quench the fuel after a LOCA or SFD transient. Mass flows of up to 4 kg/s may be injected into the test section to quench the fuel.

During depressurization of the test section, the coolant mass flow is reduced to very low levels. Without adequate cooling, the heat transfer from the fuel sheath to the coolant becomes impaired resulting in elevated temperatures on the fuel element. Fuel sheath temperatures may be controlled during the blowdown phase of the experiment with changes in reactor power, steam purge flow, helium purge flow, or a combination of these parameters. During a SFD experiment, the test fuel is surrounded by a thermal shroud to limit the radial heat flux to, and prevent overheating of, the pressure tube (Figure 2). The temperature of, and the pressure within, the thermal shroud will be monitored during the irradiation. Through this data, the performance of the shroud under steady-state and transient reactor operations will be assessed.

The pressure tube has been instrumented with sixteen thermocouples and seventeen self-powered neutron detectors. The thermocouples measure both axial and circumferential temperature gradients on the tube. Three of these instruments are connected to alarms and reactor trip circuits. The axial neutron flux profile is measured with a variety of platinum, vanadium and cobalt detectors.

There are two options to terminate a high temperature (LOCA or SFD) transient. The fuel may be cooled quickly with cold (reheat) water or more slowly with enhanced steam purge flows. In either case, the reactor is shut down at the conclusion of the transient.

During a LOCA or SFD experiment, the coolant of the purge flow would sweep all hydrogen generated and any fission products released from the failed test fuel out of the BTF test section, down the blowdown line, and into the blowdown tank. The BTF is equipped with a number of monitoring systems to measure coolant properties (temperature, pressure, flow, density, momentum), wall temperatures of the blowdown line, fission product activity, and hydrogen generation during an experiment (Figure 3). Coupons may be installed in both the reactor test section located directly below the test fuel and in the blowdown tank. Post-irradiation examination of these coupons will provide useful data on the deposition of important fission products onto materials typically found in the containment of a reactor building. Up to 14 samples of effluent from the blowdown line may be analyzed by a gamma spectrometer located in the fission product activity sampling pit. This spectrometer may be remotely moved in the horizontal and vertical directions thereby measuring the gross gamma spectra at and in the vicinity of each sample. A sample is collected in a separate, 3 mm diameter, stainless steel, u-shaped tube. The u-tubes have been designed to retain liquid in their lower, horizontal portion while vapour may float above in each of their two vertical legs. Consequently, partition coefficients may be determined for fission products in both the liquid and vapour phases.

The Programmable Logic Controller (PLC) controls the operation of all important valves of the facility, such as the two test section isolation valves, the blowdown valve, and each pair of solenoid isolating valves on each sample u-tube. It also relates the status of these valves to the principle data acquisition system, REDNET.

Figure 2: Principal components of the reactor test section of the Blowdown Test Facility with a trefoil fuel assembly.
COMMISSIONING PROGRAM

Major commissioning activities for the Blowdown Test Facility include: (i) the development and testing of the two seals used on the reactor test section, (ii) in-reactor non-nuclear testing and (iii) nuclear testing.

BTF Seal Development and Testing

Test Section Pressure Boundary Seal (Top Closure)

A variety of instruments are used on an in-reactor, experimental fuel assembly to measure local coolant properties and fuel element conditions, including thermocouples, pressure transducers, flowmeters and special-purpose valves. The leads from these instruments must pass through the seal at the top of the BTF reactor test section, and hence the test section pressure boundary, to permit data logging during the irradiation. In addition, for some experiments there may be a requirement to pass small diameter tubing through this pressure boundary to permit the injection of inert gas into the test assembly.

A development and qualification program was successfully completed on a graphite-based seal gland to permit up to 77 penetrations for instrument leads and tubing (Figure 4).

Hydrostatic Pressure Test:

Each top closure seal must successfully pass three hydrostatic pressure tests at 1.5 times the design pressure, i.e. under the following conditions:

- pressure = 20.7 MPa, and
- temperature = 21°C
The acceptance criterion for this test was a pressure drop less than 0.07 MPa over a 10 minute period.

Autoclave Tests:

Static Test

Each seal gland passed 3 static tests at 13.9 MPa pressure and 335°C temperature for a period of 10 days each. The average leakage rate detected over a 10 day period did not exceed 167 μg/s of liquid water. At least one static test was successfully performed upon completion of a blowdown test: no significant change in leak rate was observed.

Blowdown Test

Each seal gland was subjected to a minimum of 10 depressurization transients (blowdowns) from a pressure of 13.9 MPa to 0.1 MPa within a period of 50 s or less. The water leak rate did not exceed the acceptable limit of 167 μg/s. The pre-blowdown temperature of the seal gland was set at 335°C and held at this level for about 2 hours before each blowdown.

Thermal Cycle Tests

Each seal was also subjected to 20 thermal cycles starting from a normal loop operating temperature of 299°C, then dropped to about 50 °C for a 1 to 2 hour period with fan cooling, and then returned to 299°C. The pressure of the water was maintained at about 8.6 MPa throughout the thermal cycle. The average leakage rate detected over this period was maintained below the acceptance limit of 167 μg/s of liquid water.

BTF Test Section Sliding Seal

The reactor test section is of a re-entrant design. During high temperature conditions within the test section, the inner Zircaloy re-entry tube may thermally expand at a different rate than that of the outer, stainless steel pressure tube. This differential expansion is accommodated by a two-component sliding seal at the bottom of the test section.

The primary seal is based on wedged-shaped packed graphite while two metallic c-rings serve as secondary seals (Figure 5). An extensive development and testing program was conducted to qualify the performance of the seal design under conditions expected within the BTF test section:

1. Pressurized water cooling at 10 MPa and 290 °C which are typically used to pre-condition fuel and generate an inventory of fission product activity within the fuel matrix.
2. Steam cooling at 10 MPa and 311 °C just prior to a LOCA or SFD transient.
3. High temperature, low pressure testing at 0.35 MPa and about 770 °C which are expected conditions in the test section during a SFD experiment.

Prototype seals were tested in a climate-controlled pressure vessel (Figure 6). Differential pressures of up to 350 kPa were applied across the sealing surface. Heaters were used to subject the seal to the desired temperature. The seal was stroked over a 12.5 mm distance several times and the leakage was measured both before and after stroking. The test duration was varied for each of the above three testing environments. The principle acceptance criteria for the seal design were based on leak below:

1. 120 g/s for pressurized water conditions.
2. 75 g/s for high pressure steam conditions.
3. 2 g/s for high temperature, low pressure steam conditions.

Results

Tests performed for up to 15 days under pressurized water cooling did not indicate significant leakage (<5 g/s) past the seal. Acceptably small frictional forces of about 450 N were required to stroke the seal.

Tests performed for up to five hours under high pressure steam conditions showed negligible leakage (< 5 g/s) past the seal. The frictional force required to stroke the seal was also about 450 N.

Tests performed between 3 and 5 hours under low pressure, high temperature (770 °C), steam conditions were also very successful with leakage rates below 0.25 g/s. Stroking of the seal was achieved with about 150 N force.

Degradation of the graphite seal is not expected to be a serious problem under the operating conditions planned for the BTF. Prototypic graphite seals were machined with three longitudinal slots, which had transverse cross-sectional areas of 0.06 mm², 0.25 mm², and 0.56 mm², to simulate a defect, and then tested under the same thermalhydraulic conditions. Leakage past all purposely-flawed seals remained within the above acceptance criteria and degradation of the seal material was observed to be minimal.
Additional tests were performed with the primary graphite seal removed and with only the secondary, metallic c-rings installed. Leakage rates past this seal were marginally higher than the dual seal configuration but still within all acceptance criteria.

Non-Nuclear, In-Reactor Commissioning

The goal of the BTF-10006 test is to characterize the Blowdown Test Facility with a instrumented, unfuelled assembly. This experiment focused on the thermal-hydraulic performance of the facility. The specific objectives of this experiment were to perform the following activities:

1. In-situ calibration, with a travelling fission chamber, of the seventeen self-powered neutron flux detectors located on the BTF pressure tube.
2. Measure the gamma heat production of the test section components and the heat loss to the moderator under steady-state reactor operations.
3. Complete in-situ calibration of the BTF gamma spectrometry system.
5. Measure axial neutron flux profiles within the reactor test section.
6. Allow the operating staff and experimenters to become familiar with the facility characteristics and procedures.

Description of the In-Reactor Simulator

The design of the in-reactor Fuel Test Assembly Simulator (FTAS) was similar to that of the instrumented, fuelled test assembly with the noted exception that it contained no fissile material. The distribution of materials and the pressure drop are similar between the FTAS design and fuelled assembly design. This permitted an evaluation of the in-reactor performance of key components on the assembly such as the test section pressure boundary seal and the thermal shroud without the potential complication of fuel element failure and fission product release.

The simulator was instrumented to measure local temperatures of the coolant (8 Type K thermocouples) and of the thermal shroud (7 Type K thermocouples). A bi-directional turbine flowmeter provided in-core, volumetric flow data during the irradiation. These instruments were replicated with others located on the facility that monitor coolant properties (temperatures, pressures and flow) at the inlet and outlet of the test section.

The simulator was also designed to permit the insertion of a travelling fission chamber. This chamber was used to characterize the axial neutron flux profile of the BTF test section and to calibrate the pressure tube neutron detectors.

Results

A preliminary analysis of the available data follows.

The reactor test section was calibrated for heat loss to the moderator at zero-reactor power and for gamma heat production at a variety of reactor power levels. These calibrations were obtained under steady-state operating conditions. Linear regression analysis was used to develop the following correlations:

Heat Loss = \(-7.458 \times 0.159 \times T_2 - 0.021 \times JRU\)

200 °C < T_2 < 310 °C and subcooled.

0 MW < JRU < 125 MW.

Gamma Heat = 8.78 + 0.916 \times JRU

0 MW < JRU < 125 MW.

where T_2 is the temperature of the coolant at the inlet of the test section and JRU is the reactor thermal power.

Insertion of the travelling fission chamber into the test section with the reactor at a variety of steady-state power levels confirmed that the flux distribution is relatively flat over a 1.00 m length in the central region of the core. The maximum flux at the test section could be related to the steady-state thermal power of the reactor:

<table>
<thead>
<tr>
<th>Reactor Power (MW)</th>
<th>Maximum Neutron Flux [n/(s·cm^2)]</th>
</tr>
</thead>
<tbody>
<tr>
<td>12</td>
<td>(1.04 \times 10^7)</td>
</tr>
<tr>
<td>60</td>
<td>(5.04 \times 10^7)</td>
</tr>
<tr>
<td>120</td>
<td>(7.55 \times 10^7)</td>
</tr>
</tbody>
</table>

Figure 6: Schematic of the out-reactor facility for the development and testing of the BTF test section sliding seal.
Seven blowdown transients were successfully completed with the simulator in the reactor test section. Several conclusions were drawn from this experiment:

1. The reactor could be restarted after an initial conditional trip or shutdown.
2. The reactor could be controlled at low powers levels (i.e., 5 MW to 20 MW) for periods up to 45 minutes followed by a successful return to full power operation (Figure 7).
3. Temperatures of the pressure tube were maintained below the design limit during a high temperature transient (Figure 7).
4. The thermal shroud survived seven harsh thermal-hydraulic and high temperature transients.
5. The BTF Steam Purge System successfully provided a controlled flow of steam in the range of 2 g/s to 25 g/s (Figure 7).
6. The two reactor test section seals performed well under steady-state and high temperature transient conditions with no evidence of deterioration of performance with time.

**Nuclear, Ti-Reactor Commissioning**

The goal of the BTF-102 experiment is to characterize the Blowdown Test Facility with a instrumented, UO₂-fuelled assembly. The specific objectives of this experiment are to perform the following activities:

1. Measure the temperature of the components on a nuclear-fuelled assembly and of the BTF reactor test section during a blowdown transient of the test section resulting in maximum fuel sheath temperatures in the range of 400 °C to 1150 °C.
2. Demonstrate the ability to control fuel element temperatures through changes in reactor power, steam purge flow, inert gas flow, and a combination of these parameters.
3. Demonstrate the ability of the thermal shroud to maintain low temperatures on the pressure tube during high temperature, transient fuel assembly operation.
4. Measure the fuel assembly cooling rates after high temperature operation.
5. Further commission the low flow steam purge system and begin the commissioning of the BTF Inert Gas Purge System.
6. Further test fabrication, assembly, instrumentation, and experimental operating procedures in advance of conducting LOCA and SFD tests in the BTF.

Two instrumented fuel assemblies are being constructed for this experiment. Each assembly contains three stainless steel-clad, UO₂, fuel elements. The fuel centreline temperature of each element is measured with a W/Re, mineral-insulated thermocouple. Seven Type-K thermocouples will be welded to the stainless steel sheath of each fuel element. Five will be laser-welded to the sheath's outside surface. The other two will measure the inside wall temperature. Consequently, axial, radial and circumferential temperature gradients will be determined on each fuel sheath during the operating period of these thermocouples.

A development program was successfully conducted to establish the optimum welding parameters required to attach thermocouples to the inner surface of the stainless-steel fuel sheathing and to commission the welding apparatus and procedures. The development program was expanded to permit the welding of thermocouples to the inside wall of Zircaloy fuel cladding which will be used on fresh (un-irradiated) fuel elements to be tested in future LOCA or SFD experiments.

In addition to these instruments, each BTF-102 fuel assembly will be equipped with a bi-directional turbine flowmeter, an eddy-current pressure transducer, and a number of thermocouples to measure coolant volumetric flow, pressure and temperature in the vicinity of the test fuel.

A prototypic thermal shroud is used on each BTF-102 fuel assembly to test the effectiveness of the shroud design for limiting radial heat transfer during high temperature transients (Figure 2). Each shroud is instrumented to measure the internal pressure and wall temperatures during the irradiation. Krypton is used as the filling gas for the shroud because of its low thermal conductivity.

A number of blowdown transients will be conducted during this nuclear commissioning experiment for the BTF (11). A summary of the proposed test matrix is given in Table 1.
Table 1
BTF Nuclear Commissioning Test Matrix

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Series One</th>
<th>Series Two</th>
<th>Series Three</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of Tests</td>
<td>3</td>
<td>3-6</td>
<td></td>
</tr>
<tr>
<td>Target Sheath Temp. (°C)</td>
<td>400</td>
<td>700</td>
<td>950</td>
</tr>
<tr>
<td></td>
<td>to</td>
<td>to</td>
<td>to</td>
</tr>
<tr>
<td>Reactor Power</td>
<td>Low</td>
<td>Low</td>
<td>Low</td>
</tr>
<tr>
<td>Steal</td>
<td>Purge Flow (L/s)</td>
<td>2-5</td>
<td>2-5</td>
</tr>
<tr>
<td>Inert Gas Purge Flow (L/s)</td>
<td>0-1</td>
<td>0-1</td>
<td>0-1</td>
</tr>
<tr>
<td>Cooling Mode</td>
<td>Quench &amp; Slow</td>
<td>Quench &amp; Slow</td>
<td>Slow</td>
</tr>
</tbody>
</table>

EXPERIMENTAL PROGRAM

The principal purpose of the Blowdown Test Facility is to provide experimental data from integrated, all-effects tests on CANDU and other nuclear fuel designs. Through analysis of the data and destructive post-irradiation examination of the fuel, a better understanding will be achieved of fuel and fission product behaviour under postulated reactor accident,"LOCA and SFD' conditions. This data will be used to validate computer codes that calculate steady-state and high temperature transient fuel performance and fission product release, transport and deposition over a wide range of thermal-hydraulic and power conditions. This validation process and further application to reactor safety studies provides confidence that all important physical phenomena have been incorporated in the process models of these codes.

Several in-reactor experiments are planned. They can be grouped into three main categories:

1. Steady-state irradiations. Zircaloy-clad, CANDU-design fuel elements are conditioned under typical power (15-60 kW/m) and burnup (125-140 MW.h/kg U) levels to be subsequently used in LOCA and SFD experiments. Table 2 summarizes the characteristics of this fuel.

2. Loss-of-Coolant Accident tests. Instrumented, Zircaloy-clad, CANDU-design fuel elements are subjected to LOCA conditions to expand the database of in-reactor, integrated experiments (9) and to provide further data on clad deformation and failure in the 1200 °C to 1400 °C temperature regime. The first BTF LOCA experiment is scheduled for 1990.

3. Severe Fuel Damage tests. Instrumented, Zircaloy-clad, CANDU-design fuel elements will be subjected to temperatures within the range of 1300 °C to 2500 °C. Table 3 summarizes the four experiments conducted over the next three years to be conducted within the Blowdown Test Facility.

Table 2
Characteristics of Conditioned Fuel For In-Reactor Experiments

(a) Twelve CANDU fuel elements conditioned:
- To study fuel behaviour and fission product activity release under normal reactor operations and LOCA conditions.
- 45 kW/m to 60 kW/m liner rating.
- 125 to 140 MW.h/kg U nominal burnup.
- UO₂ - 5% enriched with ²³⁵U ceramic fuel.
- Zircaloy fuel sheathing (13.08 mm outside diameter) with brazed-on appendages.
- Capillary tube connected to internal void volume of each fuel element to permit on-line monitoring of the internal gas pressure during the irradiation.

(b) Up to seven CANDU fuel elements in preparation:
- To study fuel behaviour and fission product activity release under severe fuel damage conditions

Table 3
Severe Fuel Damage Test Matrix for the Blowdown Test Facility

<table>
<thead>
<tr>
<th>Test Objectives</th>
<th>Test Date</th>
<th>Target Temp. (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measure fission product release (FPR) in the diffusion-dominant temperature regime. Slow cool to preserve the fission product deposition and fuel geometry.</td>
<td>1989 Dec.</td>
<td>1550</td>
</tr>
<tr>
<td>Measure FPR in the UO₂ grain growth-dominant temperature regime: slow cool.</td>
<td>1990 Nov.</td>
<td>1800 - 2100</td>
</tr>
<tr>
<td>As for test #2 but with a rapid quench to assess the effect of thermal shock on FPR.</td>
<td>1991</td>
<td>1800 - 2100</td>
</tr>
<tr>
<td>Measure FPR at temperatures where significant liquefaction of the ceramic fuel is expected.</td>
<td>1991</td>
<td>2100 - 2500</td>
</tr>
</tbody>
</table>
SUMMARY AND CONCLUSIONS

Features of the NRU Blowdown Test Facility (BTF) and of the commissioning program for the BTF were discussed. Results from laboratory tests on the two reactor test sections were presented and indicated that these seals are expected to operate satisfactorily under high temperature, in-reactor conditions. The in-reactor commissioning experiment, Exp-BTF-10006, with an instrumented fuel assembly simulator in the test section, was completed in 1989. Preliminary analysis of the data indicated that the seals operated satisfactorily in a neutron flux environment and after a number of successfully conducted, high temperature, coolant depressurization transients within the BTF reactor test section. The facility is ready for commissioning with a UO2-fueled, instrumented, experimental assembly (Exp-BTF-102). A single LOCA test and four SFD tests for CANDU-designed fuel are scheduled for BTF over the next three years.

ACKNOWLEDGEMENTS

The authors would like to greatfully acknowledge the contributions from the many people involved with the design, construction, project management, safety analysis, and data acquisition for the Blowdown Test Facility and the BTF Commissioning Program, including: C.J. Alleway, D.A. Barrington, A. Beauchamp, A. Boudens, L.R. Bourque, M.A. Branecki, J.S. Burger, R.D. Delaney, H. Duff, P.J. Fehrenbach, H. Graper, G.O. Hooper, M.G. Jonckheere, N.A. Keller, R.J. Klock, G. Kyle, N.H. MacDonald, K.M. MacFarlane, J.P. Murphy, M. O'Kane, E.P. Penswick, N.B. Poulsen, W.R. Richmond, F. Santone, S.K. Schisski, D.S. Shields, P.J. Valliant, B. Wilder, Y.F. Woo, and R. Wray.

Development and testing of the reactor test section seals used for the Blowdown Test Facility were funded, in part, by the CANDU Owner's Group.

REFERENCES


A SYSTEM TO EXTEND THE ASSESSMENT OF THERMAL FUEL BEHAVIOUR IN THE BLOWDOWN TEST FACILITY

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ABSTRACT

To assess transient fuel behaviour during high-temperature, in-reactor experiments in the Blowdown Test Facility (BTF) and to relate this behaviour to the timing and extent of fission product release, measurements of fuel and sheath temperature during the tests are required. At the upper end of the temperature range (1100°C to melting), the continued integrity of the thermocouples cannot be assured.

A microcomputer-based System to Extend the Assessment of Thermal Fuel Behaviour (SEATFB) in the BTF is described in which measured temperatures in the lower temperature region, coupled with a simple fuel model, enables the continued on-line estimation and visualization of fuel temperature distributions in the BTF assembly.

INTRODUCTION

In order to assess transient fuel behaviour during high-temperature, in-reactor experiments in the Blowdown Test Facility (BTF) (1), and to relate this behaviour to the timing and extent of fission product release, measurements of fuel and sheath temperatures during the tests are required. However, the scope of the BTF tests includes sheath temperatures in the range of 1100°C to melting, fuel temperatures in the range of 1100°C to shear melting (ie. approximately 1850°C for oxygenated Zircaloy), and fuel temperatures in the range 1300°C to 2500°C. At the high end of these temperature ranges, the continued integrity of the thermocouples (T/Cs) used to measure these temperatures cannot be assured for the duration of the test (ie. on the order of 100's of seconds at maximum temperature).

A System to Extend the Assessment of Thermal Fuel Behaviour (SEATFB) in the BTF is described which, after failure of the T/Cs in the higher temperature region of a fuel element, would enable the continued, on-line estimation of those temperatures based on:

1) the measured temperatures in the lower temperature regions, where T/C survival is more probable, and/or

2) a fuel model which provides a progressive estimate of the thermal and mechanical condition of the fuel elements as the test is performed.

The SEATFB can be used to enhance the assessment of the axial and circumferential temperature gradients which can occur on the sheath and in the UO2. These gradients can have a significant effect on the development of sheath strain and on the timing and extent of fission product release. As it is impractical to instrument the assembly with enough T/Cs to provide direct, detailed measurement of these gradients, an alternate approach, using a combination of actual and "virtual" T/C data, is adopted.

Virtual temperatures are those temperatures at various locations where actual T/Cs are not located that are estimated using relationships established between the thermal behaviour at actual T/C sites and the selected sites of virtual T/Cs. An example of these two types of T/C sites is shown in Figure 1.

As actual test measurements are used in evaluating virtual temperatures, the virtual T/C model provides an on-line recalibration for the virtual temperatures. In conjunction with these test and virtual temperatures, a fuel rod thermal model provides estimates of radial temperature profiles through each fuel element at each thermocouple plane.

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FIGURE 1: ACTUAL (●) AND VIRTUAL (○) T/C LOCATIONS AT AXIAL PLANE B1 IN BTF-103.

As T/Cs begin to fail during the test, more indirect relationships are used in the virtual T/C model to estimate virtual temperatures. As the number of failed T/Cs increases, the confidence in the derived values decreases. This is indicated on the colour graphic display by altering the colour used to display that value (see Figure 2). This stage of the test also involves the on-line development of a fuel thermal model. Temperature measurements and virtual temperatures are used to continually update, in real time, the thermal model of the test fuel as the experiment proceeds. At any time, the thermal model can provide an estimate of sheath-to-coolant heat transfer, sheath oxide thickness, and fuel and sheath temperatures as a function of the axial and circumferential location on a test element.

A third information source for test assembly behaviour is the expected behaviour and expected consistency between different components in the BTF test assembly. For example, during single phase cooling, most T/Cs should display similar temperature/time profiles. When most of the test assembly instrumentation has failed, models, trends, and subjective relationships such as this are the only means of assessing test assembly temperatures.

Inconsistencies between the on-line T/C data, virtual T/C model, and the expected performance need to be resolved. This resolution may:

i) require the recalibration of the models used, or
ii) indicate when instrument degradation has begun.

SYSTEM DESCRIPTION

Overall Structure

SEATFB is being introduced into the BTF program in phases. Therefore, the overall system will be described in terms of its component parts. These component parts include models and methods to enhance the:

i) visualization of fuel thermal behaviour,
ii) evaluation of fuel element circumferential and axial temperature gradients, and
iii) estimation of fuel temperatures in regions that have experienced thermocouple failure.(9)

As will be discussed in the following section, these models and methods will be useful to enhance the understanding of fuel thermal behaviour prior to, during and after a test in BTF.

How SEATFB works during a test in conjunction with the existing data acquisition system is shown schematically in Figure 3.

Selected fuel, sheath, coolant and thermal shroud temperatures are stored in the SEATFB microcomputer. If all the thermocouples are operational then the virtual temperature method may be used to estimate the circumferential temperature gradients at all thermocouple plane locations. The measured and virtual temperatures are also used to estimate the fuel temperature distribution of the test elements via a component of the online, fuel thermal behaviour model. Any or all of this information is then displayed in a manner that will enhance the visualization of the fuel thermal behaviour during a test.

If thermocouples start to fail, the historical trend data can be used to estimate those temperatures from other operative T/Cs. If many of the thermocouples fail then the predictions of the on-line, fuel thermal behaviour model may provide a better estimate of the fuel temperatures.

Figure 2. AN EXAMPLE OF THE COLOUR GRAPHIC DISPLAY FOR THE ENHANCED VISUALIZATION OF THERMAL FUEL BEHAVIOUR IN BTF.
The SEATFB in the BTF can be utilized prior to a test to assist in planning, during a test to assist in on-line interpretation of the test results, and after test to assist in post-test interpretation of the test results.

Prior to a test, the SEATFB can provide a structured method to:

i) judge whether the nature and extent of instrumentation is adequate to ensure that the experimental objectives are attained,

ii) assess the range of expected fuel and instrument response via the results of parametric studies, and

iii) confirm which measurements, and therefore which instrumentation, are key to the interpretation of the test results.

During the test, the SEATFB could:

i) enhance on-line visualization and understanding of the test results,

ii) provide a continued estimate of fuel thermal behaviour in areas where instrumentation has failed, and

iii) ensure the potential exists to continue a test even though much of the fuel assembly instrumentation may have failed.

After a test, the SEATFB in the BTF could:

i) provide enhanced visualization of the measured fuel thermal behaviour,

ii) assist in the qualification of thermocouple measurements, and

iii) assist in the explanation of anomalies in the thermocouple data.

Input to SEATFB for pre-test use employs pseudo-test results of temperatures in the range of interest evaluated from simulations or prior test results. During the test, input to SEATFB consists of on-line T/C data. Post-test usage would rely on qualified T/C data input and the results of bench-marked simulations.

**Virtual Temperature Evaluation**

The equations for the virtual temperatures are calculated using test assembly symmetry in conjunction with available T/C measurements. There are two types of virtual temperature equations that can be developed taking advantage of the 3-fold symmetry of the BTF trefoil test assembly. The first is a set of planar equations that are calculated at each of the three specific T/C planes (B1, Y/Z and X1). The second is a set of axial equations that are calculated along each of the three guide tubes in the test assembly by relating values at corresponding planes.

**Methodology to Establish Planar Equations**

The planar equations are further subdivided into two types:

i) using trefoil T/C temperatures exclusively to determine virtual temperatures, and

ii) using the trefoil temperatures and measurements of either the rib T/Cs or the steam temperature at the center of the trefoil.

Each of the 3-fold-symmetry, trefoil equations requires any combination of three thermocouple or virtual temperatures from one fuel element and the temperatures from the corresponding locations of another fuel element. For example, suppose that three temperatures at a given plane on element 1 are designated as \(a_1, b_1, \) and \(c_1,\) with their counterparts on element 2 designated as \(a_2, b_2,\) and \(c_2.\) Each equation is then written in the form:

\[
(a_1 - b_1) = (a_2 - b_2) \left(\frac{c_1}{c_2}\right)
\]

The two corresponding pairs, \((a_1 - b_1)\) and \((a_2 - b_2),\) relate the circumferential gradient between fuel elements 1 and 2. The ratio, \(c_1/c_2,\) effectively normalizes the results in the event that the fuel element average temperatures differ.

For the second type of planar equation involving the temperature of rib (appendage) T/Cs, three temperatures in total are used per equation. The primary assumption is that an increase in rib temperature is due to radiative heat transfer from the two closest fuel elements. The rate of increase of the rib temperature varies linearly with the heat transferred to it or

\[d(T)/dt = C_1 Q = C_2 \left( (T_1^4 - T_4^4) + (T_2^4 - T_4^4) \right)\]

where \(Q\) is the heat transferred to the rib, \(C_1\) and \(C_2\) are constants of proportionality, and \(T_1\) and \(T_2\) are the temperatures of the two closest fuel elements. Since radiant heat transfer is only dominant for temperatures above 1000°C, the above equation would only be valid above this temperature.
For the second type of planar equation involving the temperature of the steam at the center of the refloof, four temperatures are used per equation. The main assumption is that an increase in the steam temperature is due to convective heat transfer from the three closest points on each of the fuel elements. The governing equation becomes

\[
\frac{dT}{dt} = C_1 Q = C_2 [(T_1 - T) + (T_2 - T) + (T_3 - T)]
\]

where \(C_2\) is a proportionality constant and the \(T_i\) are the temperatures of the three closest points.

As the temperatures are likely to be somewhat "noisy", the above time derivatives may have to be filtered by averaging over several time steps.

Methodology to Establish Axial Equations. Each of the axial equations requires any combination of two thermocouples or virtual temperatures from one plane of a given fuel element and the temperatures from the corresponding locations on another plane from the same element. Each equation is then written in the form:

\[
(T_1^x - T_2^x) = (T_1^y - T_2^y)
\]

where \(x\) and \(y\) designate the different element planes.

On-Line Fuel Thermal Behaviour Model

A schematic of the development and use of an on-line fuel thermal behaviour model in the BTF is shown in Figure 4. While most of the T/Cs on the fuel assembly are operational, the measurements are used in conjunction with the linear power and coolant flow to estimate:

i) the rate and extent of the Zircaloy/steam reaction (ZSR) as a function of the location on the fuel sheaths of the test elements,

ii) the distribution of oxide thickness,

iii) the heat generated due to the exothermic ZSR,

iv) the fuel temperature distribution,

v) the rate of heat removal by convection and radiation, and

vi) the rate at which additional heat is stored in the fuel and sheath material.

The fuel rod model is a one-dimensional solution of the transient heat conduction equation using five fuel annuli, a fuel/sheath gap, and a single sheath annulus. Temperatures are evaluated at the surface of each annulus. The solution procedure employs Gaussian elimination and a Crank-Nicholson convergence scheme. Thermal properties are functions of local temperature. Zircaloy oxidation kinetics and ZSR heat production are calculated using the Urbanic-Hiedrick equations (2).

The total heat produced in a time period (the sum of the heat produced in the fuel and the exothermic ZSR) is equal to the total heat removed due to convection and radiation and the additional heat stored in the fuel and sheath. During the period when most of the fuel assembly instrumentation is operational, measurements are used to estimate the ratio of the heat added to heat removed. To obtain this ratio as a function of time, it is assumed that for the higher temperature tests in the BTF, where extensive fuel assembly instrumentation failure may occur before the test target temperature is attained, that the test procedure will involve maintaining a constant, low inlet flow and incrementing the element power in a step-wise fashion.

As the extent of instrumentation failure increases, the estimates of temperatures using the methodology described becomes more and more uncertain. At this time, the on-line fuel thermal model can be used to estimate fuel and sheath temperatures assuming the flow is still measured and that the element linear power is known. If the element linear power can no longer be calculated from measurements of inlet and outlet coolant temperatures due to T/C failure, then a previously established calibration curve of the element power to the reactor power level can be used to estimate fuel element power. The temperature estimates from the on-line fuel thermal model are then compared and rationalized to those obtained using other information sources. The best estimates of the temperatures are then used to provide an on-going assessment of the fuel temperature distribution, the ratio of additional heat stored to the heat removed, and the distribution of the rate and extent of the ZSR. Some iteration between the estimates of these parameters and the temperatures may be required.
The proposed implementation schedule for the SEATFB in the BTF is shown in Figure 5. The component which uses colour graphics to enhance visualization of the fuel thermal behaviour will be ready for use in the BTF-102 trefoil test. The assessment of virtual temperatures and the extended temperature assessment methodology will be available for the BTF-102 and the BTF-103 trefoil tests. For the BTF-104 test and all subsequent tests, the full SEATFB system would be available.

<table>
<thead>
<tr>
<th>INTRODUCED FOR:</th>
<th>SEATFB CAPABILITY</th>
</tr>
</thead>
<tbody>
<tr>
<td>BTF-102</td>
<td>Enhanced visualization of fuel thermal behaviour</td>
</tr>
<tr>
<td>BTF-103</td>
<td>Extended assessment of downstream thermal behaviour of previously irradiated fuel element via virtual temperature method</td>
</tr>
<tr>
<td>BTF-104</td>
<td>Integrated evaluation of test element behaviour - virtual temperatures - parallel element and shroud thermal model</td>
</tr>
</tbody>
</table>

**REFERENCES**


Session 16:

CANDU Operating Experience II

Chairman:

V. Austman, Ontario Hydro, Darlington
Ontario Hydro Bruce Nuclear Generating Station B

ABSTRACT

Bruce Nuclear Generating Station "B" (NHG by "B") consists of 4 - 930 MW Candu units (Units 5 to 8). The station, located on the shores of Lake Huron, is part of the Ontario Hydro Bruce Nuclear Power Development.

This paper describes the bleed valve body wall loss discovered during operation and the subsequent efforts to correct the problem. The type and extent of damage is outlined as well as the repair method used. To ensure reliable, long term operation of the valves, a rigorous investigation was undertaken to identify the failure mechanism. Once a failure mode was established, long term solutions could be implemented to ensure valve integrity.

EVENT

The following event involved the Primary Heat Transport (PHT) System bleed valves. These 2 - 100% 2" control valves (C.V.s) form an integral part of the PHT continuous feed and bleed control system. This system itself is part of the PHT pressure and inventory control scheme. It also provides a continuous flow of Heavy Water (D_2O) through the purification system for chemistry control. Figure 1 is a simplified flow diagram of the feed and bleed system and illustrates the role of CV5 and CV6.

On October 29, 1988 unit 5 tripped due to a fault on the main transformer lightning arrester. During a routine vault survey, an operator reported seeing water dripping from the body of a PHT bleed valve. Analysis of the water confirmed it was D_2O. It was decided to perform radiography (R.T.) on the suspect valve (CV6) as well as its partner (CV5). The results revealed a pin hole leak in CV6 and up to 40% wall loss in CV5. Figures 2 and 3 represent the radiographic cross section through the valves and show the extent of erosion found.
Since a generic problem was suspected and unit 8 was shut down at the time, radiography was performed on the valves in this unit as well. The results revealed that CV5 was intact with no damage, while CV6 had up to 50% localized wall loss. Because of the redundancy offered with 2 valves, approval to run the unit on single valve operation was sought. Bruce Design confirmed that single valve operation was acceptable from a safety and reliability standpoint and the Atomic Energy Control Board (AECB) authorized unit start-up in this mode. The damaged valve (CV6) was removed from the system and the open piping capped. Unit 8 returned to power on November 7, 1988 on single bleed valve operation.

SEEKING SOLUTIONS

Since valve body damage was found on both bleed CVs in Unit 5, start-up was delayed until the valves were either replaced or repaired. These particular valves are nuclear class 1 components. This classification signifies that a failure of the system or component, in the absence of Safety System action, may result in radiological hazard to the public in excess of authorized limits. Therefore, in order to verify the integrity of the valves for this important application, a detailed history docket (pedigree) must accompany each component. This documentation outlines the material traceability, fabrication processes used and testing the valves underwent to meet code requirements.

Since replacement energy costs exceeded $10K/hour for thermal generation, valve replacement was the most desirable option in order to expedite a solution and allow the earliest unit startup. However, because spare valves did not exist for these components, Bruce Engineering Department (BED) were informed. They in turn contracted Atomic Energy of Canada Limited (AECL) to assist in resolving the problem. Coordinated efforts were put forth in parallel to:

1. Search for fully documented replacement valves
2. Establish a valve repair procedure.

This included a search for fully documented, similar material with which to perform repairs

Replacement Valves

An exhaustive search was undertaken to identify if replacement valve bodies existed at other nuclear facilities or manufacturers. Replacement valves could only be considered if they met the following constraints:

1. Correct size and flow characteristics
2. Full documentation for traceability
3. Correct specifications for various operating and code requirements

The purchase of new valve bodies was rejected due to the 12 to 15 month delivery time quoted. Although several similar valves were in fact shipped to the station from Darlington NGS, extensive modifications would have been required to make them compatible with the existing design installation. In addition, similar valves were removed from Douglas Point NGS which has been taken out of service. Although these CVs were similar to the problem CVs, the documentation on the valves could not be located. For this reason the valves were rejected. It became apparent that replacement of the valves was not a viable short term option.

Valve Weld Repair

Immediately after the CV erosion was identified discussions on valve weld repairs were initiated. Concerns and questions on the details of repairs had to be addressed before any procedures were attempted. Factors to consider included:

1. Acceptability of repairs by Regulatory bodies
2. Temporary vs permanent repair methods
3. Practicality of repairs
4. Details of repair procedures including non-destructive examinations (N.D.E.)

After several days of discussions and research, approval was received from B.E.D. to initiate activities on a permanent weld repair procedure. Regulatory authorities were informed that a welded flush patch repair was to be performed on all damaged valve bodies. A welding engineer specialist was employed to prepare the detailed procedures to be used for the repair process. He was experienced in both the technical and practical requirements of this type of procedure. In addition, BNGS"B" welders were consulted to determine the practicality of the job in terms of welding dynamics and accessibility on such a small component. Agreement was reached that weld repairs should proceed since the desired repair would meet code requirements and confidence was expressed in the ability of the trades people to perform the work including the necessary Quality Control (Q.C.) requirements. Concurrence from the Ministry of Consumer and Corporate Relations (M.C.C.R.) and the A.E.C.B. was received for the repair procedure.

Acquisition of Repair Material. Before any weld repairs could be attempted, suitable material had to be procured for the task. Acceptability of the material depended upon the assurance of:

1. Proper geometry in terms of thickness and curvature
2. Compatibility of material
3. Nuclear classification which included complete documentation of material for traceability
To meet these requirements the material had to be nuclear class 1, stainless steel (304L or 316). It had to be supported by full documentation, equal or exceed 3/4 inches in wall thickness and be able to be machined to conform to the curvature of the damaged valve body.

The search for such material began on site with Construction and quickly fanned out to all Candu stations in Canada as well as manufacturers who held licences for the fabrication of nuclear components. Eventually a valve bellows spool piece extension was purchased from a valve manufacturer which either met or could be modified to meet all the requirements imposed. Similar problems had to be overcome in acquiring other more common nuclear class items such as nuts, bolts, pipe caps, channel iron etc.

**Valve Repair Procedures**

**Scope of Work**

Analyses of damaged CVs indicated that most of the wall loss occurred on roughly a 2-1/2" diameter area at the bottom of the valve body directly below the seat. The objective of the repair was to remove the majority of the damaged area and weld in a matching patch. Any additional damage was to be repaired with weld overlay involving the deposition of weld metal. All phases of the repair were subject to N.D.E.

**Installation of Flush Patch**

To facilitate the repairs it was decided to bore out the damaged area of the valve via a 2-1/2" circular hole. The surfaces of the cut were then bevelled for weld preparation and a liquid penetrant examination performed (P.T.). Several flush patches were then machined from the forged valve bellows extension piece acquired for this purpose. The inside and outside patch curvatures were confirmed to mate with the valve body. Once again bevels were ground and P.T. performed on the cut surfaces. A mock-up was fabricated and a test patch welded in place to verify that the repair was achievable and to provide experience for the welders. After a successful mock-up demonstration a patch was then fitted to the valve body and tack welded in place. Once welding was completed the outer and inner surfaces were ground to blend with the original valve surface. P.T. and R.T. examinations were performed on the repair. Figure 4 illustrates the repair procedure followed and appendix 1 outlines the specific details of the procedure used by maintenance personnel.

**Weld Overlay**

Although most of the valve wall damage occurred in a small area where fluid impinges on the bottom of the downstream cavity, there was minor pitting over a wider area as well as beneath the seat. Before the valve could be reinstalled this damage had to be removed and the inner surfaces restored to their original condition. The major obstacle in the restoration was the limited access to the damaged area. Any grinding and welding would have to be performed with access through the valve seat with the valve bonnet removed. The procedure, as outlined in appendix 2, involved grinding of the damaged areas with a pencil grinder and then weld build up of the prepared surface. Weld deposition was performed and a minimum wall thickness of 0.5" over the entire valve body was confirmed. When sufficient weld had been deposited the repaired area was ground smooth to blend with adjacent material. A P.T. was performed on the repair.

**Valve Removal/Installation**

The bleed CVs are located in a congested area providing difficult access. Additionally, the valves are housed in a radiological environment within the reactor vault necessitating the use of plastic suits with external air supplies. These factors would make any in-situ work difficult and dose intensive to perform. With this in mind the overriding desire was to minimize the amount of work done in the field. This was accomplished with some success by several steps:

1. Erection of an enclosure around the work site with dedicated ventilation
2. Cutting CVs out of system at a more convenient accessible location
3. Performing more rigorous transition and stainless steel welds in the shop
4. Assigning dedicated personnel to job
5. Rehearsing details of work on a mock-up
6. Establishing detailed logic for control and mechanical maintenance interactions
Some of these provisions were not identified or implemented until after Unit 5 work was complete. Dose reductions of up to 500% were realized in subsequent units. Much of this can be attributed to the experience gained from the initial efforts in Unit 5 and the subsequent actions taken to improve the task.

ANALYSIS AND FAILURE MECHANISM

In order to perform an analysis of this problem several pieces of information were examined. These included:

1. Exact nature and extent of damage on each valve in each unit
2. Operating history of CVs
3. Design review of operating characteristics based on valve specifics and fluid parameters

Ideally all this information should have led to the root cause of the wall loss. Only then could measures be taken with confidence in providing a permanent solution to the problem.

Nature of Damage

Although the extent of wall loss varied from valve to valve and unit to unit, one common factor was the type of damage found. It consisted of densely populated surface pits of various sizes and had a honeycomb-like appearance. The primary damage area was roughly a 2 to 2-1/2 square inch area which look the shape of a circular band at the bottom of the downstream valve cavity. Minor, secondary damage was found under the seat of some CVs as well as slight pitting on the valve walls outside the major wear area. Realizing that more than one phenomenon may be occurring the analysis focused on the severe damage found. The minor damage did not jeopardize the integrity of the valves.

Operating History

In order to determine if the problem occurred at specific times in the valves life, a close look into the operating history was undertaken. All valves were initially supplied with the anti-cavitation Cavitrol III trim (figure 5). However, because of the small port openings in the trim, severe valve plugging with magnetite occurred early in performance commissioning. The manufacturer agreed to provide a replacement commissioning trim for short term operation. The large ports in this trim would minimize plugging. As can be seen in table 1 the number of operating days using this trim varied greatly from unit to unit. In fact because of insufficient valve bonnets required for the Cavitrol III trim housing, Unit 5, CV6 still contained the commissioning trim at the time the leak was found. Also in table 1 are the unit total operating hours for each valve. The majority of valve operation has occurred at high power under normal operating conditions. Although the valves are control valves they are nominally open about 25% to 40% to achieve the desired feed and bleed flow.

<table>
<thead>
<tr>
<th>Table 1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unit 6</td>
</tr>
<tr>
<td>CV5</td>
</tr>
<tr>
<td>Days on Commissioning Trim</td>
</tr>
<tr>
<td>Damage - % through wall</td>
</tr>
<tr>
<td>Operating Hours</td>
</tr>
</tbody>
</table>

FIGURE 5 - TWO STAGE CAVITROL TRIM

Operating Characteristics

Normal bleed valve operation occurs with a large pressure drop across the valve. The upstream conditions are 10.3 MPa at 265°C. The downstream conditions are 1.62 MPa at about 200°C (saturated conditions). Two phase flow will exist to some degree on the discharge side of the valve. The valves operate in the flow to close direction to suit the 2-stage Cavitrol III trim. This trim was provided to control cavitation during low temperature operation. Low temperature operation would occur infrequently during unit cooldowns and warm-ups.

When looking at the operating characteristics of CVs at BNGS "B", an obvious progression was to look at similar data for the same valves at BNGS "A". The CVs at "A" plant had been in-service much longer under almost identical conditions and no problems had been experienced. Major differences do exist however in CV design. The BNGS "B" valves are standard globe valves with no Cavitrol III trim and operate in the flow to open mode. In this mode the discharge fluid tends to disperse after leaving the seat and impinges on the valve body at an angle on the neck. The BNGS "B" CVs discharge through a 1" diameter seat and impact directly on the bottom of the valve body at 90° to the discharge flow. The initial reaction was to conclude that the CVs were incorrectly designed and installed at BNGS "B" based on the configuration at BNGS "A". Further investigation revealed that slightly different valve specifications accounted for the difference. It would have been very easy to leap to an "instant solution" based on the early findings.
Failure Mechanism

Four probable failure mechanisms were identified:

1. Solid particle impingement
2. Cavitation
3. High velocity flow erosion
4. Wet steam impingement

The nature of the damage (pitting) was not indicative of an erosion type of wear pattern. Since smoother wear contours would have been expected with erosion the mechanism was rejected. The flow discharge pattern of the commissioning trim was analyzed. It can be visualized as a cylinder expanding to a hollow conical section with higher velocities on the outside. Any water droplets or solid particles would impinge directly on the valve body. If the damage was caused by this mechanism it could be expected to be a function of operating time with this trim and it should exhibit a circular wear pattern with deeper wall loss at the outside diameter. This was indeed confirmed by radiographic and borescopic examinations and by the fact that CV6 in Unit 5, which operated longest with the commissioning trim, was the valve to develop a throughwall leak. This is illustrated in fig. 6. Under cold operation with the commissioning trim some cavitation damage is inevitable and may explain the minor damage areas on the sides of the outlet chamber.

Although the Cavitrol III trim operating conditions are the same, the characteristics of the trim are different. The high velocity 2 phase flow dissipates much of its energy as it is directed through the ports towards the centre line of the valve. The flow then turns 90° and takes the form of a solid cone as compared to the hollow cone pattern with the commissioning trim. The flow is dispersed over an approximately 20 times larger area minimizing impingement concentration. In addition the trim provides low temperature cavitation protection and is suitable for normal operating conditions. For these reasons the majority of the damage is believed to be caused by wet steam impingement during commissioning trim operation. Local cavitation under cold conditions and possibly some particle impingement might also have contributed to wear during operation with the commissioning trim.

CONCLUSIONS

A review of the valve design together with operating records indicates that damage to the valve bodies was caused during operation with the commissioning trim. The existing Cavitrol III trim is expected to be satisfactory for the range of operating conditions over the long term. All valves in all units have been inspected and/or repaired and are now fitted with the Cavitrol III trim. Neither technical nor economic reasons could be found to justify a change to a different design. Reassessment of the valve operating characteristics has since resulted in resumption of parallel operation of the CVs.

As a result of the incident and the obstacles encountered in dealing with this problem, a number of actions/recommendations are in progress or planned:

1. A number of replacement valve bodies have been ordered as a precautionary measure to replace repaired bodies if necessary.
2. An inspection program will ensure at least 1 unit per year will have its CVs inspected for damage.
3. A spare parts task force has been organized to look into the problem of acquiring nuclear class spare parts and materials. Retention of exotic materials for repairs and parts fabrication is essential.
4. A welding engineer specialist has been assigned to site and will provide immediate expertise for similar problems.
5. Bruce Design is investigating the necessity of acoustic monitoring techniques and additional inspections to track the scope and status of the CV problem.
6. Increase the overlap among the various Design, Research, Technical and Maintenance groups. Immediate interaction between all groups is essential.

REFERENCES

WELD REPAIR OUTLINE PROCEDURE

1. Remove valve seat as required for access.
2. Survey valve interior with borescope & sketch damage area.
3. Machine opening in body spanning damage area. The MAXIMUM hole size to be drilled prior to Tech Unit inspection is 2.5".
4. Visual & thickness check to confirm damage removal. Rectify as necessary by i) machine larger opening; ii) grinding; or iii) weld reinforcement.
5. Grind or machine weld preparation bevels on valve body.
7. PT valve weld preparation.
10. Fit purge dams to inlet, outlet & bonnet openings.
11. Purge interior with argon per WPS.
12. Manual GTAW root pass per WPS.
13. PT root pass. Borescope inspect interior. Rectify as necessary by local grinding with pencil grinder or removing patch.
15. Grind blend OD & ID surfaces.
16. Borescope ID
   PT OD
   RT repair weld repair.
17. Perform hydrostatic test on valve at 17.2 MPa.

WELD OVERLAY OF CV VALVE BODY

1. Determine area requiring repair (min thickness 0.5 inch).
2. Grind repair area to sound metal, removing all oxides, pitting, etc.
3. Degrease surface using approved solvent.
4. Weld deposit using WPS P-155 with K316L-16 SHAW electrodes and/or KX316L GTAW filler rod.
5. Remove slag between each pass with chipping hammer, wire brush etc.
6. When sufficient metal has been deposited, grind weld surface smooth, and blend with adjacent material.
7. PT weld deposit.
HEAVY WATER LEAK DUE TO FRETING OF DN TUBE

JONG - WON PARK

Wolsong Nuclear Power Plant
Korea Electric Power Corporation

ABSTRACT

Wolsong nuclear power plant has experienced four occasions of reactor shutdown owing to heavy water leaks since its commercial operation. Among these heavy water leaks, only one case was acute and brought about reactor shutdown but the other cases listed below were chronic and repaired after manual reactor shutdown.

<table>
<thead>
<tr>
<th>LEAK POINT</th>
<th>DATE OF EVENT</th>
<th>OUTAGE PERIOD</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sampling Tube</td>
<td>86-10-24</td>
<td>86-11-15 03:14 ~ 86-11-19 10:13 (4 Days)</td>
</tr>
<tr>
<td>of DN Monitoring System</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(&quot;J-11&quot; CHANNEL)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sampling Tube</td>
<td>88-08-13</td>
<td>88-08-16 01:35 ~ 88-08-19 11:01 (3 Days)</td>
</tr>
<tr>
<td>of DN Monitoring System</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(&quot;W-11&quot; CHANNEL)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HT Low AP Sensing Tube</td>
<td>88-09-05</td>
<td>88-09-09 23:30 ~ 88-09-11 21:00 (2 Days)</td>
</tr>
<tr>
<td>(68334-9H)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

All these three events were quite similar to each other. Heavy water escaped through a pinhole which was produced at the outer surface of the tube. The pinhole was caused by wear attributed to vibration and abrasion against the other tube or the clamp contacted. The differences are location, leak rate and values in heavy water loss and tritium level etc.

This paper presents a brief description on these tube leaks including symptoms, causes, effects, corrective actions and lessons learned, principally concerning the second DN tube leak which has occurred on August 13, 1988.

EVENT DESCRIPTION

Introduction

The Delayed Neutron(DN) Monitoring System is put into operation when a fuel failure is suspected.

As shown in figure 1, this system mainly consists of BF3 counters, light-water filled moderator tanks and 380 sampling lines.

![Figure 1. The Delayed Neutron Monitors](image)

Neutrons from the decay of fission products, primarily I-137 and Br-87, are slowed down in the moderator tank and detected by a BF3 counter. Sample lines from each outlet feeder in both heat transport system leek' convey coolant samples to two accessible activity monitoring rooms in the reactor building.

The sample line is seamless tube and made of TP 304 L(Nuclear Class I) stainless steel and its size is 1/4 inch in diameter and 0.049 inch in thickness.

Figure 2 shows the schematic sample flow diagram of the DN monitoring system.
Followings were presented as evidences of safety.
- There was almost no change in level of all heavy water tanks.
- The tritium concentration in air of the reactor building seemed to be saturated.
- Tritium release to environment was about 200 Ci per day, much lower than 1 % DEL of 300 Ci per day.
- Past experience; the tritium concentration reached to 65 MPCA at the highest when there was heavy water leak through a small pinhole on the DN tube in November 1986.

We decided to continue to operate the plant, maintaining the reactor power at 100 % full power and collecting enough data necessary to locate the exact leak point.

At 14:00, on August 14, after lots of data were collected and analyzed we could suspect that heavy water leaks would be C-side reactor vault(R-107) and the most probable leak point would be the DN tube.

At 16:30, two and half hours later, an operator managed to access to the vicinity of the suspected area and found D2O vapour being blown.

It took only less than 30 hours from the time when we found the first symptom of heavy water leak to the time when we found the exact leak point located in the inaccessible area during power operation. It was very noticeable to locate a pinhole in such a short period. Therefore I would like to introduce the most important data which led to the successful result.

1) Tritium Concentration in Various Locations

We took a lot of air sample in various locations including inaccessible area during normal operation.

* Sampling points were located at every elevation of the reactor building including normally inaccessible area such as feeder cabinets and reactor vault areas. Sampling frequency was every 6 hours generally, but samples were taken at some other places and more frequently as the tritium level varied.

Figure 3 shows the trend of tritium concentration in air of the relatively important areas.
We drew out two general principles from the data:

- The tritium concentration of C-side areas are higher than those of A-side areas.
- Among all the areas, the C-side reactor area (R-107) has the highest tritium concentration in air.

2) Heavy Water Recovery

There was no level increase in the liquid recovery tank 3333-Tk3 and sumps in the reactor building. Also we couldn't find any trace of heavy water in the active drainage.

But there was quite a change in D2O vapour recovery tank level.

First of all, the D2O isotopic of water in the vapour recovery water tank, 3831-Tk2, was increased from 1.2% to 5.4% D2O.

Particularly the D2O content in the recovered water taken at the downstream of 3831-DR2 and -DR11 was approximately 20%. Also their tritium concentration were about 0.39 Ci/Kg-D2O and coincided with that of PHT heavy water. At that time, the tritium concentration of moderator D2O was 23 Ci/Kg-D2O.

As well as the D2O isotopic, the heavy water quantity recovered by vapour recovery system was increased sharply.

Table 1 shows the data of heavy water recovery.

<table>
<thead>
<tr>
<th>DATE</th>
<th>Vapour recovery</th>
<th>LIQUID</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>PHT D2O</td>
<td>MOD D2O</td>
<td>RECOVERY</td>
</tr>
<tr>
<td>88.08.13</td>
<td>53.4</td>
<td>1.0</td>
<td>0     54.4</td>
</tr>
<tr>
<td>88.08.14</td>
<td>106.1</td>
<td>1.3</td>
<td>0      107.4</td>
</tr>
<tr>
<td>88.08.15</td>
<td>135.3</td>
<td>1.0</td>
<td>0      136.3</td>
</tr>
<tr>
<td>88.08.16</td>
<td>144.6</td>
<td>3.7</td>
<td>0      148.3</td>
</tr>
<tr>
<td>88.08.17</td>
<td>253.2</td>
<td>15.1</td>
<td>9.7    278.0</td>
</tr>
<tr>
<td>88.08.18</td>
<td>130.5</td>
<td>4.7</td>
<td>0      135.2</td>
</tr>
<tr>
<td>88.08.19</td>
<td>65.1</td>
<td>7.6</td>
<td>0      72.7</td>
</tr>
<tr>
<td>88.08.20</td>
<td>28.9</td>
<td>1.0</td>
<td>0      29.9</td>
</tr>
<tr>
<td>88.08.21</td>
<td>33.9</td>
<td>1.5</td>
<td>0      35.4</td>
</tr>
<tr>
<td>88.08.22</td>
<td>24.1</td>
<td>4.9</td>
<td>0      29.0</td>
</tr>
<tr>
<td>SUM</td>
<td>975.1</td>
<td>41.8</td>
<td>9.7   1026.6</td>
</tr>
</tbody>
</table>

* Recovered during PHT resin deuteration process.

From these data we found that:

- The leak source was PHT system.
- Heavy water leaked in vapour form.
- The leak point was located in either reactor vault area or boiler room.

3) Heavy Water Loss and Inventory

The main route through which heavy water escaped from the plant was the main stack.

Table 2 shows heavy water loss per route during the transient period.

<table>
<thead>
<tr>
<th>DATE</th>
<th>via STACK</th>
<th>via ALW</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>88.08.13</td>
<td>67.7</td>
<td>1.5</td>
<td>69.2</td>
</tr>
<tr>
<td>88.08.14</td>
<td>192.4</td>
<td>0.3</td>
<td>192.7</td>
</tr>
<tr>
<td>88.08.15</td>
<td>244.7</td>
<td>2.1</td>
<td>246.8</td>
</tr>
<tr>
<td>88.08.16</td>
<td>332.7</td>
<td>1.4</td>
<td>334.1</td>
</tr>
<tr>
<td>88.08.17</td>
<td>133.4</td>
<td>0.8</td>
<td>134.2</td>
</tr>
<tr>
<td>88.08.18</td>
<td>44.3</td>
<td>1.2</td>
<td>45.5</td>
</tr>
<tr>
<td>88.08.19</td>
<td>30.6</td>
<td>1.1</td>
<td>31.7</td>
</tr>
<tr>
<td>88.08.20</td>
<td>21.4</td>
<td>1.1</td>
<td>22.5</td>
</tr>
<tr>
<td>88.08.21</td>
<td>16.0</td>
<td>0.1</td>
<td>16.1</td>
</tr>
<tr>
<td>SUM</td>
<td>1,083.2</td>
<td>9.6</td>
<td>1,092.8</td>
</tr>
</tbody>
</table>

* ALW : Active Liquid Waste
Figure 4 shows the decreasing trend of heavy water inventory in the main PHT system. The heavy water inventory was calculated every hour by "HDI" program which had been developed by wolsong staff in 1986.

![Graph showing the decreasing trend of heavy water inventory](image)

**Figure 4. D2O Inventory in Main PHT System**

From these data, we could confirm that:
- Heavy water leaked in the vapour form.
- The leak rate was approximately 25 Kg/Hr.
- The leak rate was 27 KG/Hr according to the final assessment of heavy water loss and recovery.

4) Conclusion

Considering the past experiences of heavy water leaks together with all the above analysis results, we made a conclusion:
- The DN tube in C-side reactor vault was the most possible leak point.

**Maintenance**

At 03:15 August 16, the unit was shutdown manually to repair the leak.

After cooling (to 54 °C) and depressurizing (to 2 Bar) the system and a proper isolation of the leak point was made by using ice plug, at 19:30, the defect part of the tube was cut away and then the tube was reconnected by swagelok fittings. At first, welding was considered but rejected because there was no proper purging point and the tube welder did not function precisely high temperature environment.

**The Status of the Defect Tube**

The defect DN tube was T 1/4"-63100-286 and the channel number was W-11.

The tube had relatively severe fret with a small pinhole on the outside surface.

Shapes, conditions and causes of the tube fret was as same as the first DN tube leak case which had occurred on October 24, 1986.

The size of the tube fret was bigger than that of the first defect DN tube. This time, the fretted area was about 40 mm long and 6 mm wide, but the first one covered 14.8 mm by length and 5.7 mm by width.

Photo 1 and 2 shows the outer and inner part of the defect DN tube respectively.

**PHOTO 1. The Fretted Tube**

![Photo of the fretted tube](image)

**PHOTO 2. The pinhole viewed from the Inside**

![Photo of the pinhole](image)

Figure 5 is a sketch of the side view of the tube. It indicates that the wear part is not uniform.

**Figure 5. A Sketch of the Defect Tube**
Causes

Direct Causes. The tube fret and the pinhole was produced due to vibration of the tube against the other 3/8" tube which was in contact with the DN sampling tube by right angle.

Figure 6 shows the general lay-out of the DN tubes, local air coolers and reactor and figure 7 shows the detail leak point.

When the first pinhole was found on the DN sampling tube in 1986, the cause of the defect was assessed into two points as follows.

- Tube fret due to vibration.
- Construction defect; Erroneous cutting by grinder

A few exclaimed that the construction defect would be more realistic because there was no evidence of tube fret on the other tube contacted with the defected one.

But it has been proved that it was wrong by the recurrence of the tube fret.

The location of the defected tube has strong wind of 8 m/sec. The fans in local air coolers which are located just beneath the tube bundles are blowing upwards. Thus the DN tubes installed horizontally are were vibrated up and down severely.

Root Causes. As a result of close inspections, several defects were identified in terms of tube protection.

- Wear rings were missing. They might have not been installed, because there was no trace of coating on the tube.

As you may know very well, wear ring must be installed as a sacrificial protector for the stainless steel tube. Prior to installation of wear rings, thermalox 70 coating shall be taken on the sticking area of wear rings and tubes. The reasons why:
  - Not to miss the wear ring.
  - To protect wear ring itself against corrosion.

* trademark of the Dampney Corporation

- Tubes were not restrained properly.

DN tubes in the reactor vault are disclosed to strong wind by LAC fans. So the wind would cause vibration against some tubes, especially a single tube not fastened to the restraining devices. On the other hand, restrainers were not sufficient to prevent tubes from vibrating.

Effect of the Event

Heavy Water Management Data Summary is shown in the following table. The figure in the table 3 was taken
from data measured from August 13 to 22.

Table 3. Heavy Water Management Data Summary

<table>
<thead>
<tr>
<th>Unit: Kg-D20</th>
</tr>
</thead>
<tbody>
<tr>
<td>RECOVERY</td>
</tr>
<tr>
<td>NORMAL</td>
</tr>
<tr>
<td>ABNORMAL</td>
</tr>
<tr>
<td>LOSS</td>
</tr>
<tr>
<td>NORMAL</td>
</tr>
<tr>
<td>ABNORMAL</td>
</tr>
<tr>
<td>ESCAPE</td>
</tr>
<tr>
<td>TOTAL</td>
</tr>
<tr>
<td>ABNORMAL</td>
</tr>
</tbody>
</table>

Radiation Exposure Summary attributed to the event is shown in the table.

Table 4. Radiation Exposure Summary

<table>
<thead>
<tr>
<th>JOB</th>
<th>MAN POWER</th>
<th>MAN-REM (mRem)</th>
<th>AVG DOSE (mRem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>LEAK SURVEY</td>
<td>24</td>
<td>6,766</td>
<td>282</td>
</tr>
<tr>
<td>RADIATION SURVEY</td>
<td>10</td>
<td>3,193</td>
<td>319</td>
</tr>
<tr>
<td>MAINTENANCE</td>
<td>25</td>
<td>4,006</td>
<td>160</td>
</tr>
<tr>
<td>DECONTAMINATION</td>
<td>7</td>
<td>571</td>
<td>82</td>
</tr>
<tr>
<td>OTHERS</td>
<td>9</td>
<td>646</td>
<td>72</td>
</tr>
<tr>
<td>TOTAL</td>
<td>75</td>
<td>15,189</td>
<td>203</td>
</tr>
</tbody>
</table>

Environmental Effect. The average tritium concentration in air in the vicinity of the plant was 308 pCi/m³. It was quite lower than the maximum permissible concentration of 400,000 pCi/m³ and the effect to environment and resident was negligible.

Brief Description of PHT Low ∆p Sensing Line

Now, I would like to introduce the third tube leak event which has occurred on September 10, 1988. This event was very similar to the DN tube leak events, and so I am going to explain the event very shortly.

The differential pressure between the reactor inlet header (RIH) and the reactor outlet header (ROH) is measured to provide the low differential pressure trip for over power protection. There are four ∆p measurements, two per PHT circuit.

The heavy water leak point was on the sensing line 3/8" 68334-2H between RIH #2 and the transmitter, 68334-PT2H.

Figure 8 shows the flowsheet of the PHT low ∆p sensing line.

CORRECTIVE ACTIONS

Field inspection and reinforcement have been done for DN and FDR cabinet tubes during 1989 annual maintenance outage.
Inspection Period, Manpower and Method

- Period: 1989.02.21~03.09 (Actual working days: 14)
- Manpower: 76 Han-days
- Method: Only visual and tactual inspection method has been adopted. No special equipment except mirror was used.

Findings from field inspection

- There was metallic sound from tube bundle as the result of;
  - Several tubes ran freely without restraints
  - Strong wind from local air cooler had a bad influence on the tube vibration.
- Wear rings were not installed at all the crossed points in DN tubes.
- It was very hard to check and find the tube fret perfectly because the tube bundle was very compact and hardly accessible due to high radiation, high temperature and particularly narrow work space.
- Clamping between tube and feeder pipe were displaced due to thermal expansion.

Reinforcement

First of all, we installed wear rings where they did not exist. And then, we fixed tubes rigidly using bracers, clamps, asbestos tapes.

Following figures show the results of the reinforcement.
RECOMMENDATION

As we have pointed out already in the field inspection result, most of tubes inside Feeder Cabinet can not be accessed, even in shutdown period. And also, maintenance job on the tube was almost impossible. Therefore, proper space for check and maintenance should be secured in the future. Therefore, it is necessary to simplify tube run to reduce the potential problem.

LESSON LEARNED FROM EVENTS

Being placed into commercial operation on April 22, 1983, we have experienced three occasions of reactor shutdown owing to heavy water leak from the tubes.

All these events have occurred between 1986~1988 by the tube fretting wear and all leak points were located in inaccessible area during normal operation. Therefore deep concerns should be raised to this area and regular inspection and reinforcement job should be implemented on the tubes in this area during annual maintenance outage.
ABSTRACT

In nuclear generating plants inverters are required to provide clean uninterruptible AC power to critical loads such as safety systems, computers, and process control. Upon inverter failure, many systems are designed to have a static switch transfer the load, in a no-break manner, to a regulated bypass supply. Failure of the transfer can greatly increase the risk of a plant trip.

Recent failures of both the inverter and transfer equipment warranted a review of system design and maintenance practices. The review findings indicated that, due to age-related degradation, a number of the inverter critical components were not functioning properly. This resulted in the "violent" failure of solid state components within the inverter. Resultant transfers to the bypass supply were not successful because noise-induced spurious operation of logic permissives at that critical time. In order to increase the system reliability, the inverter components and transfer logic has been upgraded along with an improved maintenance program.

INTRODUCTION

At the Point Lepreau Generating Station, the Uninterruptible Power Supply (UPS) system consists of three independent supplies designated Channel A, B and C. Each channel of the 120 VAC UPS system has a 60 kVA static inverter, static bypass voltage regulator, and a static transfer switch. Typical loads on the 120 VAC inverters are station control computers, transmitter power supplies, station instrumentation, and reactor safety shutdown systems controls.

For normal operation, the inverter output is the preferred source. The power supply for each inverter is from a 250 VDC battery and rectifier system. Normal voltages for this system are 264 VDC, with a yearly equalize of 280 VDC. If the inverter fails, the static switch automatically completes a no-break transfer of the load to a 120 VAC regulated bypass supply. The voltage regulator is required to supply critical loads such as computers which cannot tolerate power line voltage variations. The load requirements demand that the total sensing and transfer time be less than 1/4 cycle. The oscillator of the inverter is normally synchronized to the bypass supply so that transfers to bypass are effectively in-phase. The design, as-commissioned, blocked transfers if the bypass was in an over/under voltage condition or the two supplies were not in phase.

Each failure experienced on inverter INV1B resulted in a loss of voltage to the 120 VAC Channel B bus; the transfer scheme subsequently failed to operate. This particular bus does not supply the dual station computers or critical process systems, therefore the effect on plant operation was limited. One of the triplicated safety shutdown system channels tripped, but this did not cause a reactor upset.

INVERTER PAST PERFORMANCE

During the commissioning stage of the 120 VAC inverters, in 1982/83, design modifications were made to the transfer logic. These were required in order to prevent unwanted transfers to bypass supply. These transfers were being initiated by transients on the station service distribution system, which affected the inverter oscillator through its synchronizing circuits. After these changes were incorporated, the inverters operated from 1983 to 1988 with very few problems.

During the commissioning phase, 1982/83, there were also a number of 600 VAC inverter failures brought about by inverter room high temperatures. Air conditioning was added to the existing ventilation system to correct this problem. This no doubt, has contributed to the excellent operation of the 120 VAC inverters.
RECENT INVERTER FAILURES

Between January 09 and April 24/1988, the 120 VAC channel B inverter INV1B failed six times, while in service, with no resultant transfer to bypass supply. Each time this caused a loss of supply to channel B bus until a manual transfer could be completed.

The initial investigation indicated several components could have caused the failures. In each case failed components were replaced but normal testing of hardware and logic, at this point, did not result in finding the root cause of their failures.

The following inverter components were found to have failed:

- Varistors across the inverter SCR’s had shorted. In each case the varistor physically exploded after being overstressed.

- Inverter main fuse F1 and also the upstream 250 VDC supply fuses. This indicated that a coordination problem existed with these fuses.

- SCR’s and diodes shorted as their maximum junction temperature was exceeded during the high fault current conditions. Also, one diode failed due to localized overheating caused by improper installation on the heat sink.

- Inverter filter capacitor lead wires had burned, due to a design deficiency on wire sizing. Using a thermovision camera, a temperature of 170°C was observed on these wires which were only rated for 105°C.

- Contactor contacts in the inverter soft start panel (SSP1) circuit welded because of the high fault current conditions and lack of maintenance.

- Varistor terminal blocks broke down under normal voltage conditions because of impregnated carbon from previous varistor explosions.

- Main inverter oscillator overheated because of the heat from the input voltage dropping resistors located directly on the printed circuit board.

Inverter In-Service Failures

The in-service failures of inverter INV1B in most cases can be attributed to varistors failing in the commutation circuit. These devices are installed in this circuit to absorb voltage spikes which could otherwise over stress the SCR’s and diodes. The normal voltages stresses that the varistors encounter can not be reduced significantly below the original design value, unless a complete change of the existing commutation circuit, including the magnetics, is made.

From the failures experienced, it was concluded that the varistors were subjected to higher than normal voltage spikes. This could be the result of changes in the input voltage conditions and/or the
degrading of the capacitors or the surge suppressors within the inverter.

During high temperature excursions in the inverter room the probability of failure is increased because the performance of these solid state devices is heat dependent.

In order to alleviate these problems, four actions have been taken:

- Components which are subject to age related degradation will be replaced on a routine basis, based on the manufacturer's recommendations.
- All filter and commutation capacitors will be tested annually and replaced as necessary.
- The coordination, between the maximum voltage stresses within the inverter and the peak inverse voltage (PIV) limit of the solid state components, was upgraded. This was done by replacing the existing SCR's, diodes and varistors with ones that have a higher PIV stress limit.
- The taps on transformers and reactors were adjusted in order to reduce circulating currents in the inverter. This could be done, in this case, because only 20% of its capacity is used. The adjustment of taps will result in reduced heat losses and voltage stresses within the inverter.

Inverter Failure During Auto Transfer Tests

Inverter SCR's and diode failures were also occurring during normal testing of the inverter transfer equipment, at the instant the input breaker tripped. This indicated that the PIV stress limit was being exceeded for the solid state devices during the shutdown process. If the filter capacitors and surge suppressors do not properly absorb the excessive voltage spikes, the solid state devices are overstressed and fail. This problem has only recently appeared, indicating that the suppressors and filters have degraded to a level such that they are not providing proper protection.

In order to minimize failures, it was necessary to restore protection to the original design levels.

Inverter Transfer Logic Failures

From the analysis of the various field tests, transfer circuit design and transfer failures, it was concluded that part of the permissive logic was blocking the signal which would have fired the static bypass transfer switch. In order to prove this conclusion, a test was devised which would produce the same "violent" inverter failures as previously noted. Various test points of the inverter logic were monitored. The test results revealed that the transfer signal was blocked by the bypass undervoltage permissive, because of noise induced in the control logic.

The manufacturer, after reviewing their design, determined that the logic permissives of phase angle and bypass under/over voltage were not required to protect the UPS system.

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SCR And Diode Installation Problem

The reason for one of the failures of inverter INV1B was a diode in the commutation circuit, had not been centered properly on its heat sink. This occurred because the space between the front and back heat sinks was not large enough to allow for proper positioning of diodes and SCR's, during replacement. The higher PIV SCR's and diodes are thicker and allow easier installation on the heat sink.

INCREASE RELIABILITY OF STATIC BYPASS SWITCH

The bypass switch circuit contains an electromechanical contactor M1 in series with a static bypass switch SBS1. The M1 contactor was designed to close only if the inverter and bypass supply were in synchronism. With this permissive removed, the contactor now closes as soon as the transfer logic is energized. If the contactor were to fail open, an auto transfer could not take place. For this type of failure, the contactor would add to the unreliability of the circuit. Since the permissive was removed, and thus one of the reasons for the contactor, a review of the logic was done to determine if the M1 contactor could be removed altogether. It was concluded that, two scenarios where the contactor serves to increase the system reliability and that it should remain in the circuit. These are as follows:

- If the M1 contactor is removed and the static bypass switch SCR's failed while the inverter and bypass are in synchronism, then the two supplies would be paralleled continuously until the operator manually opens the S1 isolation switch. This would increase the common mode failure probability of the whole system.

- If the SCR's fail while the inverter and bypass are out of synchronism, then the two supplies will be paralleled continuously while out of phase and result in a loss of voltage to the 120 VAC bus.

To confirm that the contactor M1 is closed during normal operation, auxiliary contacts on the contactor have been wired into the inverter alarm circuit.

ACTIONS TAKEN TO RESOLVE FAILURES

The design and fault analysis showed that all of the problems experienced were correctable and inverter reliability could be improved if modifications were made to the original design. A complete replacement of the present system for new technology was discussed, but found not to be cost effective or practical at this time.

The following actions were recommended for each of the three 120 VAC inverters:

- Increase the PIV of SCR's from 900 to 1200 Volts, Diodes from 1100 to 1200 Volts and Varistors from 800 to 1000 Volts. This should virtually eliminate the type of failures that have recently been experienced.

- Replace these components at regular intervals:
  - Metal oxide varistors every 2 years.
  - Selenium surge suppressors X1, X2 every 2-4 years.
  - DC electrolytic capacitors C1 every 5-7 years.
  - AC capacitors C2, C3, and C6 every 10 years.

- Remove the transfer inhibits from the transfer logic, bypass under/over voltage and phase angle permissives.

- Improve the assembly of SCR's and diodes and heat sinks by changing to a higher PIV rating, which will increase the thickness from 14 to 26 mm.

- Monitor the contactor M1 in the transfer circuit, by using auxiliary contacts, to detect if M1 opens when the bypass and inverter are energized.

- Replace all power cables that have cracked and degraded from heat, with TEM 105°C type cable.

- Coordinate the main 250 VDC supply fuses with the inverter fuse F1.

- Incorporate routine scanning of the inverters with a thermovision camera in order to check for possible overheating of components.

- Check the oscillator/gate coupling module outputs as part of normal maintenance. Use an oscilloscope across the SCR gates in order to compare pulse quality against design specs.

- Measure the current drawn by the commutation capacitors C2A, C2B and wave shaping capacitor C3, every year, with a true RMS meter. If current reduces by more than 3%, check the individual capacitors and replace the defective ones.

- Check the soft start contactor SSP1, DC breaker CB1, SCR's, and diodes after every misfire and replace as necessary.

- Check the varistor terminal block resistance after each varistor failure and replace as necessary.

- Replace all of the varistors after each failure of the inverter, that causes the main fuse to blow.

- Work with the manufacturer to provide extra training at the skill and engineering levels on the existing inverters.

CONCLUSIONS

The inverter failures that have taken place over the past year, on 120 VAC inverter INV1B, can be directly attributed to insufficient maintenance. The preceding five years of excellent operation, led to a false belief that only minor maintenance was required, when in fact major components were slowly degrading in-service. These problems can be corrected with routine replacement of the components which are required to provide protection to the inverter circuitry.
The static bypass transfer logic was made too complicated, by trying to block transfers for conditions that rarely happen and would not cause damage to the UPS equipment. This lead to failed transfers, which decreased the reliability of the total 120 VAC system, for only marginal gains. Removal of these permissives increases overall reliability with no adverse impact on the equipment safety.

The inverters were originally sized too large for the application. Two of the inverters run continuously at 20% capacity, while the inverter that has given us the most problems operates at only 7%. This caused high circulating currents in the inverter voltage regulator which increased heat losses and stresses.
1.0 INTRODUCTION

The Darlington Tritium Removal Facility (DTRF) was originally constructed to help reduce the occupational tritium hazard and environmental tritium emissions at Ontario Hydro Generating Stations. The DTRF has also become a vital part of the entire Ontario Hydro Heavy Water Management System. The DTRF product, detritiated water, reduces the demand for virgin heavy water production at the Bruce Heavy Water Plant.

DTRF construction started in the summer of 1983 and the plant was declared in-service in October of 1988. At equilibrium the DTRF is expected to remove over 20 million Curies annually while maintaining an average tritium concentration of less than 10 and .5 Ci/kg in the moderator and heat transport system of all Hydro's reactors.

The DTRF is the first large scale plant of its type in the world. With this distinction there have been many setbacks that have delayed the commissioning of the plant and also contributed to the plants initial high unavailability. This paper chronicles the major setbacks that have been encountered in the commissioning and initial operation of the DTRF.

2.0 PROCESS DESCRIPTION

The DTRF process has been previously described in detail at the 1980 CNA Conference (1), and other publications (2),(3). To review, a simplified schematic (figure 1) and brief description of the main DTRF subgroups is given.

2.1 FRONT END

Feed Treatment System (FTS) removes dissolved gases and impurities from the Heavy Water feed using a degassing column, adsorption beds (activated charcoal) and filtration. The clean feed goes onto the Vapour Phase Catalytic Exchange (VPCE) where the tritium rich heavy water feed is contacted with tritium depleted deuterium gas over a platinum catalyst (at superheated conditions). Tritium in the vapour phase exchanges with deuterium in the gas phase by the following reaction:

\[
\text{DTO}(v) + \text{D}_2(g) \xrightarrow{\text{Pt}} \text{DT}(g) + \text{D}_2O(v). \\
\text{200 Deg.C.}
\]

Figure 1

DTRF - SIMPLIFIED FLOW DIAGRAM

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The resultant streams are a tritium rich deuterium gas stream and a tritium depleted heavy water stream. The heavy water stream is returned to the nuclear stations for reuse and the deuterium is dried in the Dryer Unit (DU) and sent to the cryogenic end to concentrate the tritium.

2.2 CRYOGENIC DISTILLATION

Tritium and deuterium are separated using cryogenic distillation at 25 Kelvin. The distillation end is separated into two cold boxes. The Low Tritium Cold Box (LTCB) containing the Low Tritium Column (LTC) which is approximately 30 m high and .85 meters in diameter at its widest part. The LTC bottom product is passed on to the High Tritium Cold Box (HTCB) to further concentrate tritium and the head product (depleted D2 gas) is returned to the VPCE. The HTC contains three separate High Tritium Columns (HTC). The column diameters range from 9 cm to less than 1 cm. The design product concentration at the bottom of the final HTC column is greater than 99.9% T2.

2.3 CRYOGENIC REFRIGERATION

Cooling to the LTC and HTC is supplied by the Cryogenic Refrigeration System (CRS). The CRS is a closed hydrogen refrigeration loop that uses a turbine expander and Joule-Thomson loop in parallel to cool the hydrogen to cryogenic temperatures.

2.4 AUXILIARY SYSTEMS

The auxiliary systems include the Drain and Purge (DPS), Recombiner (RS) and Deuterium Make-up system (DNS).

The DPS is designed to remove gases from various parts of the process. The system consists of a train of vacuum pumps and storage tank to hold the inventory evacuated from part of the system. The DPS is linked to a flame type Recombiner that oxidizes the process gases enabling collection and storage as heavy water.

The DNS is a conventional Teledyne electrolyser and is used to provide a source of deuterium for inventory adjustment.

2.5 AIR AND GLOVEBOX CLEAN-UP SYSTEM

The Air and Glovebox Clean-up Systems (ACS and GBCS) use a catalytic recombiner in series with a molecular sieve dryer to oxidize and collect tritium that has been released. The ACS is used to clean-up room atmospheres when room tritium concentrations exceed certain levels. The GBCS is provided to continuously clean the Argon atmosphere in the Tritium Immobilization System Glove box.

2.6 TRITIUM IMMOBILIZATION SYSTEM

The Tritium Immobilization System (TIS) accepts the HTC product and immobilizes the tritium as a tritide, on titanium beds for permanent storage. The titanium beds, Immobilized Tritium Container (ITC), can store up to 500,000 Ci which is equivalent to approximately eleven draw-offs from the HTC. The ITC are used as permanent storage devices. Uranium beds are provided also for temporary tritium storage.

3.0 COMMISSIONING AND INITIAL OPERATION OF THE DTRF

Commissioning of the DTRF was divided into five phases (see Figure 2). Phase 1 involved preoperational checks to visually, mechanically and electrically inspect process and control equipment. Services and auxiliary systems were made available including power, steam, water, instrument air. Once services were established the process systems were filled with Helium and heavy water and operated to confirm system integrity and equipment operation.

Figure 2
TRITIUM REMOVAL FACILITY COMMISSIONING OVERVIEW
1987 - 1988

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In Phase II the helium was replaced with the proper process gas (i.e., hydrogen or deuterium). All equipment was run to test performance on the proper process gas. In mid January 1988, Phase III, the DTRF was operated in a normal operating configuration using virgin heavy water and deuterium. In this time the liquid levels were established in the LTC and HTC and a number of draw-offs to TIS were demonstrated. In Phases IV and V tritium was introduced to the system. First at low concentrations (3mCi/kg) then later in reactor concentrations (15-17 Ci/kg).

Phase V of the DTRF commissioning ended on October 31, 1988 when the plant was declared in-service. Since that time tritiated heavy water has not been processed due to a series of critical equipment failures. The sequence and nature of the failures are summarized in figure 3. To summarize, the major setbacks that prevented successful operation of the DTRF in the initial operating months were:

- three separate failures of the CRS turbine unit.
- leaking tubes in the VPCE and FTS evaporators due to stress corrosion cracking.
- inability to establish and reliably maintain liquid levels in the HTC.
- Tritium surface contamination, control and monitoring problems.

Because of the wide nature of problems encountered both in commissioning and operation of the DTRF, each subsystem commissioning, operation and performance is discussed independently.

3.1 FRONT END

FEED TREATMENT SYSTEM

The FTS has worked well in cleaning the (FTS) heavy water feed to the VPCE process. The only operational problem experienced was leaking evaporator tubes caused by stress corrosion cracking combined with crevice corrosion at the rolled joint area. Using Eddy current inspection techniques four of twenty-eight tubes were found to be defective and plugged.

The failure was attributed to chlorides and other impurities from the VPCE catalyst, being washed to liquid collection and reprocessed through the FTS and concentrating in the evaporator.

VAPOUR PHASE CATALYTIC EXCHANGE (VPCE)

The VPCE was commissioned on Helium and H20/D20 during Phase 1 to prove its integrity. During Phase IV the first cooldown and operation of the plant using D20 and D2 gas was accomplished. The VPCE demonstrated its ability to operate on protium extraction. Phase III was followed by operation on a trace tritium feed of 3mCi/kg (Phase IV). Detritiation factors of 15-17 were consistently achieved during the tracer run. This compared favorably with the design value of 12 at a feed concentration of 3.4 Ci/kg. The concentration profiles in the liquid samples across the VPCE stages were uniform, indicating consistent performance across each stage.

Preliminary results obtained during preparation for the high tritium performance run indicated that the plant would operate as per design. The VPCE processed 62.5 Mg of D20 at an average concentration of 15 Ci/kg. The average product concentration was 0.45 Ci/kg.
The major problems encountered during VPCE commissioning were:

- Liquid ring compressor vibration levels unacceptable.
- Oxygen removal from both VPCE and Dryer units.
- Evaporator tube leaks (see below).

Chlorides and other impurities from the Catalyst leached out and concentrated in the VPCE evaporators. This resulted in stress/crevice corrosion cracking of evaporator tubes in the tube to tubesheet joint area. All evaporators except the first stage (was not exposed to catalyst impurities) suffered leaking or significant tube damage.

In December 88, while the CRS was shutdown for turbine repairs, the opportunity was taken to further inspect the VPCE evaporator tubes and as a result approximately 13 tubes were plugged. In January 1989 the decision was made to replace 9 evaporator tube bundles as a result of chloride stress corrosion and cracking. The tube material will be changed from 316L stainless steel to Inconel 625. The source of chlorides was traced to the original charge of catalyst (1% platinum on activated charcoal) which contained 1100 ppm chlorides.

DRYER UNIT (DU)

The DU was commissioned in the same manner as the VPCE. The system has operated well since deuterium was loaded. The dryer unit (DU) has consistently supplied gas with a dewpoint in the range -100 C to -110 C in deuterium gas. This is well below the design value of -90 Deg. C.

The major problems with the system have included:

a. Liquid Ring Compressor vibration and bearing failure. (Similar to VPCE compressor problems)

b. Tendency of Dewpoint probe to become wet and give irrationally high readings.

c. Excessively complex sequences make optimization of dryer bed operation difficult.

3.2 CRYOGENIC DISTILLATION

LOW TRITIUM DISTILLATION (LTC)

The LTC was commissioned with deuterium/protium, trace amounts of tritium and high tritium. In the separation of deuterium and tritium the column proved to be about 10% over designed. Protium extraction was also successfully demonstrated. The LTC provided the most reliable and stable operation of the three main cryogenic groups. Liquid deuterium levels were established and maintained easily and the LTC was relatively insensitive to process upsets. Equipment problems with the LTC were minor, the most significant being:

1) lack of instrumentation causes difficulties during plant start up and shutdown.

2) lack of control over the Nitrogen and Oxygen adsorbers.

High Tritium Distillation (HTC)

As with the LTC, the HTC's performance exceeded design expectations. During the High Tritium run, bottom product purity greater than 98% was achieved. Instrument inaccuracies and contamination in the TIS sampling system made bottom product analysis unreliable.

The HTC operation, unlike the LTC, was normally unstable and any fluctuation in the CRS operation often resulted in loss of liquid deuterium levels. The cause of the apparent inherent instabilities have not yet been pinpointed. But, it is felt that the problems may be associated with:

- poor design and control of Liquid Hydrogen to HTC condensers
- natural instability of the "syphoning" action used for mass transfer between columns.

3.3 CRYOGENIC REFRIGERATION SYSTEM (CRS)

The Cryogenic Refrigeration System was successfully commissioned with (Nitrogen) Helium, and Hydrogen. The demonstrated capacity exceeds both design expectations and process requirements. The over capacity has, however, led to process instabilities during low cooling demand.
Although the CRS process performance meets expectations, equipment unreliability has led to a high unavailability of the CRS. The most significant equipment problems encountered were:

1) Fluctuations in Low Pressure Service water supply combined with poor quality relief valves on the compressor LPSW cooling circuit has caused numerous compressor trips.

2) Turbine unit freeze up after compressor trips resulted in premature turbine unit failure. The cause of the freezing is a result of both inadequate turbine protection and operating procedures.

3) Turbine unit failure due to loss of braking and lubricating oil on failure of oil pump motor.

During commissioning and operation it was demonstrated that the availability of the entire DTRF is directly related to the availability and reliability of the CRS. During 1989 a number of changes are being instituted to improve the CRS reliability. These changes include among others a recirculating cooling water system and isolating valve for turbine unit.

3.4 AUXILIARY SYSTEMS

DRAIN AND PURGE SYSTEM (DPS)

The DPS consists of 3 pumps/vacuum pumps in series. The compressors have been shown to be undersized for evacuating the large gas volumes in the TRF process and also unreliable. Particularly in view of the number of evacuations required during commissioning.

RECOMBINER SYSTEM (RS)

The RS has been a constant source of problems. The inability to maintain a flame has been the major problem with this system. Because of the variety gas mixtures proper operation is usually achieved in the manual mode with a secondary hydrogen flow to promote burning. Operating problems are further compounded by the installation of laboratory standard flow elements unsuitable for an industrial application. As a result flow measurements have been erratic. Flow control valve calibration has been changed to assist in stable flame operation. A feature has been installed to provide excess H2 at the operators discretion to obtain stable operation. This results in a small downgrading penalty. D2O produced in collection tank has a very low pH (1), probably caused by high N2 (200 ppm) in the O2 supply.

DEUTERIUM MAKE-UP SYSTEM

This system has operated well, during the past year the major problem has been filter replacement due to impurities in the supply D20.

3.6 CLEAN-UP SYSTEMS

AIR CLEAN-UP SYSTEM (ACS)

Air clean-up system did not meet the original design requirements during commissioning. However, its performance was adequate to allow TRF commissioning to continue.

Deficiencies, which are or have been addressed include:

- Dryer beds incapable regenerating quickly enough to maintain design dewpoint values with the swamping vaporizer in service. The swamping vaporizer has since been isolated.

- System regeneration has been a problem due to a combination of passing regeneration system isolating valves, low heater output, and heat loss to atmosphere.

- Isolation dampers cannot prevent some degree of tritium cross-contamination to unaffected rooms while operating. The ACS has been automatically started on several occasions and has detritiated rooms containing low concentrations of elemental tritium. The amount collected has not been significant.

GLOVEBOX CLEAN-UP SYSTEM (GBCS)

This system has suffered similar problems to ACS.

The GBCS has been able to maintain the glovebox Argon atmosphere < 1 ppm O2 at a dewpoint of < -40 C consistently. As such it does meet its requirements. Regeneration system has required constant attention. Poor heat
transfer in Argon, combined with poor insulation and passing valves are thought to be the cause of the regeneration cycle not achieving its design temperature. To date no water has been collected in the collection tank. This is most likely due to using liquid as the source of Argon supply and a tight system. Addition of temporary insulation has improved operation, but lack of instrumentation has hampered troubleshooting activities.

3.7 TRITIUM IMMOBILIZATION SYSTEM (TIS)

The TIS proved to be a troublesome system during commissioning attributed to the adoption of lab style equipment and design to a production environment. Some of the major problems encountered are:

- Efforts to make the TIS suitably leak tight proved difficult due to the number of fittings. Tubing was also installed in such a way that it was under stress and forcing the many fittings such that leak rates were excessive.

- Inadequate cleaning of the system during installation resulted in the need to replace about 30 soft seat valve tips and one valve body.

- Tritium build up in TIS piping has become a problem due to inadequate Glovebox pressure control not providing a constant purge to the stack.

- Analysis of HTC Drawoffs has proved to be sensitive to conditioning of sample system piping. The highest purity T2 was obtained in the 8th drawoff late Oct/88 with the following results:

<table>
<thead>
<tr>
<th></th>
<th>Atomic Purity</th>
<th>Curies Drawn-off</th>
</tr>
</thead>
<tbody>
<tr>
<td>T2</td>
<td>95.44</td>
<td>97.29%</td>
</tr>
<tr>
<td>T20</td>
<td>1.93</td>
<td>44.851</td>
</tr>
<tr>
<td>D20</td>
<td>0.60</td>
<td></td>
</tr>
<tr>
<td>H20</td>
<td>0.51</td>
<td></td>
</tr>
<tr>
<td>T+</td>
<td>1.39</td>
<td></td>
</tr>
<tr>
<td>DTO</td>
<td>0.10</td>
<td></td>
</tr>
</tbody>
</table>

Actual T2 was probably higher than above but limitations of sampling system and Mass Spectrometer preclude a more accurate analysis.

- All drawoffs immobilized well onto the Titanium Sponge taking an average of 4 minutes to immobilize from 100kpa(a) to 110pa(a).

SUMMARY

The commissioning and operation of the DTRF has proved to be a long arduous task ridden with equipment failure and design deficiencies. This was expected with a plant of this nature and complexity. The plant has operated as per process design for a period of approximately 5 days. However, equipment unreliability has prevented demonstrating the DTRF's ability to achieve an 80% capacity factor. The current work programs are focused on improving the reliability and stability of the plant equipment.

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ABSTRACT

Pickering Unit 1 underwent a five month outage, ending April 1989, during which major modifications were made to all the safety systems. Since the reactor remained fuelled during the outage, complex scheduling and innovative work methods were required to ensure that all reactor safety considerations were met.

SUMMARY OF MODIFICATIONS AND MAINTENANCE

Pickering Unit 4 underwent a major rehabilitation outage from November 1, 1988 to April 16, 1989. During this outage, modifications were made to all of the safety systems that required total disabling of the systems for lengthy periods. These modifications included:

- installation of a new high pressure emergency coolant injection (ECI) system to replace the previous low pressure system
- installation of an additional 10 shutoff rods
- installation of a new pressure relief panel bulkhead connecting the reactor building with the negative pressure containment relief duct.

In addition to this extensive work on the three safety systems, there was other maintenance and modifications performed that impacted both on the complexity of the scheduling and on safety issues such as reactor heat sink availability. Some of these other jobs were:

- upgrade of the unit control computers with the addition of a new annunciation system
- pressure tube inspections and removal of two pressure tubes for further analysis
- adjuster rod replacement
- vertical in-core flux detector replacement
- rebuilting of all three condensers
- replacement of the low pressure turbine spindles and refurbishment of the high pressure spindle
- inspection and potential rewedging of the main generator stator
- replacement of all the turbovisory equipment with a new system
- installation of a new generator temperature monitoring system
- testing and/or replacement of all relief valves on the unit
- inspection and repairs of equipment in the calandria vault such as the bioshield piping and many structural supports.

Previous work of this nature had been performed on Units 1 and 2 in the period from 1983 to 1988. This was the first time that this type of work was done on a fuelled unit and in a period of five months rather than five years.

Reactor safety considerations that had to be addressed prior to and during the outage included:

- reactor monitoring and control including reactor shutdown guarantees required
- availability of a primary and alternate heat sink at all times
- preservation of independence on channelized systems
- availability of adequate emergency core cooling capability at all times
- containment system availability for Unit 4 and for the adjacent seven other operating units
- commissioning and testing required to confirm all new equipment operated as designed.

DESCRIPTION OF OLD AND NEW SYSTEMS

The major design features of the old and new systems will be reviewed prior to describing the methods used to address the reactor safety concerns.

ECI System

The original low pressure ECI system fitted to all the Pickering Nuclear Generating Station (NGS)-A units essentially consisted of connecting pipework and valves to allow the moderator pumps to inject moderator water into the heat transport (HT) system at about 700 kPa pressure. After initial injection, the recovery of water from the floors was directed to the suction of the same moderator pumps for injection back into the HT system.

The new high pressure ECI system used the Pickering NGS-B ECI pumps to pump light water from a storage tank into the HT system at about 4.2 MPa pressure. Recovery of water from the floors was still via the moderator system as before.

The increased pressure and water volume capability of the new system was one advantage. Reliability was also increased by duplicating critical valves. Operator involvement during the early stages was lessened by additional automatic features and water volume of the new system.
Additional Shutoff Rods

The original Pickering NGS-A shutdown system consisted of 11 gravity actuated shutoff rods worth about 24 mk with moderator dump as a backup. Concerns about possible pressure tube sag and ballooning following certain types of losses of coolant resulted in removal of the automatic dump feature for most accidents. In order to increase the shutdown reactivity depth and speed, 10 additional shutoff rods were added.

These new rods gave a faster power decrease due to their spring loaded actuation and due to the extra 11 mk of negative reactivity they represented. This extra negative reactivity gave a greater shutdown depth as well.

The new rods were installed in positions previously occupied by adjuster rods so now there were only eight adjusters instead of 18 with a lower reactivity worth of 12 mk versus 18 previously. Individual adjuster rods were now more heavily loaded to achieve this reactivity worth. Some consequences of this change were reduced decision and action times, reduced poison override time and longer poison out time following a reactor trip. Shim operation was also much more difficult with the heavier adjuster rods.

Pressure Relief Panel Bulkhead

The original connection of each Pickering NGS-A reactor building to the negative pressure containment relief duct was sealed by an arrangement of swinging louvres and foil blowout panels. The main function of the louvres was to prevent backflow of steam from another unit, should it have a loss of coolant, while being able to swing open to relieve pressure into the duct from the unit they were installed on. The foil blowout panels were mainly for atmospheric separation to allow heavy water vapour and contamination control.

The new system consists of stainless steel pressure relief panels only, which are much stronger than the previous foil panels. These new panels were welded to the reactor building to the negative pressure relief panels only, which are much stronger than the previous foil panels. These new panels were welded to the reactor building to the negative pressure relief duct at a pressure of 15 kPa in the positive direction and 40 kPa in the negative direction.

The benefits are elimination of leakage from non-accident units post loss of coolant accident so that containment repressurization times were greatly improved and also better atmospheric separation.

Computer Annunciation Enhancement

The Pickering Annunciation Computer Enhancement (PACE) upgrade involved the addition of five additional microcomputers to each unit solely for annunciation purposes. The original duplicated IBM 1800s still perform all the control functions.

The new PACE computer and associated displays allowed much quicker message turnaround following upsets, greatly improved trending and monitoring capabilities and a much larger number of annunciation points (required for all the new systems).

Reactor Operating Constraints During Shutdowns and Unique Features of This Outage

During any shutdown, “CONTROL,” “COOL” and “CONTAIN” issues must be addressed. This outage due to its length, complexity and amount of change created some unique concerns and solutions.

Reactor Control

Normal practice during any extended shutdown follows these basic principles:

- the reactor regulatory system maintains power monitoring and control capability as much as possible
- the reactor is guaranteed to remain shutdown by minimizing the moderator stays in the dump tank and by preventing additions of water to the moderator system
- a certain minimum number of shutoff rods are left poised
- work done on the protective system is done one channel at a time with the channel in a safe state, and fully tested prior to working on the other channels.

During this outage, the following circumstances prevented following normal practices:

- the PACE computer upgrades required removing the control computers (and their associated reactor regulatory functions) from service for about one month each
- regulating valve maintenance was required, so they could not be used as guaranteed open devices to prevent moderator pump up
- the shutoff rods were totally unavailable for about one month during addition of the new rods
- low neutron levels near the end of the outage would mean that normal control and protective instrumentation would be off scale low
- simultaneous multichannel work was necessary to minimize the overall duration and due to the complex removals of old wiring.

Reactor Cooling

Normal practice during shutdown is to always maintain a primary and backup heat sink. These heat sinks must have sufficient independence in terms of flowpath, fluid motive force, power supplies required and final heat sink such that any credible single failure would not disable both. The backup heat sink does not have to be continuously available, but must be easily put in to service within one hour of being required.

Fluid circulation methods include main HT pumps, shutdown cooling pumps and thermosyphoning. Final heat sinks include shutdown cooling heat exchangers, the HT bleed cooler, main boilers and blowdown, main boilers and steam release valves and reactor vault air cooling units. Some of these methods require HT pressurization as well.
As a backup to these two heat sinks and to cater to pipe breaks, the EC1 system is maintained available within a defined recall time based on fuel decay power.

Some of the heat sink concerns that had to be addressed during this outage included:

- the condensers were out of service for retubing so the normal flowpath of feedwater to the boilers was not available
- two shutdown cooling loops would be out of service for up to a month to allow tie-ins of the new high pressure EC1 pipe work
- the HT headers would have to be drained to mid-level to allow replacement of shutdown cooling loop valve bonnets and actuators as part of the EC1 upgrade
- pressurization of the HT system was not possible during much of the outage due to limitations imposed by pressure tube inspections and due to the EC1 work.

As well, during about one third of the outage, neither old or new EC1 systems were available due to the removal and installation work that was ongoing.

**Containment**

Normal practice is to maintain a fully functional containment boundary around the shutdown unit and maintain a connection to the vacuum building while maintaining separation from other units via the louvres and foil blowout panels at the bulkhead to the pressure relief duct.

During this outage, horizontal flux detectors were being changed that were longer than the reactor building airlocks. Also, installation of the new pressure relief panel bulkhead had to occur without disturbing the separation or connection provided by the old bulkhead.

**Outage Planning Organization**

Due to the number of reactor safety issues that had to be addressed in a coordinated fashion, extensive preplanning of the outage was required.

A dedicated outage team was set up almost a year before the start of the outage to begin laying out the strategy to be followed. This team consisted of a Technical Superintendent, Shift Supervisor and a Planning Supervisor. They worked very closely with another special group that was set up on a part-time basis, the Reactor Safety Review Team, consisting of several Technical Superintendents from various disciplines.

The Reactor Safety Review Team produced, very early on, an overall strategy document entitled "The Unit 4 Reactor Safety and Outage Policy Guideline." This document was used to create the overall outage logic, was used by the Technical Section personnel to guide the direction of the work plans for the outage and was used as a training document to inform the Production Section personnel of the background reasons for some of the ways work was being done. Extensive briefing sessions were held with the supervisors of each Production Section shift crew to review the guidelines and overall outage logic prior to the start of the outage.

**Methods Used to Address Reactor Safety Issues**

Since the outage critical path and most of the extensive modifications were involved with the installation of the new EC1 system, all other work was scheduled to mesh with the EC1 work.

The outage was divided into five phases, each approximately one month duration (see Figure 1). This allowed the scheduling of work requiring compatible conditions into each phase and broke the outage down into manageable chunks with several clearly defined milestones to aim for.

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**FIGURES 1: HPEC1S INSTALLATION PLAN**
The first phase was actually prior to unit shutdown. As much preshutdown wiring and ECI installations as possible were completed prior to bringing the unit down.

Phase 2 during the first month of the outage was reasonably normal as outages go. Major work during this phase involved:

- installation of the PACK computer upgrade on one of the control computers
- pressure tube inspections and isolation of one pressure tube for later removal
- removal of 18 old adjuster rods and installation of eight new heavier worth adjusters and 10 additional shutoff rods
- removal of instrumentation associated with the low pressure ECI system.

During phase 2, automatic ECI initiating controls were unavailable, but sufficient annunciation and indication were left in place to permit the operators to recognize a loss of HT coolant and take action to initiate manual recovery and injection of water.

Phase 3 saw some of the most complicated work performed such as:

- major tie-ins of high pressure ECI piping requiring total disabling of the old ECI system
- electrical work associated with tie-ins of the 10 new shutoff rods requiring total disabling of all the shutoff rods
- installation of the PACK computer upgrade on the other control computer
- removal and replacement of a pressure tube.

During phase 4, sufficient portions of the ECI system were once again available to credit manual actuation. The original 11 shutoff rods were also returned to service.

The work that had the most effect on scheduling during this phase was draining of the HT main headers to allow installation of new shutdown cooling isolation valve bonnets and actuators. This draining and deliberate opening of the HT system disabled most of the normally used reactor heat sinks.

Phase 5 was mainly a commissioning and testing of new systems phase.

The techniques used to deal with the CONTROL, COOL, CONTAIN concerns during the above phases were as follows.

Reactor Control

During phases 2, 4 and 5, a standard moderator dumped reactor shutdown guarantee was used except that two channels of dump valves were guaranteed open instead of using the regulating valves as is normal practice. This allowed the completion of essential maintenance on the regulating valves while still maintaining an open flow path from calandria to dump tank to prevent inadvertent moderator pumpup. Because of the three channel arrangement of the dump valves, there was also still a flowpath that remained closed but available to open under protective system action. This was an important consideration that was especially vital during phase 3 when all the shutoff rods were unavailable and moderator dump was the only method of fast reactivity reduction.

An additional step taken during phase 3 to enhance the normal moderator dumped shutdown guarantee was to drain almost all of the moderator water out of the main moderator system into the helium storage tank outside the reactor building and guarantee that it stayed there. This physical method of preventing inadvertent pump up and criticality was judged to be much more certain than other methods proposed such as overpoisoning the moderator water in the dump tank.

One control computer was kept available at all times to continue reactor power monitoring. The protective system instrumentation was also still functional and able to trip the reactor via dump valve action. This required some jumpers in the protective trip line due to the electrical work in progress associated with shutoff rod installation.

About two thirds of the way through the outage, the decay power dropped low enough that the normal instrumentation went off scale low. Reactor startup instrumentation was then connected into the protective system to allow continued flux monitoring. For the first time, this instrumentation was installed in a fully channelized manner, including channelized power supplies.

Simultaneous multichannel work was permitted for the installation of new systems provided that different people worked on each channel and provided full commissioning of the system was performed prior to placing the system in service.

Reactor Cooling

The two aspects of fuel cooling, normal cooling via a primary heat sink with a standby alternate, and emergency cooling during pipe breaks, had to be addressed for each phase of the outage.

Since the condensers were being retubed, special bypass pipework had to be installed from the main demineralized water storage tank to the auxiliary condensate extraction pump to allow the crediting of the feedwater fed heat sinks. Much of this pipework required housing and tracheating as it ran outside.

Work was generally scheduled during each phase so that one (of two) shutdown cooling loops in each HT loop was always available as the prime heat sink. For much of the time the opposite shutdown cooling loop (with pump fed by different power supplies and cooled by a sufficiently redundant service water system) was credited as the alternate heat sink. The boilers and blowdown valves were used as the alternate during ECI piping tie-ins to the opposite shutdown cooling loops.

During phase 4, the vault air cooling units were the only heat sink that could be used with the main HT headers drained to half level. This required removal of all the reactor face jigsaw insulation panels to allow the cold air to blow across the feeder pipes. With the feeder pipes acting like
radiators and thermosyphoning providing the circulation, tests were done to confirm the capacity of this heat sink. These tests showed that the initial rate of rise (from 27°C) of the HT system temperature was 3°C/hr peaking at 40°C. The alternate heat sink in this case was credited as shutdown cooling which would have required closing up the HT system and refilling within the time allowed of three hours. Rigid control of water level via temporary manometers was used to ensure that no feeder pipes were inadvertently uncovered thereby allowing that channel to heat up separately from the bulk system.

Both computers were returned to service prior to this work to ensure full channel temperature monitoring was available.

The other aspect of fuel cooling, that of ensuring adequate cooling during pipe breaks was also a concern during phase 3. During phases 2, 4 and 5, manual actuation of the ECI system components was available, but during phase 3 the pipework tie-ins disabled all normal flowpaths for recovery water.

The heat transport recovery system was credited as an alternate ECI system during this phase. However, since this system only had a recovery capacity of about 19 kg/sec, it could not feed all break sizes. To eliminate the possibility of larger breaks occurring, the following precautions were taken during this period:

- no HT pressurization was allowed
- no maintenance on the HT system pressure boundary was allowed
- no craning was allowed over large bore HT piping
- fuelling machine bridge movements were carefully controlled to ensure there was no damage to pressure tube end fittings.

As well, a flow test was conducted to prove the capacity of the bP0 recovery pumps prior to crediting its availability.

Containment

Normal practices were followed to ensure that systems were not simultaneously opened up inside and outside the reactor building, thereby bypassing the containment boundary.

Installation of the new pressure relief panel bulkhead was carefully controlled to ensure continuous connection to the pressure relief duct when required along with separation from the other reactor buildings. By building the new bulkhead in front of the existing bulkhead containment testing requirements were also met.

Flux detector replacement required the installation of airlock extensions to allow bringing in the long flux detectors without opening both airlock doors simultaneously.

Testing and Commissioning

All of the newly installed equipment was extensively tested prior to releasing it for service, and fully commissioned using the same procedures used for a new station. This included activities such as:

- issuing detailed commissioning specifications with formal Design Division acceptance
- issuing detailed commissioning procedures
- conducting commissioning completion assurance meetings to confirm to station management and the AIC of the systems met their detailed commissioning specifications.

The ECI system was completely functional checked right up to the point of actually injecting water. Wire-by-wire checks were done on all newly installed wiring. Independent work verification and system cleanliness checks were conducted at all critical points.

Complete reactivity worth physics tests were done because of all the new reactivity devices and to measure the new core flux as a result of having refuelled 24 channels during the outage after pressure tube inspections. As well, three channels were left empty to reduce pressure tube growth and allow continued operation until 1991.

SUMMARY

This outage was very successful due to several factors:

A dedicated team from the Production, Technical and Planning Sections worked very well at ensuring all aspects of outage planning were covered before the outage.

Extensive preplanning of most of the outage allowed optimization of all the modifications to minimize conflicts and avoid any reactor safety concerns. All work plans and outage logic subnets were essentially complete prior to the start of the outage. Very few changes were required in this overall logic during the course of the outage.

The "Reactor Safety and Outage Policy Guide" was a very useful tool for everyone concerned with the outage. This guide and the close review by the Reactor Safety Review Team ensured that all potential problems were addressed before they became a concern.

Extensive use of special crews for all of the major maintenance was a big factor in eliminating rework and in minimizing loss of production due to turnover problems. These dedicated crews were used for all of the turbine/generator/condenser work, for all the ECI wiring, shutoff rod wiring and computer changes.

Preoutage briefings with all of the duty shifts ensured that they were aware of all the issues and were committed to doing their part of the work. Generally, all of the routine maintenance was handled by the duty crews.
In the end, the outage was completed with no lost time injuries, no unplanned dose, all major work completed as planned and a significant amount of additional pressure tube work performed with only a 14 day delay over the originally approved schedule.

The same outage planning and organization will be used for the upcoming eight-unit outage in 1990 for the regular vacuum building inspection.
THE IMPACT OF FISSION PRODUCTS ON RADIATION FIELDS IN ONTARIO HYDRO CANDU REACTORS

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ABSTRACT

This report summarizes reactor face radiation field measurements made since 1986 in Ontario Hydro Nuclear Generating Stations. These fields are attributed to Depositing Fission Products (Zr/Nb-95 and La-140) and Activated Corrosion Products (predominantly Co-60) based on gamma spectroscopy measurements. The results show that while Depositing Fission Product field contributions are minimal in Bruce reactors (<20%), they are dominant in some Pickering reactors (>80%). This disparity has arisen because Pickering reactors, unlike Bruce reactors, do not employ Delayed Neutron Detection Systems which facilitate on-power removal of defective fuel. It is concluded that significant radiation field reductions result from the use of Delayed Neutron Detection Systems and that their maintenance and operation should be continued at those stations where they exist. In addition, it is concluded that fission product radiation fields can be minimized in future CANDU reactors by employing Delayed Neutron Detection Systems.

HEAT TRANSPORT SYSTEM RADIOACTIVITY - SOURCES AND BEHAVIOUR

Radioactive fission products are not the only contributor to HTS associated radioactivity. Other contributors may be broadly referred to as activation products because they are produced when inactive species contained in the HTS are exposed to neutron flux. Two of the most significant activation products are tritium and Cobalt-60.

Tritium is produced when the D₂O used to cool (and moderate) CANDU reactors is exposed to neutron flux. Tritium is a beta-emitter which only poses a hazard when it escapes its confines into the surrounding plant environment where it can be inhaled or absorbed through the skin. Tritium is not a gamma hazard.

Cobalt-60 (Co-60) is produced from activation of corrosion products of the HTS components which contain inactive Co-59. Co-60 is characterized by a 5.3 year half-life and the emission of gamma radiation at 1173 and 1333 keV. Once activated, Co-60 deposits downstream of the core where it becomes incorporated into the oxide layer on the inner surface of HTS components such as feeder pipes and boilers. The above characteristics make Co-60 a significant HTS gamma radiation hazard which has frequently dominated contributions to radiation doses during maintenance outages.

Generally, Activated Corrosion Product fields can be controlled by ensuring that HTS alloy levels of cobalt are minimized during the design and construction of the plant. HTS chemistry control is also an important factor in controlling Activated Corrosion Product transport and growth. In cases where Activated Corrosion Product fields have
become excessive, HTS decontamination (CANDECON) has been performed to reduce radiation fields (1).

The behaviour of Activated Corrosion Products in the HTS, particularly Co-60, has been documented elsewhere (1) and is not discussed in detail in this paper.

Fission products produced by the fissioning of UO₂ fuel are normally contained within the Zircaloy-sheathed fuel bundles. However, when sheath defects occur, fission products are released to the HTS. Some of these fission products are depositing in nature and emit gamma radiation so that they pose a hazard similar to Co-60 as described above. Other gaseous and soluble fission products such as the noble gases and radioiodines are short-lived and do not pose a significant radiation hazard during HTS maintenance as do the Depositing Fission Products and Activated Corrosion Products. Like tritium, the gaseous and soluble fission products are only hazards when they escape the confines of the HTS.

The predominant Depositing Fission Products giving rise to radiation fields are Zirconium-95 (Zr-95), Niobium-95 (Nb-95) and Lanthanum-140 (La-140) (2) which emit gamma rays in the range of 400-1600 keV. Unlike the gaseous and soluble fission products, Zr/Nb-95 and La-140 are bound within the UO₂ matrix and are released only when the UO₂ matrix itself breaks up and is released to the HTS (3). Zr/Nb-95 and La-140 have half-lives of 64 days and 40 hours, respectively. A previous study (2) has demonstrated that these isotopes may exhibit effective half-lives of up to 12 months due to the continued fissioning of released uranium ("tramp").

Figure 1 shows that a 3 year period was required for fields to decay to normal levels following a uranium release event to the HTS of Bruce NGS-A Unit 1 in 1979.

It has also been demonstrated (2) that the level of Depositing Fission Product radiation fields in the HTS is proportional to the level of in-core defective fuel (which may also be correlated with HTS Iodine-131 concentrations - Figure 2). This shows that Depositing Fission Product radiation fields may be controlled by limiting the level of operating defective fuel in-reactor.

In summarizing this background information, we can say that the predominant contributors to HTS gamma radiation fields are the Activated Corrosion Products (predominantly Co-60) and the Depositing Fission Products Zr/Nb-95 and La-140. The remainder
of this paper will concentrate on the recent contribution of Depositing Fission Products to radiation fields (relative to that of the Activated Corrosion Products) in Ontario Hydro CANDU reactors.

RECENT MEASUREMENTS IN ONTARIO HYDRO NUCLEAR GENERATING STATIONS

During scheduled maintenance outages, gamma radiation fields are measured at the reactor face, feeder cabinets and boilers. Gamma spectrometry equipment is used to identify the radioisotope sources. Table 1 summarizes the reactor face radiation field data measured at the Pickering and Bruce Nuclear Generating Stations since 1986. The fields are broken down according to their source: Depositing Fission Products (predominantly Zr/Nb-95 and La-140) versus Activated Corrosion Products (predominantly Co-60).

<table>
<thead>
<tr>
<th>TABLE 1</th>
<th>SUMMARY OF FISSION PRODUCT CONTRIBUTIONS TO REACTOR FACE RADIATION FIELDS (RECENT MEASUREMENTS SINCE 1986)</th>
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*Average values of selected lattice positions covering the reactor face. Measurements made 1 in from the reactor face.

A significant difference in fission product contributions exist between Pickering and Bruce reactors (Figure 3). Pickering reactors exhibit Depositing Fission Product fields in the range of 50-225 mR/h constituting up to 80% of the total measured field. Bruce reactors exhibit Depositing Fission Product fields in the range of 5-30 mR/h which constitute less than 20% of the total measured field.

DISCUSSION

Although some of the difference observed in Depositing Fission Product radiation fields in Pickering and Bruce reactors may be due to design variances, the primary reason for the difference observed in Figure 3 is that Pickering reactors do not employ On-Power Defective Fuel Location Systems as do Bruce reactors (Delayed Neutron Detection System). As a result, defective fuel may be left in-core for many months in Pickering reactors releasing significant quantities of uranium and Depositing Fission Products. Defective fuel is generally removed from Bruce reactors before significant uranium and Depositing Fission Product release can occur.

The variances observed within each station are due primarily to differences in fuel defect rates. At Pickering, HTS debris has caused fretting defects in fuel in Units 5,6 and 7. HTS 1-131 concentrations and post-discharge fuel examinations have suggested that the problem was most severe in Units 5 and 7. This is consistent with the observed fields in Figure 3. In Bruce reactors, a smaller variance is observed despite the fact that the defect rates in Units 3,5 and 6 have been higher than in other units. This demonstrates that on-power fuel defect removal minimizes the impact of increasing fuel defect rates.

CONCLUSIONS

The results of this study support the following conclusions:

1. A significant difference exists between fission product radiation fields in Bruce and Pickering reactors.
(2) Pickering fields are highest because those reactors do not employ On-Power Defective Fuel Location Systems.

(3) Fission product radiation fields can be minimized in future CANDU nuclear generating stations by employing proven, reliable On-Power Defective Fuel Location Systems. Presently, the Delayed Neutron Detection System is the only system qualified for this role.

(4) In CANDU stations which currently incorporate Delayed Neutron Detection Systems (Bruce A&B, Gentilly, Point Lepreau, Embalse and Wolsung), continued operation and maintenance of the system will minimize fission product radiation fields.

(5) In CANDU stations which currently incorporate Feeder Scanning Systems (Pickering A&B and Darlington), improved use of the system (particularly on-power) will result in fission product radiation field reductions.

ACKNOWLEDGEMENTS

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ABSTRACT

On April 25, 1987 during a planned 800 MW(e) load rejection test on Unit 8, a violent transient occurred in the deaerator/deaerator storage tank which caused the 500 ton vessel-piping network to tear away from its supports and move approximately three inches in an axial direction. Considerable effort was brought to bear and within one week the unit was returned to service. This paper describes the physical nature of the event and the role of the Bruce NGS "B" training simulator in developing and testing control modifications to prevent its recurrence.

SYSTEM DESCRIPTION

Purpose of the Deaerator

The primary purposes of the Deaerator Heater/Deaerator Storage Tank (hereafter referred to simply as the Deaerator) are to:

i) Strip oxygen and other non-condensible gases out of the incoming condensate.
ii) Heat the condensate by utilizing 7th stage turbine extraction steam.

The combined heating/degassing function is achieved by passing condensate down through spray valves and trays against a counter flow of turbine extraction steam.

As a secondary function the Deaerator, by virtue of its elevation, provides about 33 m of head for the boiler feed pumps thereby ensuring sufficient NPSH (net positive suction head). In addition, the size of the deaerator (377 Mg of water at normal 75% level) provides enough inventory for five full power minutes of feedwater to the boilers should the condensate system become unavailable. The primary inputs and outputs are shown in Figure 1 for a typical Bruce B unit.

Normal Deaerator Operation and Control

The Deaerator normally operates in thermodynamic equilibrium between incoming condensate, extraction steam, and heater drains. Any sudden change to any of these parameters creates a transient of varying degrees in either deaerator level or pressure. The severity of the transient will vary from minor for a small load change to extreme for a full load rejection. A high level transient is a concern because of the possibility of flooding the extraction steam lines and subsequently admitting water into the turbine. At the low end, the concern is the potential loss of feedwater to the boilers. In addition to level transients, pressure transients can cause internal damage to deaerator trays, or severe flashing and boiling of the deaerator contents.

Realizing that the deaerator normally operates at saturated conditions of approximately 400 kPa(g) and 152°C for 377 Mg of water, one can appreciate the large amount of stored energy present during normal operation. This stored energy (2.42 x 10^9 Joules of specific enthalpy) is contained in the deaerator storage tank whose pressure boundary consists of 12mm (1/2") of carbon steel wall. A Bruce "B" boiler has a stored energy of 0.47 x 10^9 Joules specific enthalpy and a pressure boundary wall of 68 mm (2.67"). Compared with a boiler, the deaerator has five times the stored energy contained within one fifth the wall thickness. It is clear that smooth and reliable control is essential to ensure that the stored energy does not precipitate structural damage.

Level Control. Level is controlled by the Deaerator Level Control (DLC) program, running in the unit computers. DLC is a three-element proportional plus integral cascaded controller using deaerator storage tank level, condensate extraction pump, and boiler feedwater flow. The program calculates a new condensate control valve (LCV) position every two seconds using a control scheme as represented in Figure 2.
During commissioning at BNGS'B', the controller parameters (proportional bands and reset times) were tuned to provide maximum level sensitivity while maintaining adequate stability during transients. To some extent this was done at the expense of responsiveness to condensate/feedwater flow mismatch. The strong bias toward level sensitivity in the controller parameters was intended to minimize the possibility of exceeding full tank level and consequently flooding the extraction steam lines and possibly causing water induction to the turbine. This concern is also covered by hard-wired logic which trips the condensate LCV's closed on very high level (90%).

The flow controller portion of DLC is designed to respond to feedwater flow changes due to the Boiler Level Control (BLC) demands. In an effort to keep boiler levels steady, BLC modulates feedwater flow in response to changes in levels, steam flows, feedwater flows, steam pressure, and reactor power. These translate to condensate flow changes through the flow controller portion of DLC.

There are two additional sources of mass flow which are unmonitored and could affect level. The largest is heater drains flow which is approximately 20% of total mass input, and the second is extraction steam flow which is about 5% of total mass input.

**Pressure Control.** At normal operation under saturation conditions, deaerator pressure is simply the pressure resulting from the temperature achieved after incoming condensate is heated by condensing extraction steam. The saturation temperature and pressure of the deaerator is therefore a function of unit load.

Under transient conditions where extraction steam is lost, there is no pressure control until the deaerator pressure decays to 69 kPa(g), the setpoint of the poison prevent steam controller. The poison prevent steam pressure control valves (PCV's - see figure 1) can admit large amounts of primary steam sufficient to warm 630 kg/s condensate flow (60% of full power flow) from 30°C to 115°C. The rationale for the setpoint of 69 kPa(g) (saturation temperature 115°C) is simply the requirement of to maintain a minimum feedwater temperature to the boiler and preheater tubesheets. Hence, the poison prevent PCV's are meant more for feedwater heating than for any deaerator pressure control.

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**The Event**

On April 25 1987 a planned 800 MW load rejection was conducted on Unit 8, primarily to study torsional vibration on the turbine. As part of this load rejection, previous concerns regarding deaerator level control were to be monitored both in the field and in the main control room.

Deaerator level control during a load rejection had always been a problem in all previous units and a perception had developed that restricting or shutting off the poison prevent steam during the transient aided in settling out deaerator level oscillations. Taking this one step further, it was decided to only allow half the poison prevent steam to be available (by selecting one isolating valve closed*) and to slow down the response of the PCV controller. The design setpoint of 69 kPa(g) was left unchanged.

With an operator dedicated to the field to observe the deaerator storage tank level sightglass, the unit load rejection was manually initiated. The main control room indications showing large level and pressure swings were quite normal and consistent with previous load rejections (refer to Figures 3 and 4). The field observations, however, were quite different from those of the main control room.

The field operator reported the following:

"...there was a series of several bangs over what seemed like about four minutes - at lower and higher levels on the sightglass ... the support posts holding the junction boxes for the immersion heaters were moving with each bang ... The main steam line overhead was swaying back and forth, the floor was rumbling and the lighting above was swinging. The deaerator appears, piping included, to have shifted ..."

Indeed the deaerator had shifted approximately 5 to 7 cm and the severe banging was credited to the implosion of voids created by rapid and uncontrolled boiling of the deaerator storage tank contents (reference 1).

Once the extent of the event was fully realized, the unit was immediately shut down in a controlled manner. Station technical and Head Office design staff were assembled and a plan of analysis, inspection, and repair were drawn up. The priorities in the plan were as follows:

i) Notify the representatives of the Atomic Energy Control Board and the Ministry of Consumer and Commercial Relations.

ii) Discuss the root cause of the event and formulate whatever immediate changes may be required to keep the other three units operating and eventually to restart Unit 8.

* One poison prevent steam isolating valve was similarly kept closed during a Unit 8 850 MW load rejection on March 21 1987 (see Figure 4).
iii) Get the vessel moved back to its original position.

iv) In parallel with iii), decide on the areas of high stress and devise an inspection plan acceptable to the authorities. These high stress areas were to be formally analyzed and confirmed by design staff within one week.

v) Once the vessel was returned to its original position, perform the inspections required to confirm the integrity of all suspected high stress areas.

vi) Making a repair resulting from v) and draft a formal explanation of the event, the inspection, and repair, and any remedial changes required to prevent recurrence.

The above activities were completed and within one week of the initial event, the unit was returned to service at full power.

In reference to iii), iv), and v) above, the vessel structure was returned to its original position within three days of the event by site construction forces. They achieved this by judicious placing of wedges and hydraulic jacks. Inspection of stress areas proceeded around the clock following the return of the deaerator to its rest position. Over 125 weld inspections were done primarily using magnetic particle technique. Only two areas showed indications. The first was a 1-2 mm hairline crack on the structural steel (not pressure boundary) on the deaerator fixed support. The other indication was a 3/16" deep by 3/4" long crack on a 3/8" wall pipe at a point where this pipe penetrated the floor sleeve one floor below the deaerator. Both indications were repaired prior to restart.

In ii) of the above plan, the immediate explanation of the event was that the deaerator movement resulted from severe banging and steam hammer in the storage tank. This, either alone or in conjunction with a wave action sheared twenty 1" carbon steel hold-down bolts and moved the structure and its attached piping. The degree of turbulence in the storage tank was attributed to 377 Mg of water at saturation temperature having its pressure rapidly reduced by the termination of extraction steam. Poison prevent steam was admitted much too late (at 69 kPa(g)) and not fast enough to have any effect on deaerator pressure. In fact, the majority of the event had already occurred prior to the poison prevent steam pressure setpoint being reached. Realizing that deaerator level control concerns would have to take a back seat to what transpired in Unit 8, pressure control became the focus of action, in the short term.

It was apparent from the data that the long period of depressurization following a load rejection could be shortened if poison prevent steam were admitted much earlier (at a higher setpoint) than was currently the design. On this basis the poison prevent steam pressure setpoint was raised from 69 kPa(g) to 210 kPa(g), a value consistent with other CANDU stations. In addition, all poison prevent steam would be available (both isolation MV's, when selected "Auto", would open on a power load unbalance signal as well as upon approach to controller setpoint) and the controller response was speeded up. No action was taken to strengthen the anchoring of the deaerator for fear of transmitting the shear forces into the vessel in any repeat incident. Therefore, the original grade of bolts was used for re-anchor-

The above control changes, along with a commitment to the authorities to further analyze the event in detail allowed the restart of Unit 8 and the continued operation of Units 5, 6 and 7. It was clear at this point that the training simulator was to play an important role in the coming analysis effort.

UNANSWERED QUESTIONS

The interim modifications to the poison prevent steam system were based on a limited understanding of the detailed mechanisms responsible for the damage. Although it was clear that an excessive rate of depressurization was at the centre of the problem, there were still some key questions to be answered before the details of any permanent modifications could be decided:

i) Why was there an initial apparent drop in level, even though condensate flow had increased and feedwater flow was dropping rapidly?

ii) After the initial drop in pressure, how could the level rise so suddenly when there was no condensate flow?

iii) How strongly is the depressurization rate linked to condensate flow?

iv) To what extent could DLC tuning be expected to limit depressurization rate while maintaining adequate level control for all operating configurations?

v) To what extent could the early admission of poison prevent steam limit the rate and extent of depressurization?

vi) Why didn't the 850 MW Unit 8 load rejection just a month prior to the event (see figure 4) display any severe banging or movement? A field observer in that case noted no undue noise or unusual indications.

THERMODYNAMIC MODEL

Working Hypothesis

Limited availability of data dictated that many of the details of the incident and subsequent damage must remain open to speculation. Nevertheless, once the unit was returned to service and the initial analysis was in hand, a multidisciplinary design/operations team was able to construct the following working hypothesis upon which to base our longer term solutions:

The sudden and complete loss of heating steam, coincident with a large inflow of cold (30°C) condensate, resulted in a rapid depressurization of the deaerator compartment. The rapid pressure drop caused severe flashing (and banging upon collapse of the voids) of the stored water in the deaerator storage tank. A mixture of liberated steam and water moved...
very rapidly up through the equalizing lines and downcomers, thereby preventing the water in the deaerator vessel from falling normally into the storage tank.

As the water in the storage tank approaches equilibrium, and poison prevent steam is admitted, the pressure drop stops, and the held up water flows quickly down into the storage tank.

In addition, it was postulated that a "water-piston" may have been precipitated by an effect known as a Kelvin-Helmholtz instability (reference 2). Because of the rapid boiling, the liquid/steam interface was near the top of the storage tank. Waves in the storage tank were "pulled upward" by the passage of high velocity liberated steam toward the centre of the vessel. The waves contacted the top of the tank, and the resulting "water piston" was accelerated toward the end of the tank. This mechanism remains in the realm of speculation until further analysis is done.

Although the exact moment of the damage is not known, it is assumed to have been caused by shock waves generated by steam hammer, in possible combination with the above "water piston" effect or severe wave action precipitated during the transient.

Training Simulator Modelling Changes

Based on the above understanding, the following changes were made to the simulator model to improve the match with station data (reference 3):

i) A "hold-up" zone was created through which condensate passes before entering the storage tank. The rate at which the condensate is held up was made proportional to the rate of change of pressure.

ii) To more accurately model boiling in the storage tank, the calculations were made dependent on hydrostatic head.

These changes resulted in excellent agreement with station data (see figure 5).

EXPLANATION OF THE EVENT

Given the excellent agreement with station data, we were then confident to offer the following explanation of the order of events (refer to Figure 3):

Immediately at the time of the load rejection: Extraction steam flow disappears, as does HP Heater Drains flow (the pumps trip on low turbine steam pressure). Feedwater flow increases suddenly and briefly in response to the reactor stepback and increasing boiler steam pressure.** The Deaerator Level Control program responds to the feedwater/condensate flow imbalance by opening the condensate valves. The resulting drop in condensate line pressure auto starts the Standby Condensate Extraction Pump, thereby augmenting the flow increase. This increase in condensate flow roughly offsets the loss of HP Heater Drains flow, and the storage tank level initially stays constant. The influx of unheated condensate results in quenching of the steam in the deaerator vessel and the start of a rapid depressurization.

After about 1 minute: Feedwater flow is dropping rapidly to match the heat removal requirement of about 60% of nominal full flow. The indicated deaerator storage tank level begins to fall, even though condensate flow remains elevated to about 115% of pre-trip flow. Pressure is falling rapidly, and boiling occurs throughout the storage tank. The resulting localized pressure fluctuations cause erratic level indication, although the overall values are consistent with sightglass measurements in the field. The flow of liberated steam through the equalizing pipes is very rapid, and prevents the condensate in the deaerator vessel from falling into the storage tank (hence the dropping level). It should be noted that the sudden level fluctuations evident on the level trend (figure 3) coincide exactly with the observed "banging" in the field, including one occurrence at about 8 minutes, during the second depressurization cycle.

After about 4 minutes: Rate of depressurization has decreased, though pressure is still falling. Storage tank level is rising rapidly, indicating that the water "held up" in the deaerator vessel is now falling into the storage tank. Level has crossed the setpoint, so the condensate valves start to close. This, in coincidence with the admission of poison prevent steam, will bring an end to the pressure drop.

After about 5 minutes: Pressure drop ends at about 0 kPa(g) and begins to recover quickly to 69 kPa(g), the setpoint of the poison prevent steam controller. The held up water in the Deaerator vessel is now falling unhindered into the storage tank, so the level rises rapidly to 95%, even though the condensate valves close fully on hard-wired logic at 90%.

After about 8 minutes: After the held up water has fallen, the level in the storage tank begins to fall rapidly (zero condensate flow, 60% feedwater flow). This causes condensate valves to re-open suddenly and initiate a sec-

**At this point, the timing of the execution of the DLC program becomes significant, and may explain the difference between the Unit 8 850 MW(e) load rejection of one month earlier (Figure 4), and the subject event (Figure 3), in terms of the initial condensate flow response. If the program execution timing is such that the sudden increase in feedwater flow is "seen" by DLC, then the condensate valves will respond with a sudden opening. This results in a drop in the condensate line pressure, and the auto starting of the Standby Condensate Extraction Pump. Once the pump is started, flow will remain elevated until DLC integrates the signal down in response to a level error. We believe this to be the case for Figure 3 and not Figure 4.
Figure 3: Unit 8 Load Rejection From 800 MW(e) Resulting in Damage

Figure 4: Unit 8 Load Rejection From 850 MW(e), No Damage or Unusual Indications

Figure 5: Simulator Data - Match With Station Data, Load Rejection from 800 MW(e)

Figure 6: Unit 5 Load Rejection From 930 MW(e), With Modifications Installed
ond cycle similar but less severe than the first.

DEVELOPMENT AND SIMULATOR TESTING OF MODIFICATIONS

Although the details would require fine-tuning at the simulator, it was quickly apparent that the long term solution to the problem would have the following components:

i) To slow the rate of depressurization, impose a software limit on condensate valve opening following a load rejection or turbine trip to limit the inflow of cold condensate early in the transient.

ii) To minimize the severe level control oscillations (and hence pressure oscillations) for this and other transients, optimize the level controller parameters.

iii) To limit the duration of depressurization, raise the setpoint of the poison prevent steam controller to a value closer to the normal extraction steam pressure.

iv) To improve the reliability and availability of poison prevent steam, modify the poison prevent steam isolating valve logic open immediately upon the occurrence of a power-load unbalance.

Condensate Limit Tuning

It was agreed that for at least the short term, a fixed limit on condensate valve opening was to be imposed for a period of time following a load rejection. The extent and duration of the limit were the subject of simulator testing. The rate of depressurization during the event was about 1.8 kPa/s. The limit was selected by successive simulator tests to minimize the rate of depressurization while keeping the minimum level during the transient above about 60%. The limit was finally optimized to be 50% valve opening for three minutes, after which it was ramped off at 15%/min. The limit was similarly ramped off if level dropped below 55%. This reduced the rate of depressurization during simulated load rejections to about 1.0 kPa/s.

A dynamic valve limit based on rate of change of deaerator pressure was tried on the simulator and rejected. The concern was that this term could couple level and pressure in an undesirable way during other plant transients.

DLC Controller Tuning

It was clear from the outset that the parameters in effect made the controller far too level-sensitive, and not flow sensitive enough. The simulator was used to verify that proposed parameter changes would not only improve performance during load rejections, but also provide acceptable control of other plant transients (eg reactor trip, fast reactor start-up, Heater Drains Pump trips, etc).

IMPLEMENTATION AT THE STATION

The changes were installed in Unit 5 in preparation for the load rejection test to be done as part of Unit uprating in the fall of 1987.

Figure 6 shows the results of a full load rejection from 940 MW(e). No banging was observed in the field, and both level and pressure response was very good. The depressurization rate was about 1.1 kPa/s, compared with 1.8 kPa/s for the Unit 8 event.

CONCLUSIONS

Deaerator damage caused by level and pressure transients during load rejections is not a new phenomenon. To our knowledge, internal vessel damage has occurred at Gentilly, Point Lepreau (Reference 4), and Bruce "A", as well as at some thermal plants. The root cause appears to have been the same in all cases. In this instance, there was no internal damage because of design modifications to limit the localized differential pressures by using larger equalizing lines, and to stiffen the tray structures.

While the problem is not unique, we were in the fortunate position of having available a highly qualified team of design and analysis experts in various fields. In addition, we had at our disposal an extremely powerful engineering tool in the form of the training simulator. This combination allowed us to develop, with a high degree of confidence, detailed modifications for immediate implementation in all units. Normally, such modifications could only be considered for implementation after extensive on-line testing.

We feel that we have learned some valuable lessons which could have application in the design of new stations, as well as in existing stations.

1) DLC tuning is the most important modification, as it directly addresses the root cause of the problem. By increasing the flow sensitivity of the controller we avoid adding excessive condensate flow early in the event, thereby minimizing the rate of depressurization while improving level control.

2) The imposition of a valve opening limit following a load rejection is redundant if the controller parameters have been tuned correctly.

3) Modifications to the poison prevent steam system do not limit the severity of the depressurization transient, but can limit the duration of the depressurization period. They are of secondary importance in solving this problem.

4) Increasing dependence on remote indications should not replace field observation.

5) The use of the training simulator (with the correct model!) as an engineering tool
allowed a level of confidence only otherwise gained by extensive on-line commissioning tests.

6) Deaerator damage has happened on too many occasions, for the same reason. The solution lies in correcting the root cause, i.e., limiting the rate of depressurization.

REFERENCES


3) Kula, L.W., Design Engineer, Electrical and Controls Engineering Department, Design and Development Division, Ontario Hydro, Private Communication.


ACKNOWLEDGEMENTS

The work described in this paper was the combined effort of a large group of people, each of whom brought with them expertise which was essential in dealing with the incident. In particular, the following contributions are much appreciated:

Bruce Construction returned the vessel and pipework to their original position within one day. CNS Inspection Services worked around the clock to complete inspections in the minimum time.

I. Kosiba and C. Munro of the Steam Turbine Group, Power Equipment Department - Mechanical performed the stress analysis.

P. Acchione and E. Wicklund of the Instrumentation and Control Department of Design & Development Division performed the analysis and coordinated efforts at the WNTC Training Simulator.

P. Kar made the changes to the simulator model, while J. Kennard reconfigured the simulator, and M. Benjamin assisted with DCC coding changes.
Session 17:
Thermalhydraulics III & Fluid Chemistry

Chairman:
W.I. Midvidy, Ontario Hydro
RÉSUMÉ

La stabilité thermique de la morpholine en milieu aqueux a été étudiée en laboratoire dans une bombe de Parr sous les conditions de températures et pressions du cycle eau-vapeur de la centrale nucléaire Gentilly 2. Les constantes de vitesse de la réaction de décomposition sont respectivement 2.67, 8.73 et 21.25 × 10⁻⁷ s⁻¹ aux températures de 260, 280 et 300°C. La cinétique est du premier ordre et l'énergie d'activation de 131.9 kJ/mol. L'ammoniac, l'éthanolamine, l'amino-2 éthoxy éthanol, la méthylamine, l'éthylamine, l'éthylène glycol et les acides acétique et glycolique sont les principaux composés formés. Un mécanisme de la décomposition de la morpholine en laboratoire est proposé. L'analyse d'échantillons d'eau du cycle thermique de Gentilly 2 indique la présence des composés énumérés précédemment. Les résultats de cette étude sont présentés et discutés.

ABSTRACT

Thermal stability of morpholine in liquid-steam systems at temperatures and pressures close to CANDU-PHW thermal cycle operating conditions has been investigated using a laboratory high-pressure reactor. The measured decomposition rate constants are 2.67, 8.73 and 21.25 × 10⁻⁷ s⁻¹ at 260, 280 and 300°C respectively. The disappearance of morpholine follows first-order kinetics with an activation energy of 131.9 kJ/mole. Ammonia, ethanolamine, 2-(2-aminoethoxy) ethanol, methylvamine, ethvlamine, ethylene glycol, and acetate and glycolic acids are the major breakdown products identified. The proposed decomposition route has been validated in the field by tracking these volatile amines and organic acids in water samples taken at the Gentilly 2 thermal cycle. This paper presents and discusses analytical results.

INTRODUCTION

Morpholine (\(C_4H_9NO\)) is widely used to control two-phase erosion-corrosion in the steam-water cycles of fossil and nuclear power plants. It is usually added to raise the pH of the condensate, feedwater and drainwater to 9.0-9.5 to counter the action of carbon dioxide present. Typical \(C_4H_9NO\) concentrations found in the cycle at Hydro-Québec's Gentilly 2 nuclear power plant (CANDU-PHW, 600 MW) are from 5 to 9 mg/L.

Recently, significant interest in quantifying the concentrations of organic compounds in the steam-water cycles of thermal and nuclear generating stations has arisen. (1-4) Organics can be introduced into the power cycle either unintentionally as a result of system contamination (e.g., oils, greases, humic matter, halogenated hydrocarbons) or intentionally due to the addition of organic chemicals to the system (e.g., morpholine). At high temperatures and pressures, such additives are expected to decompose and the breakdown products may be corrosive for the system components, which may lead to premature equipment failure. Depending on their nature, the breakdown products could also be responsible for increased cation conductivity in steam-generator blowdowns. In such cases, the cation conductivity parameter used to control the system becomes less sensitive for detecting the ingress of other troublesome anionic inorganic species.

According to Samuel (5), aqueous morpholine decomposition at 308°C reaches 40% in 48 h, giving ethanolamine, diethanolamine and some primary amines unconfirmed by analysis. Jacklin (6), using a laboratory-size experimental high-pressure boiler, showed that morpholine added at a rate of 31 mg/L produces 0.27 mg/L of ammonia at 5.5 MPa, 270°C. A recent study by Gilbert and Saleh (7) of the distribution coefficients of morpholine at Gentilly 2 has revealed the presence of ammonia, methylamine and ethylamine in the cycle samples analysed. Burns et al. (8) observed that when morpholine is added to the secondary cycle at the Beaver Valley Unit 1 power plant, acetic and formic acids are formed. Finally, based on their theoretical investigations supported by laboratory experiments, Dauvois et al. (9) proposed a decomposition mechanism involving ammonia, ethylene glycol, and acetic and formic acids as possible end-products.

For a better understanding of the chemical behavior of morpholine in liquid-steam systems at temperatures and pressures close to CANDU-PHW cycle operating conditions (260°C and 4.8 MPa), thermal stability of this additive was investigated using a laboratory high-pressure reactor. The reaction kinetics were studied, and the breakdown products were identified. Water samples collected at different locations of the Gentilly 2 thermal cycle were analysed and the results are reported and discussed.

EXPERIMENTAL

Reactor and procedure

Morpholine solutions used in real systems are in contact with various alloys coated with oxides whose composition depends on the support material and the chemical operating parameters. The complexity of these systems makes it impossible to reproduce the field catalytic conditions in a laboratory. This study was performed using a static 300 mL high-pressure reactor shown schematically in Figure 1. The reactor was made of type 316 stainless steel, one of the materials used in the construction of thermal-cycle components. The ratio of the metal surface to the solution volume was 270 cm²/180 mL at room temperature and the surface was polished prior to testing. For each experiment, 230 mL of a 150 mg/L morpholine solution was put in the reactor. The temperature was ad-
justed at 260, 280 or 300°C and the reaction time fixed between 0.66 and 7 days. The experimental procedure and the chemical products used have been described in an earlier publication. (10)

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**RESULTS AND DISCUSSION**

**Laboratory tests**

Kinetic study. The kinetics of morpholine decomposition was studied at 200, 280 and 300°C in absence of oxygen (1.8, 6.6 and 8.9 MPa respectively). Figure 2 shows curves for In a/a-x vs. the reaction time, where each value corresponds to an individual test. These curves are linear indicating first-order kinetics. The rate constants and the half-lifes ($T_{1/2}$) calculated from these data are presented in Table 1. The Arrhenius plot of In $k_1$(s$^{-1}$) vs. 1/T (K) shown in Figure 3, gives an activation energy of 131.9 kJ/mol. The experimental results of Donaldson et al. (12) are also given in this figure. At a temperature just above the range covered by this study, i.e. 345°C, the rate constant obtained in a static autoclave of Nimonic 80A alloy free of O$_2$ falls precisely on the extension of the kinetic curve of Figure 3. At a very high temperature, i.e. 560°C, the breakdown rate obtained using a flow system equipped with a small-bore stainless-steel reaction coil, is slightly higher than the corresponding value of the extended curve. Comparison of these results reveals that the kinetic parameters determined in this study with no O$_2$ present could be used over a wide range of temperatures. On the other hand, the test performed by Donaldson et al. at 250°C with O$_2$ present shows that the degradation of morpholine is considerably accelerated ($T_{1/2} \approx 0.06$ days) which implies a different breakdown mechanism.

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**Analytical methods**

Table 1 summarizes the instrumental conditions under which morpholine and its breakdown products were quantified by high-performance liquid chromatography (HPLC) and high-pressure ion chromatography (HPIC). The volatile amines and organic acids contained in laboratory degraded-morpholine solutions were analysed by direct injection of aliquots in the chromatographic systems. The volatile amines contained in samples collected at Gentilly 2 were concentrated prior to their analysis. Details on analytical techniques are reported elsewhere. (10-11)

---

**TABLE 1: SUMMARY OF THE INSTRUMENTAL CONDITIONS FOR THE QUANTIFICATION OF MORPHOLINE AND ITS BREAKDOWN PRODUCTS.**

<table>
<thead>
<tr>
<th>Conditions</th>
<th>HPLC</th>
<th></th>
<th>HPIC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Products quantified</td>
<td>morpholine and volatile amines*</td>
<td>ethylene glycol</td>
<td>organic acids</td>
</tr>
<tr>
<td>Analytical column</td>
<td>Waters $\mu$Bondapak C$_18$, 3.9 x 300 mm</td>
<td>Dionex MPIC-NSI 4 x 200 mm</td>
<td>Dionex HPICE-AS1 4 x 250 mm</td>
</tr>
<tr>
<td>Suppressor</td>
<td>-</td>
<td>Dionex CFS-2, regenerant: 0.04 mol/L Ba(OH)$_2$, 4 mL/min</td>
<td>-</td>
</tr>
<tr>
<td>Concentrator</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Mobile phase</td>
<td>A: water B: ethanol 0-30 min 75% A 30-45 min 75 to 47% A 45-70 min 47% A 70-95 min 47 to 0% A</td>
<td>0.002 mol/L C$_6$H$_5$SO$_3$H 0.0005 mol/L HCl 0.005 mol/L C$_6$H$_5$SO$_3$H</td>
<td>0.6</td>
</tr>
<tr>
<td>Flow rate, mL/min</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Sample volume, µL</td>
<td>20</td>
<td>650</td>
<td>650</td>
</tr>
<tr>
<td>Detector</td>
<td>ultraviolet at 425 mm</td>
<td>differential refractometer</td>
<td>conductivity</td>
</tr>
<tr>
<td>Temperature, °C</td>
<td>30</td>
<td>ambient</td>
<td>ambient</td>
</tr>
</tbody>
</table>

* Aliphatic amines modified by pre-column derivatization with dabsyl chloride

** Organic acids contained in the Gentilly 2 samples analyzed after preconcentration using a Dionex AG4A column

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FIGURE 2: CONCENTRATION OF MORPHOLINE VS. REACTION TIME AT 260, 280 and 300°C

FIGURE 3: ARRHENIUS PLOT OF FIRST-ORDER RATE CONSTANT FOR THE DECOMPOSITION OF MORPHOLINE. •: THIS WORK. △: LITERATURE

TABLE 2: KINETIC PARAMETERS OF THE THERMAL DECOMPOSITION OF MORPHOLINE

| Temperature °C | Test duration (days) | Morpholine | Rate constants (κ) | n_max = (ln(2))/κ
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>260</td>
<td>3</td>
<td>149.1</td>
<td>5.93</td>
<td>4.33</td>
</tr>
<tr>
<td>5</td>
<td>133.6</td>
<td>6.69</td>
<td>0.12</td>
<td>0.19</td>
</tr>
<tr>
<td>7</td>
<td>127.6</td>
<td>10.80</td>
<td>0.48</td>
<td>0.70</td>
</tr>
<tr>
<td>280</td>
<td>2.26</td>
<td>129.8</td>
<td>7.98</td>
<td>0.48</td>
</tr>
<tr>
<td>267</td>
<td>122.6</td>
<td>9.27</td>
<td>0.72</td>
<td>1.75</td>
</tr>
<tr>
<td>4</td>
<td>115.1</td>
<td>11.17</td>
<td>1.08</td>
<td>2.41</td>
</tr>
<tr>
<td>6</td>
<td>94.9</td>
<td>5.77</td>
<td>1.59</td>
<td>2.97</td>
</tr>
<tr>
<td>300</td>
<td>0.66</td>
<td>135.3</td>
<td>2.51</td>
<td>0.18</td>
</tr>
<tr>
<td>2</td>
<td>103.8</td>
<td>2.58</td>
<td>5.10</td>
<td>2.82</td>
</tr>
<tr>
<td>3.2</td>
<td>86.3</td>
<td>6.76</td>
<td>2.82</td>
<td>4.78</td>
</tr>
<tr>
<td>5</td>
<td>60.8</td>
<td>11.93</td>
<td>0.69</td>
<td>11.17</td>
</tr>
</tbody>
</table>

Breakdown products. The major breakdown products resulting from the decomposition of C_4H_9NO were ammonia, ethanamine, 2-(2-aminoethoxy) ethanil, methyamine, ethyamine, ethylene glycol, and acetic and glycolic acids. Their concentration is given in Table 3 and typical analyses by HPLC and HPIC are presented in Figures 4, 5 and 6. Products with a retention time (T_r) of between 0 and 10 min in Figure 4 are relevant to the dabsyl chloride used for the amine pre-column derivatization; those with a T_r of 10 to 45 min were not identified. The dabsylated amines and morpholine are eluted between 45 and 60 min. Under the analytical conditions of Figure 5, only the ethylene glycol was determined with a retention time of 3.5 min. The chromatogram of Figure 6 shows the anion form of glycolic, acetic and carbonic acids with T_r of 12, 16 and 24 min respectively. The presence of formic acid (T_r = 13 min), as suggested by Burns et al. (8) and Dauvois et al. (9), was not detected.

Material balance. Table 4 presents the material balances for carbon and nitrogen resulting from the morpholine decomposition tests performed at the maximum temperature of a CANDU-PHWR cycle. The weight fraction of the element (C or N) in the compounds analyzed (Table 3) were used to calculate the percentage recoveries: 100% recovery corresponds to the amount of C or N of the initial morpholine. Despite the number of analytical techniques involved and their 5% precision, the C-balances are close to the theoretical values. For the N-balances, values slightly over 100% were obtained. In view of these results, vapor-phase losses of the volatile amines produced by the C_4H_9NO decomposition can be considered negligible under the present analytical procedure.
FIGURE 1: CHROMATOGRAM OF HPLC SEPARATION OF DABSYL DERIVATIVES OF VOLATILE AMINES PRESENT IN LABORATORY DEGRADED-MORPHOLINE SOLUTIONS: 1) AMMONIA, 2) ETHANOLAMINE, 3) 2-(2-AMINOETHOXY) ETHANOL, 4) METHYLAMINE, 5) ETHYLAMINE AND 6) MORPHOLINE

FIGURE 2: CHROMATOGRAM OF HPLC SEPARATION OF ORGANIC ACIDS PRESENT IN LABORATORY DEGRADED-MORPHOLINE SOLUTIONS: 1) GLYCOLIC ACID, 2) ACETIC ACID AND 3) CARBONIC ACID

FIGURE 3: CHROMATOGRAM OF HPLC SEPARATION OF ETHYLENE GLYCOL AND VOLATILE AMINES PRESENT IN LABORATORY DEGRADED-MORPHOLINE SOLUTIONS: 1) ETHYLENE GLYCOL, 2) UNKNOWN, 3) AMMONIA, 4) ETHANOLAMINE AND METHYLAMINE, 5) ETHYLAMINE AND 6) MORPHOLINE

TABLE 1: CARBON AND NITROGEN MATERIAL BALANCES OF THERMAL-DECOMPOSITION PRODUCTS OF MORPHOLINE AFTER TREATMENT AT 200°C

<table>
<thead>
<tr>
<th>Test duration days</th>
<th>Element</th>
<th>% mol/L Recovered</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>C</td>
<td>100</td>
</tr>
<tr>
<td></td>
<td>N</td>
<td>100</td>
</tr>
<tr>
<td>5</td>
<td>C</td>
<td>101</td>
</tr>
<tr>
<td></td>
<td>N</td>
<td>108</td>
</tr>
<tr>
<td>7</td>
<td>C</td>
<td>101</td>
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<tr>
<td></td>
<td>N</td>
<td>114</td>
</tr>
</tbody>
</table>

Reaction scheme. Figure 7 proposes a reaction scheme that involves two types of scission: formation of diethanolamine following scission of the C-O bond and formation of 2-(2-aminoethoxy) ethanol due to scission of the C-N bond. HPLC analysis under the conditions described in Table 1 revealed the absence of diethanolamine (according to a detection limit of 50 μg/L) in all the degraded-morpholine solutions, contrary to what was proposed by Samuel (5) and Daurois et al. (9) Presence of appreciable amounts of 2-(2-aminoethoxy) ethanol in all the reaction mixtures indicates that solvolysis of the C-N bond would be the preferable path. This molecule is unstable under the present thermal conditions and leads to the formation of ethanolamine and ethylene glycol as demonstrated in the laboratory. (13) The thermal stability of ethanolamine was also examined in the laboratory. The test performed at 300°C over a 5-day reaction time have shown that this molecule decomposed to give ammonia, methylethylamine, ethylamine and ethylene glycol as the major breakdown products. (18) Hydrolysis of ethylamine followed by oxid-
tion of the ethanol formed could explain the presence of acetic acid. Formation of glycolic acid could be attributed to the oxidation of ethylene glycol in accordance with Rossiter et al. (14) Formic acid which was not observed in laboratory tests but appeared in the cycle samples could result from the oxidation of glycolic acid followed by thermal degradation of oxalic acid.

Effect of temperature and reaction time. At 260°C, around 7% of the morpholine had decomposed after 3 days of testing, the reaction being limited to the formation of 2-(2-aminoethoxy) ethanol and ethanolamine, and the oxidation of ethylene glycol. As the reaction time progressed, decomposition intensified (e.g., around 15% after 7 days) and the thermal decomposition of ethanolamine increased (Table 3). The half-life of the morpholine solution at this temperature is 30 days. At 300°C, the morpholine decomposition increases exponentially with the reaction time (e.g., around 8% after 0.66 days and 59% for 5 days), and this is reflected by its $\tau_{1/2}$, which is 4 days. The results of Table 3 indicate that the formation of methyamine, ethylamine and ammonia is being the preferred path when the temperature increases from 260 to 300°C.

Field measurements

Process sampling points. A schematic diagram of the Gentilly 2 thermal cycle is presented in Figure 8. Since 1984 the feedwater and steam generators have been operating with a morpholine treatment without any hydrazine injection. Details on the cycle chemistry specifications are given elsewhere. (15) The process sampling points and the device used to collect samples are shown in Figures 8 and 9.

---

**FIGURE 7: PROPOSED REACTION SCHEME FOR THE THERMAL DECOMPOSITION OF MORPHOLINE**

**FIGURE 8: SCHEMATIC DIAGRAM OF THE GENTILLY 2 THERMAL CYCLE SHOWING THE PROCESS SAMPLING POINTS: A) COMBINED STEAM-GENERATOR BLOWDOWN, B) MAIN-STEAM, C) SEPARATOR DRAINS, D) CONDENSATE PUMP DISCHARGE, E) MAKEUP WATER AND F) HIGH-PRESSURE HEATER OUTLET**
ethanolamine in these samples confirm that under cycle conditions, solvolysis of the C-N bond is the preferential path. If diethanolamine forms at all, its concentration must be below the detection limit, which is 12 \( \mu g/L \) in these preconcentrated samples. Formation of ethylenediamine and ethylene glycol was not revealed since 1) HPLC on preconcentrated samples (factor 100) does not resolve ethylenediamine when morpholine content is high and 2) the ethylene glycol formed is below the detection limit in HPLC (\( \approx 2 \) \( mg/L \)).

The study of organic acids by HPLC and their detection by ultraviolet spectrometry (Figures 11 and 12) or by conductimetry (Figure 12) showed that cycle water contains non-ionic or weakly-ionized compounds. The origin and identity of these compounds are as yet unknown and will be studied in the future.

**FIGURE 9: SAMPLING DEVICE USED TO COLLECT SAMPLES FOR THE DETERMINATION OF VOLATILE AMINES (SAMPLING POINT 1) AND ORGANIC ACIDS (SAMPLING POINT 2). PANEL LINES DEFINED IN FIGURE 8**

Analysis. Cycle process samples, makeup water sample (demineralized water) and a morpholine solution of 5000 \( \mu g/L \) were analysed for their volatile amines and organic acid contents. The results are presented in Table 5. Figures 10 and 11 show the HPLC and IIPIC chromatograms obtained. These chromatograms have a different pattern from those presented at Figures 4 and 6 since other analytical conditions to be described elsewhere were retained. Analysis of the demineralized water indicates the presence of 16 \( \mu g/L \) of ammonia and traces of organic acids, whereas the morpholine solution before being added to the cycle contained 21 \( \mu g/L \) of ammonia and 61 \( \mu g/L \) of 2-(2-aminoethoxy) ethanol. On the other hand, analysis of cycle samples systematically shows concentrations of volatile amines and organic acids, except for ammonia, that exceed those found in the demineralized water and in the initial morpholine solution. The results confirm the morpholine's instability in the thermal cycle and partly support observations made in the laboratory. Besides, the presence of 2-(2-aminoethoxy) ethanol and absence of diethanolamine in these samples confirm that under cycle conditions, solvolysis of the C-N bond is the preferential path. If diethanolamine forms at all, its concentration must be below the detection limit, which is 12 \( \mu g/L \) in these preconcentrated samples. Formation of ethylenediamine and ethylene glycol was not revealed since 1) HPLC on preconcentrated samples (factor 100) does not resolve ethylenediamine when morpholine content is high and 2) the ethylene glycol formed is below the detection limit in HPLC (\( \approx 2 \) \( mg/L \)).

**TABLE 5: DETERMINATION OF ORGANIC ACIDS AND VOLATILE AMINES PRESENT AT DIFFERENT LOCATIONS OF THE GENTILLY 2 THERMAL CYCLE - CONCENTRATIONS EXPRESSED IN \( \mu g/L \)**

<table>
<thead>
<tr>
<th>Process sampling points</th>
<th>Glycolate</th>
<th>Formate</th>
<th>Acetate</th>
<th>Ammonia</th>
<th>Ethanolamine</th>
<th>2-(2-aminoethoxy) ethanol</th>
<th>Methylamine</th>
<th>Morpholine</th>
</tr>
</thead>
<tbody>
<tr>
<td>Combined steam-generator blowdown</td>
<td>69</td>
<td>10</td>
<td>15</td>
<td>11</td>
<td>51</td>
<td>198</td>
<td>24</td>
<td>4750</td>
</tr>
<tr>
<td>Main steam</td>
<td>*</td>
<td>*</td>
<td>19</td>
<td>40</td>
<td>14</td>
<td>192</td>
<td>56</td>
<td>5260</td>
</tr>
<tr>
<td>Separator drains</td>
<td>4</td>
<td>6</td>
<td>114</td>
<td>50</td>
<td>81</td>
<td>157</td>
<td>32</td>
<td>6020</td>
</tr>
<tr>
<td>Condensate pump discharge</td>
<td>*</td>
<td>*</td>
<td>11</td>
<td>45</td>
<td>7</td>
<td>178</td>
<td>66</td>
<td>5410</td>
</tr>
<tr>
<td>High-pressure heater outlet</td>
<td>*</td>
<td>*</td>
<td>21</td>
<td>15</td>
<td>10</td>
<td>157</td>
<td>39</td>
<td>5520</td>
</tr>
<tr>
<td>Makeup water</td>
<td>*</td>
<td>*</td>
<td>*</td>
<td>16</td>
<td>*</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>Morpholine solution</td>
<td>21</td>
<td>*</td>
<td>*</td>
<td>*</td>
<td>64</td>
<td>*</td>
<td>*</td>
<td>5000</td>
</tr>
</tbody>
</table>

* Trace < 2\( \mu g/L \)
** Non-detected
FIGURE 10: CHROMATOGRAMS OF HPLC SEPARATION OF THE DABSYL DERIVATIVES OF VOLATILE AMINES PRESENT AT DIFFERENT LOCATIONS OF THE GENTILLY 2 THERMAL CYCLE: 1) AMMONIA, 2) ETHANOLAMINE, 3) 2-(2-AMINOETHOXY) ETHANOL, 4) UNKNOWN, 5) METHYLAMINE AND 6) MORPHOLINE

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FIGURE 11: CHROMATOGRAMS OF HPLC SEPARATION OF ORGANIC ACIDS PRESENT AT DIFFERENT LOCATIONS OF THE GENTILLY 2 THERMAL CYCLE: 1) GLYCOLATE, 2) FORMATE AND 3) ACETATE

FIGURE 12: TYPICAL CHROMATOGRAMS OF HPLC SEPARATION OF ORGANIC ACIDS PRESENT IN A WATER SAMPLE OF THE GENTILLY 2 THERMAL CYCLE: 1) GLYCOLATE, 2) FORMATE AND 3) ACETATE. DETECTION BY ULTRAVIOLET SPECTROMETRY AND CONDUCTIMETRY

TABLE 6: DISTRIBUTION COEFFICIENTS OF AMINES AND ORGANIC ACIDS PRESENT IN THE GENTILLY 2 THERMAL CYCLE UNDER NORMAL OPERATING CONDITIONS

<table>
<thead>
<tr>
<th>Compounds</th>
<th>$\delta_{SG}$</th>
<th>$\delta_{HP}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Glycolate</td>
<td>&lt; 0.03</td>
<td>&lt; 0.5</td>
</tr>
<tr>
<td>Formate</td>
<td>&lt; 0.2</td>
<td>&lt; 0.33</td>
</tr>
<tr>
<td>Acetate</td>
<td>1.27</td>
<td>0.17</td>
</tr>
<tr>
<td>Ammonia</td>
<td>3.64</td>
<td>0.80</td>
</tr>
<tr>
<td>Ethanolamine</td>
<td>0.27</td>
<td>0.17</td>
</tr>
<tr>
<td>2-(2-aminoethoxy) ethanol</td>
<td>0.97</td>
<td>1.22</td>
</tr>
<tr>
<td>Methylamine</td>
<td>2.33</td>
<td>1.75</td>
</tr>
<tr>
<td>Morpholine</td>
<td>1.10</td>
<td>0.87</td>
</tr>
</tbody>
</table>
that these volatile compounds are partly removed from the cycle by the deaerator. 2-(2-aminoethoxy) ethanol presents a $\delta^{SG}$ and $\delta^{HP}$ of 0.97 and 1.22, which indicate that the distribution of the molecule between the vapor phase and the different cycle condensates is more or less equal. The morpholine distribution coefficients are in agreement with the previous results discussed in detail in another paper. (7)

CONCLUSION

The main conclusions of this study are:
1. Morpholine is an unstable additive under the thermal-cycle conditions at Gentilly 2 (260°C and 4.8 MPa).
2. Decomposition proceeds preferentially by scission of the C-N bond to produce 2-(2-aminoethoxy) ethanol, which in turn decomposes to give ethanalamine, ethylene glycol, methyamine, aminomethane, and some organic acids.
3. Under conditions at Gentilly 2, the glycolic and acetic acids are the major organic acids formed.
4. Sufficient evidence exists to indicate that morpholine could be responsible for most of the ionic organic impurities present in the Gentilly 2 thermal cycle.

ACKNOWLEDGMENTS

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REFERENCES


ABSTRACT

The Electric Power Research Institute has developed the CHEC™ and CHECMATE™ computer codes for predicting erosion-corrosion in steel piping components. These programs assist the utility industry in planning and implementing their inspection programs. The industry inspection program and how CHEC™ and CHECMATE™ are used to optimize inspection planning are described here.

INTRODUCTION

Erosion-corrosion is a flow-accelerated corrosion process that leads to wall thinning (metal loss) of steel piping exposed to flowing water or wet steam. The rate of metal loss depends on a complex interplay of several parameters. These parameters include water chemistry, material composition, and hydrodynamics. Erosion-corrosion of plant piping can lead to costly outages and repairs, and can raise concerns about plant reliability and safety. Pipe wall degradation rates as high as 1.5 mm/year have occurred, resulting in pipe ruptures at both fossil and nuclear plants.

Carbon steel pipes that carry wet steam are especially susceptible to erosion-corrosion, representing an industry-wide utility problem.

In mid-1987, the utility industry in the United States instituted an inspection program to reduce the risk of pipe ruptures caused by erosion-corrosion. To support this effort, EPRI has developed the CHEC™ (Chexal Jorowitz Erosion Corrosion) and CHECMATE™ (Chexal Jorowitz Erosion Corrosion Methodology for Analyzing Two-phase Environment) codes for predicting erosion-corrosion in steel piping components. CHEC (1) performs erosion-corrosion analysis of single-phase piping systems. CHECMATE (2) performs erosion-corrosion analysis of two-phase piping systems.

The programs were developed specifically to assist the utility industry in planning and implementing inspection programs to address erosion-corrosion in power plants, and to do so in a manner which will reduce the cost and increase the effectiveness of those inspections significantly. They also can be used to evaluate the impact of changes in operating conditions on erosion-corrosion and to evaluate the potential of new or revised piping designs to avoid erosion-corrosion damage.

This paper describes the industry inspection program and how CHEC and CHECMATE are used to optimize inspection planning.

THE INDUSTRY INSPECTION PROGRAM

Since the Surry Power Station feedwater pipe rupture in December 1986, the industry has worked steadily to develop and implement a monitoring program to prevent the rupture of piping due to erosion-corrosion. In March 1987, the Institute of Nuclear Power Operations (INPO) issued Significant Operating Experience Report (SOER 87-3) which recommended that a continuing program be established at all U.S. nuclear power plants, including both analyses to predict wear rates and regular inspections.

The Nuclear Management and Resource Council (NUMARC) and EPRI developed a resolution approach and provided guidelines for utility implementation (3, 4, 5). These guidelines were made available to the industry in June of 1987. The NUMARC/EPRI Guidelines recommended to:

1. Conduct appropriate analysis and a limited but thorough initial inspection of susceptible single-phase piping.
2. Determine extent of thinning and repair or replace as necessary.
3. Perform follow up inspections to confirm or quantify thinning and take longer term corrective action.

U.S. utilities conducted the initial inspections of these guidelines during 1987 and 1988. The U.S. Nuclear Regulatory Commission (USNRC) monitored the results of these inspections, and in May 1989 issued Generic Letter 89-08 essentially requiring that utilities:

1. Implement long term erosion-corrosion monitoring program.
2. Include all high energy carbon steel systems.
3. Include both single and two-phase systems.
4. The NUMARC/EPRI or other equally effective program be utilized.

The American Society of Mechanical Engineers (ASME), in response to a request from the USNRC, is developing rules for monitoring safety-related piping for erosion-corrosion (6). When approved, these rules will be incorporated into Section XI of the ASME Boiler and Pressure Vessel Code, Rules for Inservice Inspection of Nuclear Power Plant Components. These rules will address three main issues: (1) what specific components should be examined and how often, (2) what non-destructive examination (NDE) methods and techniques should be used and (3) how components should be evaluated for...
acceptable service. They are expected to be comparable to the NUMARC/EPRI guidelines.

EPRI's key role in the NUMARC/EPRI program has been to provide the technical tools, CHEC and CHECMATE, to optimize the planning and implementation of the inspection program. CHEC was developed and first released in July, 1987 to help utilities respond to the Surry piping failure. Utilities had an urgent need for a tool to assess the susceptibility of their plants to single-phase erosion-corrosion for planning and evaluating initial inspections.

CHEC has been updated several times since the first release. CHEC capabilities were expanded to include analysis of two-phase systems with the release of CHECMATE in April, 1989. These programs are being utilized by nearly all of the nuclear utilities in the U.S., and many utilities with fossil fuel facilities. They have also been licensed to utilities and other organizations in Europe and Japan.

THE EROSION-CORROSION PHENOMENA

Flow-accelerated corrosion is the process by which carbon or low alloy steel components lose material through the dissolution of the poorly adherent magnetite layer. It is dependent on water chemistry - pH, oxygen and temperature; material composition - chromium, copper and molybdenum content; and hydrodynamics - flow velocity, geometry and steam quality. This phenomenon normally occurs in flowing, deoxygenated water with a pH between 7.0 and 9.5. It is not a classical erosion process in that the material loss is not caused by a mechanical process. There is not a threshold velocity below which no material is lost, and above which there is extensive damage.

A large body of experimental work has identified several key variables that influence the rate of attack. These variables are listed below with an indication of how they impact the material loss behavior. (7)

<table>
<thead>
<tr>
<th>Variable</th>
<th>Erosion-Corrosion Increases If Variable is</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fluid velocity</td>
<td>Higher</td>
</tr>
<tr>
<td>Fluid pH level</td>
<td>Lower</td>
</tr>
<tr>
<td>Fluid oxygen content</td>
<td>Lower</td>
</tr>
<tr>
<td>Fluid temperature</td>
<td>250-400°F</td>
</tr>
<tr>
<td>Steam quality</td>
<td>0.1-0.9</td>
</tr>
<tr>
<td>Component geometry</td>
<td>Such as to create more turbulence</td>
</tr>
<tr>
<td>Component chromium content</td>
<td>Lower</td>
</tr>
<tr>
<td>Component copper content</td>
<td>Lower</td>
</tr>
<tr>
<td>Component molybdenum content</td>
<td>Lower</td>
</tr>
</tbody>
</table>

The complexities of these variables and their interrelations are such that an empirical model which considers all of the variables is required to make erosion-corrosion predictions with accuracy. This predictive capability helps avoid non-productive inspection efforts.

Single-phase erosion-corrosion is most likely to occur in main feedwater piping, condensate, piping and heater drain piping. Areas demonstrated to be especially susceptible include bypass lines such as for recirculation flow around pumps and control valves. It also is likely to occur downstream of flow control valves (angle valves in particular) and in elbows in close proximity to other fittings. Instances of single-phase erosion-corrosion have also been reported in other fittings such as at the small diameter end of diffusers and J-tubes in steam generators. Two-phase erosion-corrosion often occurs in main steam piping, reheater steam piping, extraction steam piping, moisture separator reheater drain piping, blowdown piping and feedwater heater drain piping and downstream of leaky valves.

THE CHEC/CHECMATE MODEL

The CHEC and CHECMATE model formulation was developed empirically due to the complex interrelationship between these parameters. A large database was assembled from various laboratories worldwide, and an optimum model was developed using an iterative procedure. This model followed all of the experimental trends, and correlated well with the bulk of the laboratory data. The model was further refined by comparing the predictions of the model with the erosion-corrosion inspection data obtained from several nuclear power plants, and by comparison with further laboratory research (particularly to take into account various geometrical effects).

The general formulation of the model is a series of factors which, when multiplied together, yield the predicted erosion-corrosion rate. Since some of the factors are interrelated, the model is not linear. The formulation is as follows:

$$ E = F_1(T) \cdot F_2(AC) \cdot F_3(MT) \cdot F_4(O_2) \cdot F_5(pH) \cdot F_6(G) \cdot F_7(a) $$

where:

- $F_1(T)$ = factor for temperature effect
- $F_2(AC)$ = factor for alloy content effect (chromium, copper and molybdenum content)
- $F_3(MT)$ = factor for mass transfer effect (flow rate, pipe diameter)
- $F_4(O_2)$ = factor for oxygen effect
- $F_5(pH)$ = factor for pH effect (amine type)
- $F_6(G)$ = factor for geometry effect
- $F_7(a)$ = Factor for void fraction (Pressure, liquid and vapor mass flow rate, pipe diameter, pipe orientation)

CHEC and CHECMATE are complete analytical tools that are flexible and easy to use. A User Interface System is utilized to provide an integrated environment for data entry, storage and retrieval, analysis, and display and evaluation of results. It guides the user through menus, prompts and help messages to build and modify input data files, review results and interface with the various analytical modules. The output provided consists of the predicted average and current erosion rates, predicted wear and thickness and the remaining service life for every fitting analyzed.

CHECMATE also contains a chemistry analysis feature which allows a user to define the configuration of a plant, calculate and display the pH or dissolved oxygen around the complete steam cycle. This accommodates for the effects of temperature, single-phase flow and partitioning in two-phase flow for controlling amines. The user defines the plant...
configuration from the information found on a heat-
balance diagram ("typical" plant configurations are
supplied to the user). Once the configuration is
defined, the user can also perform sensitivity
analyses of the system to address questions such as:

- What is the pH in the first high pressure
  extraction line?
- What is the change in pH in the heater drain
tank if the amine is changed from ammonia to
  morpholine?
- What is the impact of Hydrogen Water Chemistry
  on the dissolved oxygen level in the heater
  drains?

Also included is a network flow analysis feature
which permits the user to enter certain component
geometric data and loss coefficient information and
obtain flow, pressure and steam quality information
for components in a selected line. Piping systems
of interest are typically small diameter lines not
generally found on Heat Balance diagrams, and may
include a variety of fittings, such as elbows, tees,
flow limiting orifices, valves or other geometries
which may cause local irreversible pressure losses.

The Network Analysis feature computes steady-
state flow, pressure and quality for the components
of a variety of defined flow network configurations.
It accounts for energy transfer from the
network to the environment and the possibility of
critical flow in one or more branches.

CHEC/CHECMATE ACCURACY

The purpose of CHEC and CHECMATE is to predict
actual plant performance. To validate these pro-
grams, actual plant inspection data were obtained
from several operating U.S. nuclear plants. Pre-
dicted values were compared with measured values at
the point of maximum wear. Only raw NDE data were
used for this validation process. Figure 1 provides
a summary of measured versus predicted pipe wall
thickness. This validation has demonstrated a wall
thickness predictive capability in the range of ±
20%.

INSPECTION PLANNING WITH CHEC AND CHECMATE

Two-phase erosion-corrosion has been a known
entity for some time. However, single-phase
erosion-corrosion was an unexpected development
affecting plant and personnel safety. As such, the
plant engineer had to develop and establish
procedures to:

1. Identify the locations most susceptible to
erosion-corrosion.
2. Perform inspections.
3. Evaluate if an erosion-corrosion problem
   exists and the extent of it.
4. Rectify the problem if it does exist.
5. Utilize the inspection data to determine when
   and where to inspect next.
CHEC Pass 1 results provide information for planning the initial inspection. Pass 1 ranks the susceptibility of all components to erosion-corrosion, and from these rankings develops a list of ten recommended locations for the initial inspection. The output tables of component rankings in order of relative erosion-corrosion rate and relative total lifetime can be used to select alternate locations to facilitate inspection access, etc., or if the user desires, to provide more locations than the 10 minimum. Selection of 5 additional locations should be based upon the operator's knowledge of the plant and engineering judgment. These component rankings can be used to assist in that selection process.

CHEC Pass 1 results also include warnings regarding potentially high erosion rates and short remaining lifetime. These warnings may be the result of plant erosion-corrosion problems or the result of conservative plant modeling, such as using worst case values for plant pH or oxygen concentration. Inadvertent modeling errors may also cause CHEC to issue these warnings. If CHEC Pass 1 analysis indicates potential plant erosion-corrosion problems, the user should be aware that inspection expansion may be required once the initial inspection results are reviewed.

Once the initial inspections of the minimum 15 components are complete, the wall thickness measurements results are entered into CHEC for a Pass 2 analysis. The Pass 2 output can be used to assist in selections for sample expansion, if required, and in planning for subsequent inspections. Utilizing this inspection data input, Pass 2 becomes a predictive model of the plant. Analysis results include predicted wear, total wear and, using current erosion rate, the predicted remaining time to reach \( t_{\text{min}} \) for all of the components modeled.

If unacceptable erosion-corrosion is indicated on any of the initially inspected components, the inspection must be expanded. The additional components should include identical components in similar trains, components in the same train in the near vicinity, and any components for which erosion-corrosion is a concern through the next plant outage. The Pass 2 output can be utilized to help identify components for an expanded sample by comparing predicted erosion-corrosion rates and total wear of similar components, and identifying other components with similarly high rates. Predictions of time to \( t_{\text{min}} \) can identify which components potentially could be of concern before the next outage.

After the user has completed the initial inspection, including Pass 2 analysis, it is then possible to design an inspection plan for the future. The long term goal for this inspection plan should be to minimize the number of required inspections, while maintaining plant safety and reliability. Future inspection plans should be based on the amount of wear found, the age of the plant and the time that has elapsed since the last inspection. CHEC Pass 2 results help the user to determine which components to inspect and the timing of those future inspections.

If a large amount of wear was found in the initial inspection, the inspection program should continue to inspect a large number of components at intervals short enough to ensure plant design criteria (whether based on \( t_{\text{min}} \) or \( t_{\text{nom}} \)) are adhered to. Such a plan should also include remedial repair and preventive measures.

If a small amount of wear was found, inspection plans can cover a small number of components (at least ten) at reasonable intervals based on the plant's acceptance criteria.

For two-phase analysis, individual models must be developed for all lines that need to be analyzed. The lines that are typically analyzed with CHEC are the reheat, extraction and MSR drain lines. The lines experiencing significant erosion-corrosion in the past should be given a higher priority for analysis.

In contrast to CHEC, CHECMATE has only one analysis mode, and does not provide a specific list of recommended inspection locations. Still, the analysis approach is the same. CHECMATE output consists of three ranking lists that can be used for planning the initial inspection. The first list is a ranking by erosion rate for all components in the line. The second list ranks these components by the predicted time to reach a minimum acceptable thickness. The third list provides the predicted wear and thickness for all components.

Based on these lists and past erosion-corrosion experience with a given line, the engineer has enough information to determine the inspection locations for the line. The results of this inspection can then be entered into CHECMATE to establish a long term inspection model. CHECMATE also has the capability to store and utilize data for those components that have been repaired or replaced previously. This feature accounts for the correct in-service time and determines the remaining service life accurately.

PERFORMING INSPECTIONS

NDE utilizing ultrasonic testing (UT) devices on a grid system marked on the component is the predominant approach being employed. EPRI recommends that the number of piping components to be inspected be optimized, and that a full coverage grid be used on each of those components. A full coverage grid provides a good data base to determine maximum wear, as well as a good baseline for future inspections. The other advantage of using a full coverage grid for component inspections is that this method ensures that the area of maximum wear is not missed. Partial grid techniques can not guarantee the detection of maximum wear on a component. Our experience has shown that it is better to inspect a limited number of components thoroughly than to do minimal inspection of an excessive number of components.

Measurements are made where lines of the grid cross, with scanning in between with large grid spacings to ensure the values at the measurement points represent the area (see Figure 2). Inspection grids should be set up to allow the reinspection of identical grid points during subsequent outages. Reinspection of identical grid points will allow for the accurate determination of component wear between inspections. The inspection information must be legibly and completely documented to allow for subsequent analysis. The recording and storage of inspection data on personal

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computer based systems allow easy and rapid interpretation of large amounts of inspection data.

Results of EPRI tests, as well as evaluation of data from a large number of power plant inspections, show that erosion-corrosion can extend into the piping downstream of a component (except diffusers and expanding elbows). Consequently, EPRI recommends that the inspection cover a component from 2 inspection grids upstream of the toe of the upstream weld to a minimum of two diameters beyond the toe of the downstream weld. Test results also show that for diffusers and expanding elbows, erosion-corrosion can occur upstream of the component as well. EPRI recommends that for these components the grid should be extended also a minimum of two diameters upstream of the toe of the upstream weld.

During the inspection process, the engineer performing the CHEC/ CHECMATE analysis should work closely with the personnel performing the component inspections. Incorrect NDE readings can have a negative effect on the correlation between predicted and measured wear. This effect diminishes as the number of inspected components increases. If an NDE reading is significantly different than the neighboring values, this reading may be incorrect and should be verified. The analyst and the NDE personnel should work closely together to quickly identify and verify suspected areas.

CHEC/ CHECMATE AS A DESIGN TOOL

CHEC and CHECMATE assist utilities in optimizing inspection activities, and can also be used as design tools to evaluate changes. This is a powerful and unique capability that can result in significant cost saving.

For example, a utility may have experienced severe single-phase erosion-corrosion damage in their plant which requires remedial action. Their options include:

- Changing the pipe and component material
- Reducing the fluid velocity by increasing the pipe size
- Increasing the pH
- Changing the pH controlling amine
- Redesigning the problem areas to avoid an adverse geometric situation

CHEC and CHECMATE allow the user to easily and quickly analyze each of the above options to determine the most cost effective approach.

EXAMPLE:

Assume that a plant has operated for fifteen years, and discovered that there is severe erosion-corrosion damage. Let us further assume that the plant was using ammonia as the pH controlling amine and that the pH was set at 9.2. The engineering staff have come up with the following options to allow the plant to operate for an additional 25 years:

Option 1 - increase the pH from 9.2 to 9.6 using the same amine
Option 2 - change the amine to morpholine and keep the pH at 9.2
Option 3 - keep the chemistry the same and change the pipe and component material to P11 (higher chromium)
Option 4 - keep the chemistry and the pipe material the same and increase the pipe diameter by one size.

To evaluate the relative effectiveness of these options, the designer need only make a Pass 1 CHEC input file with typical values for the current system, and enter the options as perturbations from the base case. Presented below is a table of relative time to reach minimum allowable wall thickness for the base case and each of the options taken directly from a Pass 1 output table.

<table>
<thead>
<tr>
<th>COMPONENT</th>
<th>RELATIVE TIME TO MINIMUM WALL THICKNESS</th>
</tr>
</thead>
<tbody>
<tr>
<td>BASE CASE</td>
<td>1.000</td>
</tr>
<tr>
<td>INCREASED DIA</td>
<td>1.192</td>
</tr>
<tr>
<td>MORPHOLINE</td>
<td>1.654</td>
</tr>
<tr>
<td>9.6 AMMONIA</td>
<td>2.053</td>
</tr>
<tr>
<td>P11</td>
<td>31.000</td>
</tr>
</tbody>
</table>

From this table, it can be seen that changing the pipe size yields only a small increase in predicted lifetime, changing the chemistry increases the lifetime by between 60 and 105% and changing to P11 pipe material increases the lifetime by 3,000%. Further analyses would show that increasing the morpholine pH to 9.3 would double the predicted lifetime.

With the above information, the designer can make informed decisions regarding the most cost-effective way to modify plant operation to avoid future erosion-corrosion problems.

CONCLUSIONS

Piping erosion-corrosion is being addressed in the U.S. with a comprehensive program combining analysis with wall thickness measurements. Guidelines have been established and the program is well underway. With the EPRI developed CHEC and CHECMATE computer codes, the industry now has the tools to optimize that program for maximum safety at minimum cost. Benefits to a utility from utilizing CHEC and CHECMATE are extensive. In summary these codes:

- Are designed specifically to address erosion-corrosion inspection planning.
- Provide accurate predictions.
- Eliminate the potential for forced outages.
• Eliminate danger to personnel from pipe ruptures due to erosion-corrosion.

• Minimize resources needed for inspection by minimizing the number of locations to inspect and maximizing the interval between those inspections.

• Can be used as a design tool to extend piping life.

• Are well tested, documented and supported codes backed by EPRI.

• Are widely used and accepted by both users and regulatory bodies.

These benefits to a plant owner are estimated to exceed twenty million dollars over the life of the plant. CHEC and CHECMATE are available under a license agreement with EPRI.

REFERENCES


4. V. Chexal and R. Jones, "Implications of the Surry Piping Failure for Other Nuclear and Fossil Units." Presented at the SMIRT Post Conference Seminar No. 2., Davos, Switzerland, August 24-25, 1987.


This paper presents the work that has been performed to enable the SOPHT code to simulate large break Loss-Of-Coolant-Accidents (LOCAs) for Hydro-Québec's CANDU 600 MWe power plant located at Gentilly-2. Modifications implemented in the code itself include the LOCA control logic and the detailed Emergency Coolant Injection System (ECIS) modeling. A comparison between a SOPHT simulation and a Firebird simulation for a large outlet header break LOCA is presented.

DEVELOPMENT OF THE SOPHT CODE

The version of the code SOPHT used at Hydro-Québec for simulating the Gentilly-2 nuclear power plant is derived from the Ontario Hydro version (1) and rendered specific to the Gentilly-2 primary heat transport system (PHTS), secondary and auxiliary systems and associated control logic. The original code was targeted at the simulation of normal operating and upset conditions. This code was further modified for secondary side event analyses. For this project, the first task was then to provide the code with the capability to calculate the severe transients encountered during a LOCA. For the code the difficulties are twofold, first physical and second numerical.

The physical problems related to cold water injection into a fully voided node was partially solved by controlling the time step prior to cold water injection. This has avoided the pressure collapse that is observed in an homogeneous code due to the fact that the homogeneous properties of the fluid in the node (density and energy) are used to calculate the pressure.

The numerical problems raised by the water packing phenomenon which is the numerical overfilling of a node during the time step at which the void of that node is fully collapsed, was overtaken by adding a time step control feature to the code's basic approach of overpressure limitation through flow redistribution and equalisation.

The critical discharge model originally used by SOPHT is a correlation based on Henry and Fauske two phase flow critical discharge correlation (2) and did not include a lower pressure bound for the critical flow determination. At low pressure (below 200 kPa) the discharge flow was calculated for all qualities using the homogeneous density and the orifice equation instead of the Henry-Fauske model.

A simplified sonic check for steam, two phase and subcooled liquid had been implemented in the code. When the Nahavandi term is omitted, even with the very large driving forces due to the large pressure differences that can be induced in a large LOCA, the velocities of the flows in the primary heat transport piping were still within the sonic limit.

The quality-void correlation used in SOPHT was revised to give a better behaviour at low void and pressure. With low void and pressure the calculated quality was zeroed leading therefore to the use of subcooled derivatives and subsequent numerical problems.

Pressure drop and heat transfer correlations have been revised to accommodate the conditions of a large break LOCA, in particular high quality: in the original coding, some correlations were developed to cover only the normal operating quality range of the primary side.

In addition to the basic software modifications specialised subroutines have been added to the code. These subroutines simulate the behaviour of the system and its control logic during LOCAs. In particular the programming of the valve opening sequence, the crash cooldown logic, the rupture disks parameters, high pressure (HP) ECI tank transient pressure calculation, and the medium pressure (MP) and low pressure (LP) pump characteristics has been added to the code.

DEVELOPMENT OF THE GENTILLY-2 ECI MODEL

Before simulating a large break LOCA a complete ECI system characterisation had to be conducted. It consisted of gathering all relevant information about the ECIS such as detailed geometrical and topological data, initial thermalhydraulic conditions, system component characteristics, instrumentation location, actuation parameters and signal values. The characterisation is based on the actual plant data and on the systems as built with the most up to date information.

The information generated by the characterisation has been documented and stored in such a way that it
can then be easily used to create or revise the desired ECI nodalisation at any level of complexity. The nodalisation objective was to represent the ECI system as accurately as possible. Figure 1 shows the SOPHT system nodalisation for the Gentilly-2 ECIS.

FIGURE 1: SOPHT NODALISATION OF THE GENTILLY-2 ECIS

As can be inferred from figure 1, both loops are represented along with the three ECI stages, HP, MP and LP. All important features such as parallel valves, rupture disks and orifices are modelled.

SOPHT FIREBIRD COMPARISON

In order to assess the overall correctness of the code modifications a comparison was deemed necessary. It was chosen to compare the SOPHT results to FIREBIRD results in order to check both the code and the ECIS nodalisation. It also seemed important to Hydro-Québec to show that the two homogeneous codes simulating the same transient on the same plant would produce similar results. It was therefore decided to compare a SOPHT calculation to results that were already available from FIREBIRD.

Scenario Description

The scenario chosen was the guillotine break of an outlet header connected to the pressuriser. The station data to be modeled were those of Gentilly-2. Figure 2 displays a simplified diagram of the primary heat transport system with the localisation of the assumed broken outlet header and definition of broken and critical pass.

Before the break occurs, the reactor is supposed to be at 103% full power to account for the power measurement uncertainty. The reactor is postulated to be tripped through a neutronic signal on shutdown system number one. When the ECI actuation pressure is reached, the ECI isolation and injection valves open. The loop isolation valve start to close at the same time. The 30 seconds crashcool countdown is also started. The simulation is done for a total of 180 seconds.

FIGURE 2: SIMPLIFIED DIAGRAM OF THE PHTS

Model Set-up

In order to establish a clear basis of comparison, the same steady state conditions had to be achieved in the SOPHT simulation as compared to the FIREBIRD simulation. Therefore the same inlet temperature was targeted by altering the heat transfer coefficient via the cruding factor between the primary and the secondary side and by changing the heat output ratio between the steam generator and its preheater. In order to obtain the same exit quality, the gross core flow was equalised through the adjustment of the pump discharge lines resistances. This correction was implemented to take care of the different pump characteristics used in the models used to conserve the overall system resistance distribution. Since we intended to simulate a large break, the primary fluid properties were those of light water from the beginning of the transient.

Table 1 gives the values used by FIREBIRD which are mostly related to Point-Lepreau Generating Station at end of life and those adjusted for the SOPHT simulation.

TABLE 1: STEADY STATE INITIAL CONDITIONS

<table>
<thead>
<tr>
<th>Firebird</th>
<th>SOPHT adjusted</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial Power Level (W)</td>
<td>103</td>
</tr>
<tr>
<td>Fuel Heat to Coolant (MW)</td>
<td>2108</td>
</tr>
<tr>
<td>Pump Heat (MW)</td>
<td>17</td>
</tr>
<tr>
<td>Heat to SG (MW)</td>
<td>2119</td>
</tr>
<tr>
<td>Boilers (MW)</td>
<td>1836</td>
</tr>
<tr>
<td>Preheaters (MW)</td>
<td>282</td>
</tr>
<tr>
<td>ROH pressure (MPa)</td>
<td>10.0</td>
</tr>
<tr>
<td>RIH pressure (MPa)</td>
<td>11.2</td>
</tr>
<tr>
<td>Pump head (MPa)</td>
<td>1.8</td>
</tr>
<tr>
<td>Core flow (kg/s)</td>
<td>1814</td>
</tr>
<tr>
<td>RIH temperature</td>
<td>268</td>
</tr>
<tr>
<td>ROH quality (%)</td>
<td>4.1</td>
</tr>
</tbody>
</table>
Preliminary simulations have shown that the nodalisation has some effect on certain results of the transient. We have therefore modified the SOPHT nodalisation for Gentilly-2 in order to avoid these differences.

The ECI lines were approximated to connect directly into the headers. The end fitting were remodelled and placed into the channel outlet and inlet flow paths. The broken rupture disks resistance was considered. As shown in figure 3, the resistance at the ECI HP injection valves was adjusted to give the same total ECI flow. Individual injection flows to the headers were not adjusted. The medium pressure injection logic is inhibited since it was not predicted in the FIREBIRD simulation.

As far as the secondary side is concerned, since Gentilly-2 and Point-Lepreau do not have the same governor valve characteristics, we left the SOPHT simulation with the Gentilly-2 secondary side characteristics. This did not have a very important effect on this particular transient.

Results

The intent of this section is to present a comparison between a SOPHT and a FIREBIRD simulation and not to describe a particular accident scenario, therefore, we have concentrated on comparing the overall behaviour of the broken loop. Consequently, in this paper we do not present any result referring to the intact loop. For the broken loop, we have selected a few typical and characteristic parameters for which we show the SOPHT and FIREBIRD results. The most important physical characteristic of a blowdown is the pressure distribution around the primary heat transport system and in particular the pressure in the headers. The second most important parameter is the void distribution and transients which give a measure of the refilling process. We will also compare the flows at specific locations in the loop including the break discharge flows.

The broken outlet header pressures (named OH3 in figure 2) are shown below in figure 4.

Both codes show a very rapid depressurisation due to the size of the break and the pressure transients are quite similar. The opposite inlet header pressures (IH2 in figure 2) are shown in figure 5.

Again the depressurisation is very rapid and both transients are similar. Nevertheless between 40 and 80 seconds into the transient, the pressure calculated by FIREBIRD is noticeably higher than the SOPHT prediction.

The difference is attributed to different pump models used by the two codes. For certain conditions FIREBIRD uses the average properties of the downstream and upstream links while SOPHT always uses the...
pump suction properties. This has been identified as the major and only significant difference between the two codes.

Figure 6 shows the critical pass core flow for both calculations.

Again we can attribute the difference in flow magnitude to different pump models. SOPHT does not predict as severe a stagnation as does FIREBIRD since a small reverse core flow can be seen between 15 and 30 seconds. This only means that the critical break size predicted by SOPHT would be slightly smaller.

Figure 7 shows a very good agreement for the break discharge flows predicted by SOPHT and FIREBIRD.

The agreement is excellent since both calculations predict the same overall transient and the same time for fuel rewetting which is the most important aspect of ECI effectiveness. The similarity of the void collapse along the broken pass in the direction of the refill, shows that the magnitudes of injection flows and stagnation duration are similar.

Discussion

The comparison shows that the two codes, SOPHT and FIREBIRD, using reasonably similar hypothesis, will predict the same behaviour for large LOCA scenarios. The small noticeable differences are essentially attributed to pump model difference in the two code. Differences in the details of the nodalisation can also induce small variations in the results. A slightly different critical discharge model can also explain the differences observed in the break discharge flows.

CONCLUSION

The SOPHT version of Hydro-Québec is now fully capable of simulating large LOCAs. The changes made in the coding to accommodate the extreme thermohydraulic conditions and numerical problems associated with large LOCAs, the ECI logic that has been programmed and the characterisation of the ECI system have been successfully tested against another similar calculation. Furthermore, through the time step control logic that has been implemented to counter the numerical difficulties exposed in the paper, the overall time step is much larger and the transient calculation is quite fast.
REFERENCES


(4) M. R. Lin et al.: "FIREBIRD Program Description", TDAI-166 Volume 1, September 1979, AECL proprietary.

ACKNOWLEDGMENTS

This work has been entirely financed by Hydro-Québec. We would like to thank C.H. Nguyen for interesting discussion during this work.
AN IMPROVED DYNAMIC SIMULATION OF A CANDU PRESSURIZER

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ABSTRACT

In the course of developing a transient model for CANDU reactors for microcomputer systems, it has become necessary to develop models for components of the heat transport system. With several major alterations, the pressurizer model of H. Tezel has been found to successfully simulate the CANDU pressurizer.

Testing of the pressurizer model using Ptl. Lepreau data remains to be completed, however testing has been performed using arbitrary functions. Testing initially involved the application of functions of outlet header pressure, with good correlation to the expected results for a wide variety of situations. The model has also been tested when interfaced to DSNP and was again found to perform successfully.

INTRODUCTION

To simulate transients within a CANDU reactor, a project has been undertaken to adapt and use the Dynamic Simulator for Nuclear Power Plants, or DSNP. The DSNP codes essentially comprise a dedicated language for the design of simulations of nuclear power plants, including the heat transport system, the reactor dynamics, and the control logic. DSNP creates a simple environment for modelling of these systems in a wide variety of configurations on computers as small as micros.

The DSNP language was originally intended for simulating LMFBR reactors; therefore it has become necessary to modify, and in some cases completely develop, the models representing various components of a CANDU power plant, in particular those included in the heat transport system. Among the first components to be specifically redesigned for CANDU simulations with DSNP is the 600 MW CANDU pressurizer.

The pressurizer model consists of three main components:

1) The conservation equations for mass and energy.
2) The logic and implementation of the pressurizer control systems (heaters and steam bleed valve).
3) The routines for calculation of the thermodynamic properties of heavy water.

The model is written in Fortran 77 standard code such that it may eventually be run in any computer environment with minimal modifications. This model is a thorough revision of the Tezel pressurizer model, most notably in the following areas:

1) Calculation of the flow into or out of the pressurizer is now dependent on the difference in the pressurizer and outlet header pressures, as opposed to a dependence on the derivatives within the pressurizer. This modification allows for the simple integration of the pressurizer model to DSNP, which calculates the flow into or out of the pressurizer on this basis.
2) The allowance for all combinations of occurrence of condensation and flashing, based on the relation of fluid and liquid enthalpies within the pressurizer to the saturation pressure. In addition, the pertinent situation is determined within the pressurizer routine itself, and not externally.
3) An improved model of the pressurizer controller logic, based on the most recent data from Pt. Lepreau logic flow sheets.
4) Correction of previous miscalculations in the heavy water property routines.
5) Tolerances, originally intended to enhance speed of calculations, have been largely removed for the sake of accuracy in determining the states of the liquid and vapor in the pressurizer.
CONSERVATION EQUATIONS

Several significant assumptions are made in this modelling of the pressurizer:

1) The heavy water in the pressurizer is assumed to exist in two homogeneous phases only, and therefore temperature stratification is not modelled. There is no thermal interaction between the two nodes except through condensation and flashing.

2) The two nodes are not assumed to be in equilibrium thermally.

3) The pressurizer is assumed to be perfectly thermally insulated, with no heat transfer through the pressurizer walls.

4) A single pressure applies to both the liquid and vapor phases in the pressurizer.

5) The amount of surge and outsurge to the pressurizer is determined by comparison to the reactor outlet header pressure when tested in isolation, and is determined by the DSNP flow calculation routines when the model is tested within DSNP.

6) Condensation and flashing are assumed to be instantaneous, with no physical transfer mechanisms between the two phases.

7) No mechanism is included to model any convection effects due to the action of the heaters.

Condensation takes place when the temperature of the vapor falls below the pressure dependent saturation temperature, thus becoming saturated. A portion of the vapor will condense, in raising the temperature of the remaining vapor to the saturation line. Similarly if the temperature of the liquid node exceeds the saturation temperature, then it is considered saturated and flashing occurs, reducing the temperature of the liquid node to the saturation line. The physical transport of the boiling and flashing heavy water is not modelled at this time.

As the temperatures of the two nodes are independent, the liquid and vapor phases within the pressurizer may individually attain saturation, thereby permitting four possibilities for flashing and condensation:

<table>
<thead>
<tr>
<th></th>
<th>Saturated Vapor</th>
<th>Superheated Vapor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Saturated Liquid</td>
<td>Flashing &amp; Condensation</td>
<td>Flashing</td>
</tr>
<tr>
<td>Subcooled Liquid</td>
<td>Condensation</td>
<td>None</td>
</tr>
</tbody>
</table>

All of these possibilities are applicable to the pressurizer when water is surging out; however, in the case of water entering the pressurizer the liquid will be forced into the subcooled region and vapor will be forced into the superheated region. No condensation or flashing will take place. Therefore, there are five possibilities as regards the modelling of condensation and flashing.

When the constituents are at saturation, the thermodynamic properties are considered to be a function on the saturation line. Therefore, a single functional dependence (i.e. pressure, enthalpy, temperature) may be used to calculate all other thermodynamic properties via the Helmholtz free energy function. If, however, the constituents are not on the saturation line two properties are required to calculate the Helmholtz function for the superheated or subcooled state. In this model pressure and enthalpy are used for this purpose.

Two conservation laws must be obeyed by the model: mass and energy. In addition the mass and energy must be conserved for both the liquid and vapor nodes of the pressurizer. The conservation equations may be outlined by:

\[
\Delta \text{Mass Liquid} = (\text{Mass of Surge}) - (\text{Mass Flashed}) + (\text{Mass Condensed})
\]

\[
\Delta \text{Mass Vapor} = (\text{Mass Flashed}) - (\text{Mass Condensed}) - (\text{Mass through Bleed Valve})
\]

\[
\Delta \text{Energy Liquid} = (\text{Energy of Condensation}) - (\text{Energy of Surge}) + (\text{Energy from Heaters}) - (\text{Energy from Pressure Change})
\]

\[
\Delta \text{Energy Vapor} = (\text{Energy Flashing}) - (\text{Energy of Condensation}) - (\text{Energy through Bleed Valve}) - (\text{Energy from Pressure Change})
\]

The presence of the condensation and flashing terms is determined by the relationship between the enthalpies of the liquid and vapor, and the saturation enthalpy in the pressurizer, which is a function of the pressurizer pressure. The model determines whether the condensation and flashing terms are applicable and then determines the form of the conservation equations.
The flow from the heat transport system is considered positive on insurge and negative on outsurge. The total volume of the pressurizer is fixed, and thus, given the specific volume of the liquid and vapor from the energy equations and the mass of the liquid and vapor, the boundary condition of the pressurizer equations is outlined as:

\[ \Delta \text{[Mass Liquid} \times \text{Specific Volume Liquid}} + \text{Mass Vapor} \times \text{Specific Volume Vapor}] = 0 \]

The known quantities used to solve the equations include the external inputs (in/outsurge, heaters, and bleed valve) and the thermodynamic variables. This information is determined by the enthalpy of the two phases of the pressurizer, which, with the pressure, yields the Helmholtz free energy function. From the energy balance equations the actual amount of flashing, condensation, and/or the time derivative of the enthalpy of the vapor/liquid may be determined in terms of known quantities. Having determined these, the equations for the pressurizer are set up to solve for the time derivative of the pressure. The equations for the solution of the pressure derivative with respect to time are as follows:

**Outsurge, Saturated Liquid, Saturated Vapor**

\[
\frac{dp}{dt} = \frac{w_x \cdot v_x \cdot w_x \cdot v_x}{h_v} + \frac{M_v \cdot (\frac{dh_v}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_v)}{M_v \cdot (\frac{dh_v}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_v) - M_v \cdot \gamma} \cdot Q
\]

**Outsurge, Subcooled Liquid, Saturated Vapor**

\[
\frac{dp}{dt} = \frac{ \cdot v_x \cdot w_x + v_x \cdot w_x - \left( \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x \right) - \left( \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x \right) - M_v \cdot \gamma} \cdot Q
\]

**Outsurge, Saturated Liquid, Superheated Vapor**

\[
\frac{dp}{dt} = \frac{ \cdot v_x \cdot w_x - \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x - \left( \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x \right) - M_v \cdot \gamma}{\frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x - \left( \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x \right) - M_v \cdot \gamma} \cdot Q
\]

**Outsurge, Subcooled Liquid, Superheated Vapor**

\[
\frac{dp}{dt} = \frac{ -v_x \cdot w_x + v_x \cdot w_x - \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x - \left( \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x \right) - M_v \cdot \gamma}{\frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x - \left( \frac{dh_x}{dp} \cdot v_x + \frac{dv_x}{dp} \cdot h_x \right) - M_v \cdot \gamma} \cdot Q
\]

Having solved for the pressure derivative, the enthalpy and other thermodynamic properties for the two nodes of the pressurizer may be evaluated. By integrating for pressure over the time step, the new values for the pressurizer are established.

**CONTROL LOGIC**

The control logic used in the pressurizer simulator emulates that of the Pt. Lepreau pressurizer control scheme. Every two seconds the pressure of the reactor outlet header is read. This pressure is compared to a setpoint pressure to determine the pressure error and, if necessary, the error is filtered with the values of the pressure error over the previous 20 seconds. This procedure follows the actual control logic, which performs this operation to smooth out instrument jitter.

If the value of the pressure error exceeds the bleed valve setpoint (which is user definable), then a demand value for the bleed valve is calculated. The bleed valve proceeds continuously towards this demand value. The bleed valve cycles from fully closed to fully open state in ten seconds. The flow through the bleed valve is a choked flow and therefore the flow rate is dependent only on the pressurizer pressure and the magnitude of the valve opening.

If the value of the pressure error tends lower than the user definable setpoint, then the variable heater is activated in much the same manner as the bleed valve. A demand rate is determined every two seconds and this rate is applied to the variable heater. As no data are available for the lag time to full power of the variable heater, it is presently assumed to be instantaneous.

In addition to the variable heater, the four ON/OFF heaters are modelled. These heaters are activated to full power when individual pressure error setpoints are reached, and are deactivated at different individual setpoints. Each of the five pressurizer heaters has a maximum output of 200 kW.
THERMODYNAMIC CALCULATIONS

Calculation of the thermodynamic properties of the heavy water in the pressurizer is performed by modified versions of the support routines from the DSNP transient modelling system. These equations have been recently included in the DSNP support library, and calculate the Helmholtz free energy and its derivatives for any two inputs (pressure, temperature, enthalpy etc.).

Saturation temperature is determined from the pressure of the pressurizer, the same for both the liquid and vapor node. The Helmholtz function and its derivatives with respect to pressure and specific volume are determined from the pressure and enthalpy for each of the two nodes. From this information, the remainder of the thermodynamic properties may be determined for each of the two phases of heavy water in the pressurizer.

It is noteworthy that while the pressurizer is intended to be run with heavy water, it has also been tested with light water, and indeed may be used with any other reasonable fluid included in the DSNP libraries.

RESULTS

The model is written in Fortran 77 standard code on a 68020 Macintosh microcomputer equipped with a math coprocessor. The dynamic model runs approximately 10% faster than real time, in isolation, although modifications under development are expected to improve upon this.

The testing of the model involves three phases:

1) Testing of the model in isolation using arbitrary functions. This will ensure the applicability of the general algorithms and assumptions, and that the model reacts correctly to simple, well understood situations.

2) The application of the resulting model to DSNP systems. Involved in this process are the modifications necessary to couple the pressurizer integration scheme to the DSNP variable time step integration scheme, as well as application of the DSNP flow routines to the in/out surge and bleed valve flows.

3) Testing of the model using data from the Pt. Lepreau pressurizer and outlet headers. This process will allow for the fine tuning and revision of assumptions made in the model and verify that the model is accurate, in comparison to the actual pressurizer.

Isolated Model Testing

The initial testing of the pressurizer model was performed on the isolated model. The reactor outlet header pressure was simulated through the imposition of forcing functions. These functions included step functions, sine waves, random variation, and feedback functions. The surge magnitude was determined by a simple comparison of the pressurizer pressure and the reactor outlet header pressure.

This form of testing indicates good correlation to theoretical results when the pressurizer is acted upon by arbitrary functions of pressure and temperature at the exit of the pressurizer. All combinations of flashing and condensation cases were successfully modelled, and there is excellent correspondence between the activation of the controllers and the reaction of the pressurizer.

In the example shown (fig. 1), the reactor outlet header pressure is set to a decaying sine wave function with a period of one minute and a duration of two minutes, after which a feedback function which gradually moves the outlet header pressure to the pressurizer pressure is invoked. The model is then allowed to continue to steady state.

Figure 1. Reaction of pressurizer to arbitrary function of outlet header pressure (sine wave period = 1 minute, duration = 2 minutes)
Two points are notable from the graph. First, the heaters and bleed valves are not controlled by the pressure within the pressurizer, but by the reactor outlet pressure. Thus, if this pressure is lower than the heater setpoints, there will continue to be heat applied by the heaters to the pressurizer, regardless of the pressure in the pressurizer itself. The second notable feature from the graph is that there is not a large amount of correlation between the two pressures until the end of the 120 second function duration. This is a result of the pressurizer being tested in isolation from a dynamic system from which feedback can be obtained, therefore reactor outlet header is independent of the pressurizer pressure except for somewhat arbitrary feedback functions which are applied in lieu of an actual system.

Integration with DSNP Model

Work has already been performed on the application of the pressurizer to DSNP. After testing indicated that the pressurizer worked adequately in isolation, the pressurizer was installed in a DSNP model. The model consists of a closed piping loop including two heat exchangers (fig. 2).

The flow in the primary (central) loop is driven by a single pump and the pressurizer is connected to the loop by a junction. The initial pressure in the loop is variable by the user and the use of valves both on the primary flow and on the heat exchanger secondary flows allow for a wide variety of tests to be run.

There are two goals for this phase of testing:

1) The implementation of the pressurizer to DSNP systems, without compromising the results of the pressurizer model or the DSNP simulation.
2) The effect of the pressurizer on a closed loop system for the maintenance of pressure in the event of pressure transients.

The initial results of these tests show that the pressurizer is, after some modifications to the isolated case code, easily interfaced to DSNP systems. The in/out surge and bleed valve flows are determined by DSNP flow routines and the pressurizer integration scheme is transformed to work with the DSNP variable time step integration scheme.

The pressurizer performs successfully within the DSNP model. When the primary flow in the DSNP model is acted on by temperature changes and valve actions, the pressurizer controllers react quickly to stabilize the pressure. The routines of DSNP allow for the actual calculation of flows and the removal of the estimates for the in/out surge flows and the bleed valve flow made in the isolated pressurizer.
In the example given in figure 3, the pressurizer is acted upon by two changes in the closed loop, one at \( t=0 \) and the second at \( t=30 \). In both cases the pressurizer acts quickly to stabilize the pressure close to the setpoint of 10.15 MPa. Initially the bleed valves are utilized to counteract an insurge from the closed loop, as the initial pressure of the closed loop exceeds that of the pressurizer. At 30 seconds the heaters turn on to prevent continued pressure loss, as a valve on the primary loop is closed, thereby decreasing the pressure in the primary loop.

While interfacing of the pressurizer to the DSNP model has been completed successfully, the effect of the pressurizer on the closed loop simulation is still being evaluated. The performance of the pressurizer itself is unhindered by inclusion into larger models and the pressurizer reacts properly to the conditions imposed on it by the DSNP simulation. However, work remains to be done on the flow and inventory routines of the DSNP simulation which now indicate significant errors of mass balance in the primary loop. The pressurizer reacts properly to the larger system, but the DSNP model has no corresponding response resulting from flows in and out of the pressurizer. Work to correct this problem is currently underway. The pressurizer provides an excellent means for testing the progress of this work.

**Comparison with Pt. Lepreau Data**

Testing of the pressurizer with data from the Pt. Lepreau pressurizer is currently underway. This work will allow the verification of the quantitative model results. By comparison of the pressurizer reaction to changes in the outlet header conditions, and the actual Pt. Lepreau pressurizer results, the applicability of the assumptions made can be evaluated.

**CONCLUSIONS**

A model for the simulation of the 600 MW pressurizer has been improved and tested for use with the DSNP simulation language. The initial testing of this improved model indicate good correlation with expected results under a wide variety of imposed conditions. In addition the model has been successfully integrated into the DSNP simulation language. Further work will test the model against actual data from the Pt. Lepreau pressurizer, and will further enhance the interaction between the pressurizer output and the DSNP simulations.

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2. SAPHIER, D., "The DSNP Language - A Short Description", Georgia Institute of Technology, Atlanta, 1985.


**NOMENCLATURE**

**Main:**
- \( h \) enthalpy (J/kg)
- \( M \) mass (kg)
- \( p \) pressure (Pa)
- \( Q \) heater input (Watts)
- \( t \) time (s)
- \( v \) specific volume (m³/kg)
- \( w \) flow (kg/s)
- \( \gamma \) isentropic expansion coefficient

**Subscripts:**
- \( f \) saturated liquid
- \( s \) subcooled liquid
- \( g \) saturated vapor
- \( v \) superheated vapor
- \( o \) insurge/outsurge
- \( b \) bleed valve
THERMOHYDRAULICS OF MODERATOR EXPULSION FROM A CANDU CALANDRIA UNDER SEVERE ACCIDENT CONDITIONS

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ABSTRACT

The computer program MODBOIL has been developed and used to predict the thermalhydraulic behaviour of CANDU cores under severe accident conditions involving the loss of the moderator cooling system. The program uses a drift-flux model to characterize the transient behaviour of the moderator under such conditions. The sensitivity of the prediction of the moderator expulsion rate to the values of the drift-flux velocity-void-distribution and void-drift parameters is shown. The paper also discusses the physical implications of variations in these parameters and draws certain conclusions.

INTRODUCTION

The separately-cooled, low-temperature moderator in a CANDU reactor provides a back-up heat sink following a loss-of-coolant accident (LOCA) should the emergency cooling injection (ECI) fail or be impaired. Analyses and experiments have shown that there will be no gross melting of the fuel in such an accident because of heat transfer to the moderator (1,2,3,4).

A study completed in 1984 for the Atomic Energy Control Board investigated the behaviour of CANDU reactor cores (specifically, a Bruce reactor core) under accident conditions more severe than those considered in the normal licensing process (5,6,7). The results of this study are also summarized in reference 4.

The study focused on a hypothetical accident sequence following a large LOCA with a failure of ECI in which the moderator cooling system also fails thus impairing the back-up heat sink. As a result, the moderator heats up, begins boiling and is expelled from the calandria. As fuel channels are uncovered by moderator expulsion, they overheat and disintegrate, falling to the bottom of the calandria, where they are quenched by the remaining moderator. Eventually, all the moderator is lost from the calandria and the core debris heats up and melts. The study predicts that molten core material would be contained in the calandria which, because of the heat sink provided by the separately cooled water-shield in which the calandria is immersed, maintains its integrity throughout the accident sequence.

In that study, the digital computer program MODBOIL (5,8) was developed to predict the thermohydraulic behaviour of the moderator and fuel channels in such an accident sequence. MODBOIL has also been used by AECL in a probabilistic risk analysis for a 600 MW CANDU reactor (9).

MODBOIL uses a drift-flux model (10) to characterize the behaviour of the moderator under sub-cooled and bulk-boiling conditions in such severe accidents. The drift-flux model requires the specification of appropriate values of two parameters, the velocity-void distribution parameter, \( C_0 \), and the weighted-mean vapour drift velocity, \( V^\delta \). These parameters depend on the geometry of the system and especially on the nature of the existing two-phase flow pattern. In the earlier studies using MODBOIL (5,6,7,8) the choice of the values used for these parameters was based on the experimental results of Ginsberg, et al, (11), for volume-heated boiling pools. A volume-heated boiling pool was considered to be a reasonable approximation to the geometry of these accident conditions, in which heat sources (the fuel channels) are fairly uniformly distributed throughout a boiling pool. Ginsberg, et al (11), found that the flow pattern in such cases was generally churn-turbulent and that \( C_0 = 1.2 \) and \( V^\delta = u_\infty \), where \( u_\infty \) is the isolated-bubble rise velocity for churn-turbulent flow conditions. Other experimental data on volume-heated boiling pools examined during the course of the more recent application of MODBOIL (9), suggests that \( C_0 \) may lie in the range of 1.5 to 2.0 and that \( V^\delta = C_0 u_\infty \) where \( C_0 \) may be as large as 2.0 (12,13).

This paper describes the results of an investigation of the sensitivity of the transient boiling behaviour of the moderator in a severe accident sequence to the values of the driftflux parameters \( C_0 \) and \( C^\delta \). The paper also discusses the physical implications of variations in \( C_0 \) and \( C^\delta \) and draws certain conclusions.

HEATING AND EXPULSION OF MODERATOR FOR BRUCE REACTOR REFERENCE CONDITIONS

In MODBOIL, the moderator is divided into a number of horizontal slices or nodes, one for each row of fuel channels, which constitute the control volumes in the analysis. A time-dependent heat source for each node is pro-rated from the overall decay heat source accounting for the number of channels in the node and their average power ratings. Allowance is also made in the heat source for the stored heat effects and for the exothermic Zircaloy-steam reaction, using information for worst-case conditions from WNRE. Transient heat and mass balances are made on each node for non-boiling, sub-cooled boiling and bulk-boiling conditions.

MODBOIL calculates the following parameters (7,8):

a) moderator liquid swell
b) pressure over the moderator
c) time to failure of the calandria rupture disks
d) initiation and development of sub-cooled and bulk boiling of the moderator
e) vertical distribution of void fraction
f) vertical pressure distribution allowing for static head effects, two-phase flow pressure losses and choked flow (using the homogeneous equilibrium model) in the relief ducts

g) rate of, and total, moderator expulsion from the calandria as a function of time

h) time to disintegration of uncovered fuel channels

i) quenching of disintegrated fuel channels in the remaining moderator and resulting changes in expulsion rate.

Results obtained for a Bruce-A reactor unit show that rupture disk failure occurs at about 9.2 minutes and that liquid begins to spill from the relief ducts at 9.4 minutes. Bulk boiling begins in the top slice at about 16 minutes and propagates rapidly downward until it becomes fully-developed throughout the calandria at about 17.6 minutes (5,7). The rapid increase of moderator average void fraction in this period, as shown by the void profiles at various times in Figure 1, results in a large fraction of the moderator being rapidly expelled from the calandria, as can be seen in Figure 2. Moderator is expelled from the calandria in surges during this period, as can be seen in Figure 3.

After bulk boiling is fully developed throughout the calandria, moderator boil-off continues at a much reduced rate as a quasi-steady-state process until fuel channel uncoverly begins. The subsequent surges in moderator discharge rate and accompanying drops in moderator inventory, shown in Figures 2 and 3, result from groups of uncovered fuel channels collapsing into the remaining moderator. Finally, all the moderator has been expelled from the calandria in about 50 minutes, and the core debris has accumulated in a quenched state at the bottom of the calandria.

The results shown in Figures 1 to 3 were obtained using the reference conditions for the earlier studies for the AECB (5,6,7) which included values of the drift flux parameters \( C_0 \) and \( C_1 \) of 1.2 and 1.0 respectively. It is evident from Figures 2 and 3 that the most significant periods for moderator expulsion occur when bulk boiling is developing over the calandria height and when groups of uncovered fuel channels collapse into the remaining moderator. The high expulsion rates during these periods result from the transient level swell and vapour release characteristics of the boiling moderator. For convenience, the present study of moderator expulsion sensitivity to the drift flux parameters was confined to the period in which bulk boiling develops over the calandria height. Bruce-reactor conditions were used in this sensitivity study.

RESULTS OF SENSITIVITY STUDY

Results obtained using MODBOIL for the onset of bulk boiling (OBB) period for the reference values of \( C_0 = 1.2 \) and \( C_1 = 1.0 \) can be seen in Figure 4, which shows the remaining moderator inventory versus time and in Figure 5, which shows the moderator expulsion rate versus time.
Figure 4 shows that about 61 percent of the moderator mass has been expelled from the calandria in less than two minutes during the OBB period. Figure 5 shows the surges in expulsion rate which are predicted to occur during this period. The accompanying pressure surges during OBB are not severe (peak pressures < 260 kPa abs.) because of the low steam quality at the exits of the calandria relief ducts (< 4%) during this period.

The effect of varying $C_0$ from 1.2 to 2.0, for $C_1 = 1.0$, on the moderator mass remaining in the calandria is shown in Figure 6.

It is evident that the mass of moderator remaining in the calandria after the OBB period increases considerably with $C_0$. For $C_0 = 1.2$, about 39 percent of the moderator remains in the calandria at the end of this period compared to about 60 percent for $C_0 = 2.0$. Figure 7 compares the moderator expulsion rates during the OBB period for $C_0 = 1.2$ and $C_0 = 2.0$ for $C_1 = 1.0$. It can be seen that the peaks of the expulsion surges are reduced and the surging behaviour becomes milder as $C_0$ is increased. The pressure surges during this period were also reduced as $C_0$ was increased.
Figure 8 shows the effect of increasing $C_i$ from 1.0 to 2.0 for $C_o = 1.2$ on the moderator mass remaining in the calandria after the OBB period. For $C_i = 2.0$, about 48 percent of the moderator remains after this period compared to about 39 percent for $C_i = 1.0$. Figure 9 shows that the effect of increasing both $C_o$ and $C_i$ to 2.0 from the reference values of 1.2 and 1.0 is to increase the moderator mass remaining in the calandria after the OBB period from about 39 percent to about 63 percent. Figure 10 shows that the flow rate surges become even less severe for $C_o = 2.0$ and $C_i = 2.0$ than for $C_o = 2.0$ and $C_i = 1.2$. Again, the pressure surges during the OBB period also become less severe.

DISCUSSION OF RESULTS

Flow Oscillations in OBB Period

The surging behaviour during the OBB period, which results in flow rate oscillations between values as high as 4,100 kg/s and as low as about 700 kg/s for the reference values of $C_o = 1.2$ and $C_i = 2.0$, was explained in reference 8 as follows: the rapid rate of vapour generation as bulk boiling first develops expels two-phase fluid rapidly through the calandria relief ducts. The resulting pressure losses in the relief ducts and flow choking at the exits increases the pressure in the calandria, which reduces the vapour generation and void formation rates. The two-phase flow rate through the relief ducts, and the accompanying pressure loss are reduced, decreasing the pressure in the calandria. The decreased pressure results in increased vapour generation and void formation rate and the cycle repeats. This process continues until the entire remaining moderator is undergoing bulk boiling.

It may be questioned whether the MODBOIL predictions of flow oscillations during the OBB period represent a real physical effect, as described above, or simply represent a mathematical effect resulting from the transient finite-difference numerical method used in the analysis. That the oscillations represent real behaviour is supported by the observations of Ginsberg, et al (11) who reported that under experimental boiling conditions in a volume-heated pool, the pool "... was in a dynamic, chugging state". Evidence from the present analysis also suggests that the predicted oscillations are a real and not a mathematical effect. MODBOIL permits independent setting of the numerical time step during the OBB period, so that it may be set at a suitably low value during this period when conditions are changing rapidly, while the normal time step for the rest of the transient may be much larger so as to minimize computer time and costs. For a normal time step of 12 seconds, computer runs were made using time steps in the OBB periods of 12, 3, 1.5 and 0.75 seconds. Some differences in the predicted behaviour were observed between the OBB time step of 12 seconds and of 3 seconds, but there was very little difference between the results obtained using the other three time steps. Differences between the predictions of the moderator inventory in the calandria at any time during the OBB period were not significant for the three shorter time steps and in particular, the predicted moderator inventory at the end of the OBB period was essentially the same (within 3%) for these different time steps. If the observed flow rate oscillations in this period were an artifact of the numerical method, it might be expected that the predicted inventory history would be significantly affected by the variation of the time steps.

Finally, excellent overall mass and energy conservation were obtained at all times during the computer runs for the three shorter time steps.

Significance of $C_o$ and $C_i$

The drift-flux parameter $C_o$, the velocity-void distribution parameter, depends, at least for circular-tube flow geometries, on the velocity and void profiles across the tube. The larger the value of $C_o$, the more peaked the velocity and void profiles are.

In the sensitivity study, a time step of 1.5 seconds was used during the OBB period.
With highly peaked void and velocity profiles, the liquid "lifted" from a volume by the level swell resulting from rapid vapour generation might be expected to be less than that for more uniform void and velocity profiles since, in the former case, most of the level swell would be concentrated near the centre of the cross-section of the volume so that liquid near the outer perimeter would not experience significant swell.

Confirmation that an increase in \( C_0 \) results in less level swell can be obtained from the simple single-node level-swell model of Grolmes (13). Using physical property and power input data representative of the Bruce reactor conditions used in the sensitivity study, and representing the calandria geometry by an open vessel of uniform cross-section equal to the average free cross-section of the Bruce calandria and a discharge area equal to that of the four relief ducts, calculations were made of the rate of level swell and the asymptotic swollen height using the Grolmes model with \( C_0 \) varied from 1.2 to 2.0 for \( C_1 = 1.0 \). The calculated asymptotic swollen height as a function of \( C_0 \) is shown in Figure 11, from which it is clear that the asymptotic swollen height decreases significantly as \( C_0 \) increases from 1.2 to 2.0.

Thus, the effect of an increase in \( C_0 \) from 1.2 to 2.0 on the thermohydraulic behaviour of the moderator, i.e., considerably less moderator expelled during the OBB period and gentler expulsion surges is explained by the reduction in average level swell of the moderator as \( C_0 \) is increased, because of the resulting greater peaking of the void and velocity profiles.

The parameter \( C_1 \) relates the weighted-mean vapour drift velocity, \( V_{dm} \), to the isolated-bubble rise-velocity in an "infinite" medium, \( u_r \). For the churn-turbulent flow regime, \( C_1 \) has a value of 1.0 for normal geometries (10). A value of \( C_1 \) greater than 1.0 means that vapour would be released more readily from a boiling pool than would be the case for \( C_1 = 1.0 \). Thus the data of Ginsberg, et al (12), and Grolmes (13) for volume-heated boiling pools, which indicate that \( C_1 \) may be as large as 2.0, imply that a relatively high rate of vapour release (or a low rate of liquid entrainment) would be expected in such cases. A higher rate of vapour release implies a lower asymptotic swollen height of a pool during the OBB period. Again, confirmation of this behaviour is confirmed from the application of the Grolmes model. Figure 11 also shows that the predicted asymptotic swollen level is reduced as \( C_1 \) is increased from 1.0 to 2.0 for all values of \( C_0 \). Thus, the effect of an increase in \( C_1 \) from 1.0 to 2.0 on the thermohydraulic behaviour of the moderator, which is similar to the effect of an increase of \( C_0 \), is explained by the increase in vapour release rate (decrease of liquid entrainment) and resulting reduction in level swell rate and asymptotic swollen height.

CONCLUSION

It has been shown that the predicted rapid rate of expulsion of moderator during the OBB period in the postulated accident sequence is reduced as either or both the drift-flux parameters, \( C_0 \) and \( C_1 \), are increased. The reduction in expulsion rate results in milder surges in flow rate and pressure in the OBB period and considerably-greater moderator inventory in the calandria following the OBB period as \( C_0 \) increases from 1.2 to 2.0 and \( C_1 \) increases from 1.0 to 2.0. Thus, higher values of either or both \( C_0 \) and \( C_1 \) would result in less severe conditions in the accident sequence since greater moderator inventory in the calandria following the OBB period would result in longer times to subsequent channel uncover and collapse and hence for operator intervention.

Also, the moderator expulsion surges which occur after groups of uncovered fuel channels collapse into the remaining moderator, would be less severe than those shown in Figure 3 for values of \( C_0 \) greater than 1.2 and \( C_1 \) greater than 1.0, since the same physical mechanisms apply as during the OBB period.

While the work of Ginsberg, et al (12) and Grolmes (13) on volume-heated boiling pools suggests that values of \( C_0 \) as large as 1.5 to 2.0 and \( C_1 \) as 2.0, rather than the values of \( C_0 = 1.2 \) and \( C_1 = 1.0 \) used in most MODBOIL studies, may be more appropriate for the moderator expulsion analysis (resulting in considerably lower expulsion rates during OBB than those predicted in those studies), it must be recognized that the actual values of \( C_0 \) and \( C_1 \) which would be most appropriate for the CANDU calandria geometry are unknown. It is expected that the presence of the calandria tubes, as well as other components, would affect the void and velocity profiles, and the volume swell and liquid entrainment behaviour (and hence \( C_0 \) and \( C_1 \)) during moderator expulsion. The appropriate values of \( C_0 \) and \( C_1 \) for a CANDU calandria geometry must be established by experimental means.

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COMPARISON OF HOMOGENEOUS AND TWO-FLUID SIMULATIONS
OF A LARGE-BREAK LOCA IN A CANDU

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ABSTRACT

Most thermal hydraulic analyses of postulated loss-of-coolant accidents (LOCAs) for the CANDU reactor, performed to date, have utilized computer codes employing a homogeneous assumption for fluid flow. Recently, advanced two-fluid codes have become available for such analyses. This paper describes a comparison of a homogeneous simulation (FIREBIRD) with a two-fluid simulation (CATHENA) for a postulated large-break LOCA in a CANDU reactor. The comparison shows that, even for a very fast transient, differences exist between the homogeneous and the two-fluid calculations. Most differences are due to the different method of modelling non-equilibrium effects in the two codes, especially fluid temperatures. However, overall system behaviour is similar, especially refilling of the channels.

INTRODUCTION

Thermal hydraulic codes employing a homogeneous assumption for fluid flow have been used for most postulated loss-of-coolant accident (LOCA) thermal-hydraulic analyses performed to date. These codes are expected to accurately predict events during the LOCA when the phase velocities are almost equal and both phases are well mixed thermally. FIREBIRD-III MOD1 (1), developed by Atomic Energy of Canada, CANDU Operations, has been used for such analysis. Although a homogeneous code, FIREBIRD has special features that address non-equilibrium fluid flow, including "slip" and "drift" for modelling unequal velocity effects, and property smoothing for unequal temperature effects.

CATHENA (2), an advanced two-fluid thermal hydraulic code, has been developed at the Whiteshell Nuclear Research Establishment of Atomic Energy of Canada Limited. A full two-fluid representation of flow is modelled, in which the liquid and vapour phases may have different pressures, velocities, and temperatures. Interphase mass, energy, and momentum transfer (e.g., condensation, boiling, interphase shear), which determine the interaction between the phases, are specified using constitutive relations derived from the literature. These have been validated using separate-effects tests, component tests, and integral tests (3).

This paper presents a comparison of a FIREBIRD simulation and a CATHENA two-fluid simulation of a postulated LOCA in the CANDU reactor. The scenario presented is a 100% break at a reactor outlet header (ROH), with pumps running, and Emergency Core Cooling (ECC) system available. This has been identified as one of the limiting critical break scenarios in terms of ECC effectiveness for the CANDU 600 reactor. A FIREBIRD model already existed to represent this event, and a CATHENA model was generated to resemble, as closely as possible, the FIREBIRD model.

THE FIREBIRD-III MOD1 CODE

FIREBIRD-III MOD1 is a general network code developed primarily for predicting the thermal hydraulic behaviour of CANDU reactor power plants during postulated LOCAs. Because of its generality, the code can also be used to solve a large variety of general flow network problems for both light and heavy water. In the code, a set of "user" routines is provided that allows the user to specify the boundary conditions, and control logic (e.g., boiler level and pressure control) for a particular problem. The code then couples these boundary conditions and control logic with its fluid flow conservation equations, fluid state equations, and heat conduction equation (for piping walls and fuel), to form the governing equations for the system being analyzed.

The FIREBIRD code also includes component models, such as pumps, valves, and pressurizers, required to model reactor components.

The homogeneous equations for fluid flow are represented, in conservation form, on a staggered mesh and solved by a finite-difference technique, resulting in a 3-equation model. Although FIREBIRD is, strictly speaking, a homogeneous code, two features have been added to model non-equilibrium effects. The first feature is the use of drift correlations in vertical pipes, and correlations in horizontal pipes. The choice of correlations is specified by the user, depending on the system being modelled. The second feature is an adjustment of the equation of state, termed "smoothing". This technique is used when a control volume, or node, is near the transition from saturated to subcooled liquid. In this region, very steep gradients exist in the properties of H2O and D2O. This adjustment of properties provides a pseudo-modelling of the non-equilibrium pressure during cold water injection and allows the FIREBIRD numerics to maintain their efficiency during cold water injection.
THE CATHENA CODE

CATHENA is a one-dimensional thermalhydraulic computer code also developed primarily to analyze postulated LOCA scenarios for CANDU nuclear reactors. The code uses a non-equilibrium, two-fluid thermalhydraulic model to describe fluid flow. Conservation equations for mass, momentum and energy are solved for each phase (liquid and vapour), resulting in a 6-equation model. Also, a noncondensible gas may be represented as part of the vapour phase, yielding a 7-equation model. Relationships specifying interphase mass, momentum, and energy transfer are flow-regime-dependent and have been derived from correlations obtained from the literature.

The numerical solution technique used to solve the conservation equations is a staggered-mesh, semi-implicit, finite-difference method. The dependent variables are pressure, void fraction, phase enthalpies, and phase velocities. If a noncondensible gas is present, the noncondensible fraction is also a dependent variable. Conservation of mass is achieved using a truncation error correction technique, similar to that used in RELAP5/MOD2 (4).

The numerical solution scheme is not transit-time-limited (5) and the time step is selected based on accuracy considerations alone. A time-step controller is implemented in CATHENA that automatically selects the next time step at each finite-difference time step.

Heat transfer from metal surfaces is handled by an extensive wall heat transfer package. A set of flow-regime-dependent constitutive relations specify energy transfer between the fluid and the pipe wall and/or fuel-element surfaces. Heat transfer by conduction within the piping and fuel is modelled in the radial direction and can be modelled in the circumferential direction as well. A variational finite-element method is used. Radiative heat transfer and the zirconium-steam reaction can also be calculated. Built into this package is the ability to calculate heat transfer from individual groups of pins in a fuel bundle subjected to stratified flow. Under these conditions the top pins in a bundle are exposed to steam, while the bottom pins are exposed to liquid.

Component models that describe the behavior of pumps, valves, pressurizers, steam separators, discharge through breaks, etc., have been implemented in the code. Control systems, required to complete the idealization of reactor systems, are defined by user-supplied input data. Systems of considerable complexity have been modelled (6).

100% ROH BREAK

The postulated LOCA scenario for this comparison is a 100% break at a reactor outlet header, with pumps running and ECC available. This case was chosen because it represents a rather fast transient for a first comparison between FIREBIRD and CATHENA. For slower transients, gravity-induced phase separation will occur, and more differences would be expected between the two code predictions.

Light water (H₂O) was modelled in the primary heat transport system for this transient. Although both FIREBIRD and CATHENA allow either H₂O or D₂O to be used for a simulation, they do not allow both to be used in the same node. In a rapid transient such as this, the blowdown phase of the D₂O is short, and H₂O ECC coolant refills the circuit early. Thus H₂O is in the circuit for most of the transient.

CANDU 600 MODELS

Figure 1 shows a simplified schematic of a CANDU 600 station that must be modelled.

An existing FIREBIRD model was used to perform the FIREBIRD simulation for this reactor. The two figure-of-eight loops were modelled, connected by piping to the pressurizer and to the feed and purification circuits. The header-to-header interconnects were modelled. The ECC injection system (not shown in Figure 1) was also included in the model. Reactor control systems for boiler level, boiler pressure, pressurizer level, loop isolation, ECC initiation, boiler crash-cooldown, and turbine unloading were all modelled. A break, 0.26 m² in area, corresponding to twice the cross-sectional area of an outlet header,
was modelled at outlet header number 3, which is connected to the pressurizer. The reactor power transient following the break was imposed as a boundary condition.

The CATHENA model was created with the objective of representing the FIREBIRD model as closely as possible. Figure 2 shows the CATHENA idealization - the FIREBIRD idealization is very similar and is thus not shown.

DIFFERENCES IN MODELLING

In a comparison such as this, the agreement or disagreement between two code predictions will, in general, depend on the following three factors:

1) the physics employed in the code model, and the numerical solution of the model (i.e., two-fluid versus homogeneous model),
2) the representation of the system being modelled (e.g., geometric data, nodalization), and
3) the modelling of components such as pumps, and boundary conditions (e.g., reactor power transient).

For this project a major effort was directed towards reducing and eliminating differences due to factors (2) and (3). Thus differences presented in the following section will largely be due to factor (1) - the modelling of fluid flow using a homogeneous versus a two-fluid representation.

RESULTS

In the comparison between predictions produced by the two codes for this scenario, about 100 solution variables were compared. In this paper, only a partial comparison of the broken loop response will be presented. Both codes predicted that the intact loop is isolated with sufficient inventory to keep the fuel well cooled during the entire transient.

The broken pass of the broken loop is defined as the pass nearest the break, between inlet header 2 and outlet header 3. The critical pass, located between inlet header 4 and outlet header 1, was found to experience a flow stagnation during the early portion of the transient, with a resulting rise in fuel temperatures.

It must be noted that this work examines only the system response to the LOCA. A detailed fuel channel was not represented in either the FIREBIRD model or the CATHENA model. Thus calculated sheath temperatures might not be representative and are therefore not compared.

Event Sequence

Both simulations were performed for 200 s covering the periods of blowdown and refill of the broken loop fuel channels with high-pressure emergency core coolant. Because the same power transient was used in FIREBIRD and CATHENA, trip times are implicitly the same. The loop isolation and injection signals (5.5 MPa(a) in two of three instrumented headers) occur at 6.3 s in FIREBIRD and 5.0 s in CATHENA. The more rapid initial depressurization rate predicted by CATHENA can be attributed to slightly lower initial stored energy in the primary heat transport system for the CATHENA steady state as well as the slightly higher CATHENA discharge. Injection to the broken loop begins at 15 s in FIREBIRD and 13 s in CATHENA. This is consistent with the difference in injection signal timing. Boiler crash cooldown is started at 30 s after the ECI signal.

Break Discharge

The predicted break discharge flows and enthalpies are compared in Figures 3 and 4, respectively. Flows are very similar for the first 30 s, with the CATHENA result showing slightly higher flows.
latter portion of the transient, flows are also very similar. However, at about 60 s, a "spike" is seen in the CATHENA calculation. This is due to a short period when the header is predicted to be full of liquid.

The enthalpy of the discharge agrees relatively well during the entire transient.

Header Pressures

Figure 5 shows the pressures predicted by FIREBIRD and CATHENA at outlet header 3, the break location. Code predictions match closely, except in the range 20 to 40 s, where CATHENA predicts a slightly lower pressure. In Figure 6, the pressure at inlet header 4 is compared. Again, CATHENA predicts a slightly lower pressure between about 20 and 45 s.

In the period of about 20 to 40 s, subcooled ECC injection starts to fill the steam-filled system, and differences are expected in the system response because of the differences between homogeneous and two-fluid modelling of this effect.

Pump Head

Both FIREBIRD and CATHENA predictions show pump 2 downstream of the break acting like a check valve preventing reverse flow to the break. Figure 7 shows very similar prediction of head developed by this pump by the two codes, although timing of events are slightly different.

With pump 1, upstream of the broken pass (shown in Figure 8), flow is pulled through the pump in the forward direction. The initial 10 s of the transient are predicted similarly with the two codes. However, after this initial period, FIREBIRD shows a more negative head. In the long term, both CATHENA and firebird predict a very low head.

Header Void

Figure 9 shows the void in outlet header 3. Differences are noted, especially the CATHENA "spikes" at about 60 and 100 s when liquid is predicted to fill the header for short periods. Figure 10 shows the void in inlet header 2. Although both codes predict a rapid voiding in this header at about 6 s and refill again shortly thereafter, CATHENA shows a high void in this header from about 40 to 95 s.

The differences noted in inlet header 2 can be directly attributed to the modelling of unequal temperatures in the two codes. During the period 40 to 95 s, both codes predict steam inflow to the header from pump 1 as well as subcooled ECC inflow. FIREBIRD calculates the resulting mixture as single-phase
liquid. In the CATHENA calculation, the condensation rate is predicted to be not high enough to condense all the liquid. However, sufficient condensation does occur in the feeders to the broken pass to ensure that this pass is liquid-filled during most of this period.

ECC Flows

Figures 11 and 12 show ECC flows to inlet header 2 and outlet header 3, respectively. Flows are predicted to be quite similar, although in inlet header 2, a slightly lower flow is predicted by CATHENA between 40 and 140 s. ECC flows are strongly influenced by the pressure distribution in the primary heat transport system. In turn, this pressure distribution will depend on the treatment on non-equilibrium temperature effects due to the subcooled ECI flow.

Channel Void

Figure 13 shows the refilling of the critical channel, at the mid-length location. Both codes predict a very similar refill time, about 90 s, with CATHENA showing two recurrences of void at about 100 s and 115 s. This indicates that CATHENA predicts the refilling front as not being unidirectional but moving back and forth in the channel during refill.

As stated previously, the prediction of the refilling time of the critical channel is a major requirement of this analysis. This will determine when the fuel will be cooled by ECC flow.
SUMMARY

1. A comparison of a homogenous FIREBIRD simulation and the two-fluid CATHENA simulation of a 100% ROH break in a CANDU 600 has been performed. Differences are noted and attributed to the different treatment of non-equilibrium flow, especially unequal temperature effects.

2. These differences did not significantly affect the refilling time of the critical pass, one of the major questions asked of such an analysis.

3. Because the CATHENA idealization was generated to be as similar as possible to the FIREBIRD idealization, differences are expected to be minimized. If a two-fluid model were generated based on the requirement of accurately representing non-equilibrium effects, more differences might be evident. An example would be the treatment of channel voiding, and its effect on the reactivity transient.

ACKNOWLEDGEMENTS

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