KINS/HR-067

GENERIC LICENSING ISSUES APPLICABLE TO WOLSONG 3 & 4 LICENSING REVIEW

**A report to the
Korea Institute of Nuclear Safety**

by

Frederick C. Boyd

**Wild & Boyd Management Advisors Ltd.
Nepean, Ontario, Canada**

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15 January 1994

re: Contract for Technical Consultation on the Generic Licensing Issues Applicable to Wolsong 3 & 4 Licensing Review 1993.10

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Applicable to Wolsong
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Executive Summary

The Wolsong 3 & 4 nuclear power plants are of CANDU type which were designed according to the rules and regulations of the Atomic Energy Control Board (AECB) of Canada.

In 1992 AECB staff issued a first formal report (to the AECB Board) on "generic safety issues affecting power reactors". This was followed by a second report in 1993. These reports dealt with safety issues associated with Canadian CANDU nuclear power plants that applied to all or several plants and were considered insufficiently resolved. In most cases the concern was lack of certainty in the related safety analyses.

The AECB staff report of 1993 identified eight "generic action items" and six "long-term research issues", three of which AECB staff have indicated may be moved into the "action" category.

This report, prepared for the Korea Institute of Nuclear Safety (KINS), reviews the background of the AECB "generic action items" and the three "long-term research issues" noted above. It also reviews an additional topic - steam line failure outside of containment - which was included in the request from KINS. In all, twelve issues are covered.

These background reviews are followed by a discussion of the relevance of each issue to the licensing review (for Construction Permit) of Wolsong 3 & 4 and recommended actions to be taken by KINS.

RECOMMENDATIONS

1. HYDROGEN BEHAVIOUR IN CONTAINMENT

1.1. The Construction Permit should include a condition that acceptable analyses for the number and location of the hydrogen igniters must be submitted (suggest within one year).

1.2. Before granting the Construction Permit the design should be checked to confirm that Class III power is available for up to 70 igniters, in all areas of the containment.

2. MERCURY-WETTED RELAYS

2.1. Before granting the Construction Permit KINS should ensure that the specifications for any mercury-wetted relays used in safety-related systems call for them to be tin-doped.

3. THERMOSYPHONING

3.1. The Construction Permit should contain conditions requiring the submission of acceptable evidence of validation of the **CATHENA** code and acceptable analyses for fuel cooling in the event of a complete loss of Class IV electrical power (suggest within one year). The Construction Permit condition should require design changes if the analyses are not acceptable.

4. POST ACCIDENT FILTER EFFECTIVENESS

4.1. Before the Construction Permit is issued KINS should ensure that there is no need to vent the containment after any accident. If venting is proposed as an operational option the design of the radioactivity monitoring systems for the containment air and the air discharge system should be reviewed carefully for expected effectiveness, reliability and ease of use.

5. SHUTDOWN EFFECTIVENESS WITH TILT

- no recommendation

6. PRESSURE TUBE INTEGRITY

6.1. The Construction Permit should contain a condition requiring the submission of the following at least one year before the Operating Licence:

- details of final inspections prior to operation;
- details of proposed in-service inspection methods;
- proposed criteria for "fitness for service".

6.2. Before the Construction Permit is issued KINS should check:

- the design and specifications for the garter springs to ensure they will not move;
- specifications for pressure tube material, with reference to experience on hydriding, corrosion, creep, etc.

7. SECONDARY SIDE FAILURES OUTSIDE CONTAINMENT

- no recommendations

8. MANAGEMENT OF AGEING

8.1. The Construction Permit should require the submission, prior to the Operating Licence, of reliability and failure mode and effects analyses of all special safety systems, including sub-systems and necessary support systems.

8.2. The Construction Permit should require detailed documentation of the initial condition of all safety-related equipment prior to the Operating Licence.

9. HUMAN RELIABILITY ANALYSES

- no recommendation

10. CONSEQUENCES OF PRESSURE TUBE FAILURE

- no recommendation

11. ECCS EFFECTIVENESS

11.1. The Construction Permit should require (suggest prior to the installation of any ECCS equipment) submission of:

- satisfactory validation of **CATHENA** code for all relevant situations;
- complete analyses of ECCS effectiveness for all LOCAs.

12. MOLTEN FUEL - MODERATOR INTERACTION

- no recommendation

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1.INTRODUCTION

The term "generic licensing issues", in the context of this report, refers to safety issues identified by the Atomic Energy Control Board of Canada (AECB) as being applicable to all or most of the CANDU nuclear power plants in Canada.

In September 1992 the staff of the AECB, the nuclear regulatory agency of Canada, issued a "Board Member Document" (BMD 92-169), a report to the five-member governing board, entitled "Report on Generic Safety Issues Affecting Power Reactors". This was the first time that such "generic" issues were identified by the AECB in a formal report. The Korea Institute of Nuclear Safety (KINS) was aware of this document and the author included this topic in a set of lectures given during his visit to KINS in April 1993.

Since it had been agreed that units 3 & 4 of the Wolsong nuclear power station (which are of the CANDU design) would be licensed according to Canadian policies and practices, KINS contracted with the above company for the author (who has had many years of senior level experience in the licensing of nuclear facilities in Canada) to conduct a review of AECB identified "generic safety issues" and provide recommendations on their impact on the Wolsong 3 & 4 licensing review.

Specifically, the **Contract** calls for:

A - Review of Generic Licensing Issues for CANDU System

- Hydrogen behaviour in containment
- Mercury-wetted relays
- Cooling without pumps (thermosyphoning)
- Post-accident filter effectiveness
- Shut-down effectiveness with flux tilt
- Pressure tube integrity
- Protection against secondary side failures outside containment

B - Review of Background of Safety Concerns for CANDU

- Review of origins of safety concerns in CANDU systems
- Review of the current status in Canadian plants

C - Recommendations of Applicable Regulatory Requirements for Wolsong 3 & 4 licensing review

- Evaluation of the extent to which these safety issues could apply to Wolsong 3 & 4
- Recommendations for KINS position regarding the issues and Wolsong 3 & 4 Construction Permit.

Subsequent to the BMD report referenced above the AECB staff presented a further report (BMD 93-107) in which they identified another "generic action item" (as they are now termed): "Treatment of Human Factors in Ontario Hydro Reliability Analyses". Both of these BMD reports also included topics classified as "long-term research issues". These are topics which AECB staff consider as being insufficiently understood and. therefore, require further research. AECB staff have informed the author that some of the "Long-Term Research Issues" will probably be re-defined as "generic" issues, specifically:

- consequences of pressure tube failure
- analyses of the effectiveness of emergency core cooling systems
- molten fuel - moderator interaction

Part A of the **Contract** included one item that is not included in the AECB staff list of "generic action items", namely, "Protection against secondary side failure outside containment". Conversely, the AECB list includes a topic not in the Part A list; "Management of Ageing".

As indicated in correspondence related to this **Contract**, these additional issues are included in this report, even though they are not called up specifically in the Contract.

2. BACKGROUND

Readers familiar with the fundamentals of the Canadian approach to nuclear power plant safety and licensing could bypass this section.

Early in the Canadian nuclear power program (1960s) the AECB adopted a "risk" approach for the licensing of nuclear power plants and developed some basic design and performance criteria to achieve the desired low probability of a major release of radioactive fission products from a nuclear power plant. The overall objective was to ensure that the likelihood of a release of a large percentage of fission products would be less than 10^{-6} per reactor year.

To achieve this, in the absence then of sound probabilistic techniques, the reactor plant was conceived as having "process" systems (the basic functional systems of the plant) and several "special safety systems", each of which had a specific safety function, such as shutting down the reactor, providing emergency cooling of the uranium fuel or preventing the dispersion of radioactive material. Each of the "special safety systems" were to be separate and independent of each other and of the process systems and to be sufficiently effective that it would take a major failure of a process system combined with complete failures of two of the "special safety systems" for a large release to occur.

A maximum frequency of "serious" failures of the process systems of 1 per 3 years and a maximum unavailability of each special safety system of 1 / 1,000 were stipulated. (A "serious failure" of a process system was defined as one that could lead to the release of radioactive material in the absence of safety system operation).

As performance requirements for the designers, maximum values of radiation dose that might be received by a member of the public (using conservative assumptions and calculation methods) were set for "single" failures of a process system (such as a large pipe failure) and for "dual" accidents where such a process system failure was associated with complete failure of one of the "special safety systems". The reference dose value for an individual for a "dual" accident was chosen to be that proposed by the Medical Research Council of the U.K. as tolerable to be received once in a lifetime (25 rem or 250 mSv), even though the postulated likelihood of a "dual" accident was 1 in 3,000 reactor years.

Assuming adequate independence and effectiveness of the "special safety systems", the likelihood of a dual accident should, therefore, be less than 1 in 3,000 per year. And, the consequences of such a "dual accident must not lead to a dose, to any member of the public, greater than 25 rem or 250 mSv.

Following this approach and assuming the requirements for separation, effectiveness and unavailability were met, a major release could only occur if there were a serious process failure combined with complete failure of **two** independent special safety systems. This should have a likelihood of less than 1 in 3,000,000 per year ($1 / 3,000 \times 10^{-3} \times 10^{-3}$),i.e., the likelihood of a major release should be less than 3×10^{-7} per year, thus meeting the objective.

This approach was last formally presented in the following paper:

- Reactor Safety and Licensing Requirements
  D. G. Hurst and F. C. Boyd AECB - 1045 1972

These "single-failure / dual-failure" criteria are still used by the AECB in judging designs for nuclear power plants although additional requirements are now applied.

Although this single/dual failure approach provides a sound fundamental means of achieving the risk-based objective some concerns and reservations arose. Among these were:

- the difficulty of achieving appropriate separation of safety support systems or dealing with their failures;
- the fact that some special safety systems must continue to operate for some time after an accident;
- the inability to take into account (provide allowance for) the great variation in frequency of various failure scenarios;
- the problem of common-cause events such as earthquakes.

Techniques such as Safety Design Matrices (SDMs) and design approaches such as the "two-group" concept were developed by CANDU designers as means to address some of these perceived shortcomings. In turn the AECB staff put more emphasis on evaluation of the detailed designs, putting more emphasis on the concept of "defence in depth", which implies many layers of safety. This evaluation has been conducted almost entirely subjectively, based on the collective experience and judgement of the AECB reviewers.

Largely as a consequence of this approach the Atomic Energy Control Board has issued very few regulatory documents pertaining to nuclear power plants. In general this has worked surprisingly well, largely due to the considerable amount of "dialogue" that has taken place with the applicants or licensees. In recent years, however, Canadian licensees have complained that the AECB rulings were becoming more dogmatic and unrealistic.

Only a few formal "Regulatory Documents", directly related to nuclear power reactors, have been issued by the AECB. Four of these clarify and spell-out the requirements for shutdown systems, emergency core cooling, and containment, that were implied by the "single / dual failure" approach. These four are:

- R-7 Requirements for Containment Systems for CANDU Nuclear Power Plants
- R-8 Requirements for Shutdown Systems for CANDU Nuclear Power Plants
- R-9 Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants
- R-10 Use of Two Shutdown Systems in Reactors.

The first three were issued as "regulatory" documents in 1991, over 10 years after having been first issued as "consultative" documents. The last was issued in 1977 and was the first technical "regulatory" document issued by the AECB.

Another "Consultative Document", C-6, "Requirements for the Safety Analyses of CANDU Nuclear Power Plants", was issued in 1980. It provides a somewhat different approach (as compared to the single / dual failure one) for the evaluation of accident sequences. Six categories are defined, with a number of assigned accident sequences, and limits for the calculated dose to a member of the public for each category. It did not, however, give assumed or acceptable frequencies (probabilities) for these categories, making it difficult to assign an accident sequence that was not listed. A modified version, with agreed probabilities, was used on a trial basis for the licensing of the Darlington NGS (in addition to the "single / dual failure" criteria). C-6 is currently being revised with a target of early 1994 for public release. Because of the on-going revision, C-6 has not yet been formalized as a "regulatory document". Nevertheless, AECB staff have stated that they would expect any new nuclear power plant to meet the requirements of C-6.

Although the AECB has augmented the "single / dual failure" criteria for safety analyses, the requirements for separate and independent "special safety systems" still are in force. as indicated by regulatory documents R-7, 8, 9. However, in evaluating designs and analyses the AECB staff now tends to emphasize the concept of "defence in depth", as noted above, applying their judgement as to the adequacy of the "defence".

The choice of "generic safety issues" comes from that background. In general they are topics for which AECB staff feel there is an undesirable level of uncertainty or where the performance in existing plants has not been fully satisfactory, thus reducing, in the judgement of AECB staff, the "depth" of the safety defence.

There are no written criteria for the choice, or priority, of "generic safety issues".

The current position of AECB staff is that each of the identified "generic" issues should be resolved before a new nuclear power plant licence would be issued but none are considered sufficiently critical to require refits on existing plants, at this time. This latter determination is based on a judgements (or, more correctly, impressions) of "cost-benefit" (although no evaluations have been conducted and no criteria have been expressed) and of "adequacy of safety" based, primarily, on the level of safety at the time of original licensing.

3. GENERIC SAFETY ISSUES

As indicated earlier, generic issues, in the context of this report, are safety issues which AECB staff have determined as existing in, or being applicable to, several Canadian nuclear power plants.

According to members of AECB staff, when it was decided, in 1991, to compile a formal list of significant generic issues they initially developed a list of over 70 topics that various members of the staff considered as being in an unsatisfactory state. Through collective evaluation and decision making, this list was reduced to eight "generic action items", which were presented in the first report on this matter from the staff to the AECB Board in September 1992 (BMD 92-169), as noted in the Introduction.

That report indicated that one of those eight issues had already been resolved. By the time of the second staff report to the AECB Board, in May 1993, (BMD 93-107) an additional "generic action item" had been added to the list - "Treatment of human factors in Ontario Hydro reliability analyses".

The most recent list of AECB "generic action items" is as follows:

- hydrogen behaviour in containment
- Ontario Hydro's pressure tube inspection program
- use of mercury wetted relays in safety-related systems
- core cooling in the absence of forced flow
- management of ageing
- post-accident filter effectiveness
- reactor operation with a flux tilt
- treatment of human factors in Ontario Hydro reliability analyses.

Of these, AECB staff have indicated that the two of most concern are:

- hydrogen behaviour in containment
- core cooling in the absence of forced flow.

There are hints that if these are not resolved soon action may be mandated for existing plants.

In addition to these "generic action items" these reports also identified some "long-term research issues":

- consequences of pressure tube rupture*
- analysis of emergency core cooling system effectiveness*
- molten fuel - moderator interactions*
- high temperature fuel and fuel channel behaviour
- fission product and aerosol behaviour

- moderator circulation and subcooling

AECB staff have informed the author that the first three (*) are likely to be re-categorized as "generic action items" for the purpose of the management action plan for such topics, as presented in the staff report to the AECB Board in November 1993 (BMD 93-181).

Sections 4, 5, and 6, discuss the background of these "generic" issues. Section 4 provides concise reviews of each of the "generic licensing issues" listed in item "A" of the "scope of work" of the contract. Section 5 covers the additional "generic action items" included in the most recent AECB staff report, and Section 6, the three "long-term research issues" noted above that are likely to be re-designated by AECB staff as "generic action items". Section 7 examines the possible implications of each of these issues for wolsong 3 & 4, with emphasis on the Construction Permit. Finally, Section 8 presents a summary of recommendations regarding the application of these AECB "generic" issues to the licensing review for a Construction Permit for Wolsong 3 & 4.

4. PART "A" ISSUES

4.1. HYDROGEN BEHAVIOUR IN CONTAINMENT

Background

In some postulated accidents in a CANDU reactor, such as blockage of a fuel channel or loss-of-coolant combined with loss of emergency core cooling, the fuel could overheat and react with the coolant, and hydrogen could be produced from the metal - water reaction between the hot zircaloy of the fuel sheaths and pressure tubes and the high temperature steam resulting from the accident. The worst scenario is considered to be a loss-of-coolant accident (LOCA) with an impaired emergency core cooling system (ECCS) such that only minimal water is injected into the fuel channels, all of which is converted to steam.

Hydrogen (deuterium) is also produced, but at much smaller rate and volume, by the radiolytic decomposition of the steam.

This hydrogen could be released into the containment where it would mix with the air atmosphere and possibly produce an explosive mixture. The generally accepted lower limit of flammability of hydrogen in air or air-steam mixture is about 4 % and the minimum concentration for detonation is about 15 %.

Since an explosion in the containment could be a "common cause" initiating event that could lead to failure of both the "process systems" and the "special safety systems", AECB staff consider that there must be a high degree of certainty that it will not occur.

The multi-unit stations of Ontario Hydro have hydrogen igniters, designed to burn the hydrogen before the concentration could build up to an explosive level. The Canadian CANDU 6 plants (Gentilly-2 and Point Lepreau) do not have igniters because the safety analyses conducted at the time of licensing (over ten years ago) concluded that an explosive concentration would not occur, assuming uniform mixing.

AECB staff are sceptical of the validity of the existing analyses. Aspects of the analyses questioned by AECB staff include the assumption of uniform mixing and the reliance on air cooling units to achieve it, since these were not designed to safety criteria for effectiveness and reliability.

This was one many issues in the licensing of the Darlington NGS. The AECB commissioned a study and subsequently required an in-situ gas mixing test in the Darlington containment. The results of that test led to a redesign of the arrangement for the igniters . Experimental work on this topic, sponsored by the CANDU owners, has been underway for several years. Nevertheless, the AECB concern, about the possibility of local "pockets" of hydrogen that could lead to local explosive concentrations, remains.

Status

AECB staff requested Ontario Hydro to reassess the effectiveness of the igniters in its plants and requested Gentilly-2 and Point Lepreau to re-evaluate their analyses which showed that igniters were not needed. In addition they have requested a review of the design and reliability of the local air cooling units which are relied upon in the analyses for mixing and cooling.

Although all licensees have responded partially, AECB staff remain unsatisfied. In particular the progress on the development and validation of more sophisticated computer codes for the concentration throughout the containment (such as the 3-dimensional **PHEONICS** code) is considered inadequate. Reportedly, if there is no significant progress over the next few months the AECB staff may be recommending some retrofit licensing action.

4.2. MERCURY-WETTED RELAYS

Background

Relays using mercury-wetted contacts have been used in the circuits of many safety and safety-related systems in Canadian nuclear power plants. They have had an unacceptable failure rate. The cause of the failures has been identified as a build up of a nickel-mercury amalgam slurry on the contacts which causes them to stick and prevents the relay from opening from its normally energized closed state.

The relays of most concern were those controlling the clutches of the shut-off rods of Shutdown System No. 1. Failure of these relays to open could jeopardize the availability and the effectiveness of the shutdown system. It may be noted, however, that since the control circuits are triplicated it would take the coincidental failure of at least two relays to prevent a rod from dropping when called to do so.

It has been determined that adding tin to the mercury ("tin doping") prevents the formation of the slurry. Also, testing of new dry-contact relays is underway with the objective of identifying other alternatives.

Status

Point Lepreau and Gentilly-2 have replaced all their mercury-wetted relays with tin-doped ones. Ontario Hydro is still reviewing its plants and replacing any that fail. AECB staff are satisfied with the action taken in the CANDU 6 plants and are continuing to monitor closely Ontario Hydro's program.

4.3. CORE COOLING WITHOUT PUMPS (THERMOSYPHONING)

In the safety analyses of all Canadian nuclear power plants there are scenarios where forced flow of the primary system coolant is lost and the cooling of the fuel becomes dependant on the thermosyphoning flow resulting from the temperature differences around the circuit. A typical scenario which depends on thermosyphoning for cooling of the fuel is loss of power to the primary pumps.

When the system is full of coolant, analyses, backed by experiments, indicate that natural thermosyphoning, whether in single or two-phase condition, should provide adequate cooling, even though the pressure drop between headers is small. Those analyses used the **FIREBIRD** thermohydraulics code. AECL CANDU has developed the new two-fluid code **CATHENA** and Ontario Hydro is developing a similar one **TUF**, to replace **FIREBIRD.**

With a partial loss of coolant, neither **CATHENA** nor **TUF** can show with confidence that there will be adequate flow in all channels. Experiments at the RD-14M multiple channel loop at the Whiteshell Laboratories in 1988 and 1989 showed that stagnation could occur.

A few years ago plans were developed to add "pony" motors to the primary pumps of the CANDU 6 plants, fed by a secure source of power, as a means of dealing with loss of power to the pumps. These have not been installed, at least partially because they were not considered to be a complete answer.

Status

Work is still underway to understand the phenomena, to improve the two codes mentioned above, and to validate them further against the RD-14M tests. Modifications were made to the RD-14M loop and further tests conducted in 1992 and 1993.

Several papers given at the **4th International Conference on Simulation Methods in Nuclear Engineering**, held in Montreal, Canada, in June 1993, dealt with this topic, e.g.;

- Recent Developments in the CATHENA Two-Fluid Thermalhydraulics Code
  B. N. Hanna et al

- TUF Version Control and Development Status
  W. S. Liu et al

- Validation of CATHENA Against Feeder Refill Experiments
  J. P. Mallory, N. R. Popov.

As the last paper states, "some details of the experimental results were not captured in these simulations [although] the overall trends were well represented".

AECB staff remain sceptical about the claims for thermosyphoning, especially for accident sequences where the primary heat transport system may not be full initially or where some loss of coolant may occur. They note that RD-14M tests show that, under such conditions, stagnation could occur earlier in some channels than predicted and continue longer than expected. This could have a significant effect on the consequences of these accidents. Also, AECB staff note that, since the electrical heaters used for the simulated fuel in the RD-14M tests had a cut-off temperature of 600 °C, the tests were not fully representative of a real situation.

Members of the AECB staff have indicated that this is considered to be the most serious "generic" issue and it is likely that they will be pushing for an early resolution of the problem.

4.4. POST-ACCIDENT FILTER EFFECTIVENESS

Background

Canadian nuclear power plants use two types of containment. The multi-unit stations of Ontario Hydro use a vacuum building, connected to several reactor buildings, to absorb the pressure resulting from a loss-of-coolant accident and to maintain the containment at atmospheric pressure. The CANDU 6 units have containment buildings designed to hold the maximum pressure arising from such an accident.

The Ontario Hydro plants have emergency filtered air discharge systems that must be used after a LOCA to prevent a pressure rise in the containment caused by in-leakage of air from the service air system and other sources. A few years ago the AECB Board demanded that the Pickering NGS make modifications to extend the time before venting is required. As part of these changes the panels separating the individual reactor buildings from the vacuum building duct were replaced with blow-out panels. Improvements were made in the Darlington NGS design during the licensing process to reduce the inleakage of air.

The containment systems for the Gentilly-2 and Point Lepreau plants do not have a separate filtered venting system. However, for some accident sequences credit is taken in the analyses for venting through the ventilation system exhaust filters.

AECB staff have expressed some concern about the effectiveness of the filters, the testing programs to ensure continued effectiveness, and the reliability of the filters and system under accident conditions. AECB commissioned a report:

- Operational readiness of EFAD Systems
  M. J. Kabat
  AECB research report INFO-0407 Feb. 1992;

which identified a number of deficiencies and made a number of recommendations.

Some of the deficiencies noted were:

- no provision for fire protection
- inadequate periodic testing
- inadequate DOP sampling arrangements
- no method of testing under simulated accident conditions

AECB staff requested licensees to:

- define the range of conditions under which the filter systems must operate and take them into account in analyzing possible failure;
- ensure that testing is done under controlled conditions;
- address the deficiencies identified in the above report.

Subsequently, the AECB commissioned a further related report:

- Operational Readiness of Filtered Air Discharge Monitoring Systems
  J. F. Lafortune, T. J. Jamieson SAIC Canada
  AECB research report INFO-0447 Aug 1993

This report includes a good review of monitoring systems for airborne radioactivity.

Status

Ontario Hydro has submitted a partial response to the AECB request outlined above but claims that the consultant's report is unduly pessimistic. Point Lepreau has proposed a program to show that the ventilation exhaust filters are adequate for post-accident use. Gentilly-2 intends to show that filtered venting will not be required.

AECB staff are still reviewing these replies which, reportedly, are considered inadequate.

4.5. SHUTDOWN EFFECTIVENESS WITH FLUX TILT

Background

Because of their large size it is possible to operate CANDU reactors with a considerable flux "tilt", i.e., a distortion from the normal cosine distribution. On a re-review of shutdown system effectiveness AECB staff identified that for certain loss-of-regulation accidents, in particular a slow uncontrolled increase of power, there are uncertainties in the analyses.

As a consequence, in 1991, AECB staff requested the Canadian nuclear power plant licensees to conduct additional analyses to determine if the regional overpower trip systems remained effective for slow loss-of-regulation accidents starting from a tilted flux configuration.

Status

Gentilly 2 has submitted an interim response and has put in place operating procedures to restrict the extent of tilt allowed during normal operation.

Ontario Hydro has informed that the necessary re-analysis cannot be completed until 1994. Point Lepreau has not yet responded formally. AECB staff are not satisfied with the Ontario Hydro and Point Lepreau responses.

4.6. PRESSURE TUBE INTEGRITY

Background

The actual AECB "generic action item" is "Ontario Hydro's Pressure Tube Inspection Program". However, a major "long-term research issue" is "Consequence of Pressure Tube Rupture". It is reasonable to combine the two topics. AECB staff focused on Ontario Hydro's program because of their particular concern about the pressure tubes in the Pickering "A" and Bruce "A" reactors, the oldest operating plants.

The zircaloy (Zr-2.5Nb) pressure tubes in a CANDU reactor are one of the unique aspects of the design and are critical to the operation. Their operating environment is severe, e.g., high neutron flux, high pressure, relatively high pressure. The basic AECB concern in this "generic action item" is the adequacy of inspection programs to detect the various possible forms of degradation of the pressure tubes.

A particular AECB concern is the possibility of a fast fracture that could lead to consequential rapid failure of the calandria tube and ejection of the hot fuel into the moderator (see item 6.3. below). Of most interest is the formation of hydride blisters, which can lead to brittle fracture. Ductile cracking, where the "leak before break" phenomenon applies, should be detected by the annulus gas monitoring system before catastrophic failure occurs. The AECB generic item focuses on the adequacy of the various in-service inspection techniques, such as CIGAR, to determine the critical factors relating to the on-going state of the pressure tubes, especially those related to a fast fracture.

Status

AECB staff are reviewing the adequacy of the 1992 revision of the Canadian standard CAN3-N285.4 (on periodic inspection) and the criteria of the CANDU Owners Group (COG) document "Fitness for Service Guidelines for Zr Alloy Pressure Tubes in Operating CANDU Reactors". (The latter is proprietary and, therefore, not available.) Reportedly, the COG Guidelines have had several revisions arising from their trial use at Bruce "A" NGS. AECB staff are also continuing to review the ability of the inspection methods and analysis to detect blisters, or better, to predict the likelihood of blister formation. They have suggested that if they are satisfied with the results of their reviews this topic may be closed as a "generic action item" , i.e., deleted from the list.

4.7. PROTECTION AGAINST SECONDARY SIDE FAILURES OUTSIDE CONTAINMENT

Background

As noted in the Introduction this topic is not in the current list of AECB "generic action items". However, it has been the focus of on-going argument between the AECB staff and the operators of the Canadian CANDU 6 plants, Gentilly-2 and Point Lepreau. In both cases there was concern that failure of the main steam line could cause damage to the control room or other important safety or safety-related systems. The Point Lepreau situation was considered the more serious situation since the steam lines ran immediately over the control room, leading to obvious safety concerns.

Status

Because of the obvious problem at Point Lepreau, the AECB staff wanted the steam lines to be moved. However, the AECB Board accepted the utility's argument that this was too expensive and, therefore, was unjustified given the low proability of a failure that could affect the control areas. The AECB Board accepted the licensee's proposal for a program of enhanced inspection. Details of that program are still being developed.

At Gentilly-2 the problem was less clear and the licensees chose to analyze various steam line failure scenarios to show that the consequences were acceptable or that the proability was so low that the particular scenario could be ignored. Several such analyses have been conducted over the past few years and some changes made but AECB staff are still not fully satisfied.

5. OTHER "GENERIC" ISSUES

5.1. MANAGEMENT OF AGEING

Background

All equipment deteriorates over time. AECB staff have expressed concern about the state of knowledge of the nature of this deterioration and about the ability of the existing programs and techniques to detect deterioration in equipment and systems that could adversely affect the safety of Canadian nuclear power plants.
(The topic of 4.6. above, "Pressure tube Integrity", could be considered as a special case of this broad issue.)

In late 1990 AECB staff requested Canadian licensees to provide information on their programs to ensure the continuing safety of their nuclear plants as they got older. Topics specifically identified to be addressed included:

- identification of ageing mechanisms
- re-assessment of in-service inspection and testing programs
- re-assessment of planned maintenance programs.

Licensees were also requested to show that the safety and reliability analyses for their plants remained valid despite the ageing of equipment.

The AECB staff concern reflects the international attention being given to this issue, as evidenced in the meetings held, and reports issued, by the International Atomic Energy Agency.

Status

Ontario Hydro submitted an early report on this issue outlining programs for:

- assuring continuing environmental qualification;
- ensuring adequate margins are included in designs and analyses;
- special inspection and maintenance programs;
- the examination of equipment from decommissioned reactors.

Ontario Hydro is developing a "nuclear plant life assurance program" that focuses on the inspection, monitoring and analysis of selected components judged to be important for long-term reliability and safety. Ontario Hydro has also indicated that it is developing an overall approach to the analysis, determination and management of ageing. AECB staff are awaiting further information on these programs.

Hydro-Quebec has also submitted a preliminary program for Gentilly-2 which is still under review by AECB staff.

5.2. HUMAN FACTORS IN RELIABILITY ANALYSES

Background

This "generic issue' originated as a specific concern associated with analyses submitted by Ontario Hydro for improvements to the filtered air discharge system at the Pickering NGS. AECB staff questioned some aspects of the approach used for the reliability of operators and maintainers. Ontario Hydro noted that the approach was the same as that taken in the Darlington Probabilistic Safety Evaluation (DPSE) and was based on the method used by the United States Nuclear Regulatory Commission (USNRC). (It may be noted that the AECB did not review closely the DPSE because it was not considered a "licensing document".)

In early 1993 the AECB engaged the services of Dr. Alan Swain, author or co-author of several NUREG reports* issued by the USNRC on human reliability analysis. Reportedly, Dr. Swain questioned Ontario Hydro's interpretation and use of the NUREG reports. AECB staff then requested Ontario Hydro to respond to a number of points raised by Dr. Swain. At the time of writing they are still waiting for a reply.

* Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications
A. D. Swain and H. E. Guttman
USNRC NUREG/CR-1278 revised 1983

Accident Sequence Evaluation Program Human Reliability Analysis Procedure
A. D. Swain
USNRC NUREG/CR-4772 1987

Subsequently AECB staff decided to identify the general question of "human reliability" as a generic issue.

Status

AECB staff have agreed to defer further action on this topic until they have reviewed Ontario Hydro's response to the points referenced above.

6. SELECTED 'LONG-TERM RESEARCH' TOPICS

As noted in the Introduction, the following "long-term research" topics are likely to be identified as "generic action items" next year.

6.1. CONSEQUENCES OF PRESSURE TUBE FAILURE

Background

This topic is related to those of section 4.6. "pressure tube integrity" and 6.3. "molten fuel - moderator interactions". As for most of the some other topics it arises primarily from a sense of uncertainty, in this case, regarding the conclusions of the existing analyses about the consequences of a pressure tube failure.

A basic concern is that catastrophic failure of a pressure tube might lead to subsequent failure of the surrounding calandria tube. This could allow hot fuel to be ejected into the cool moderator with undetermined consequences (see section 6.3. below). It is feared that the forces resulting from the interaction between the hot fuel and the cool moderator water could cause damage to the shutoff rod guide tubes, thus impairing the effectiveness of shutdown system 1. The analyses that have been submitted argue that this would not happen. Nevertheless, AECB staff are not entirely convinced and have been pushing for, or proposing, more studies and experiments over the past few years.

Some related experiments were conducted in Japan with Canadian participation but AECB staff did not feel that the results were truly relevant.

This concern has existed, to various degrees, since the beginning of the CANDU program three decades ago. Several sets of experiments were conducted in the 1960s specifically to address this issue. Those showed that failure of a pressure tube at near normal pressure would not cause damage to adjacent fuel channels. Despite the positive results of those experiments some members of AECB staff continue to express concern and have managed to have the topic identified as a "long-term research" item.

Status

A further set of experiments, sponsored by Canadian licensees, are being conducted at the Stern Laboratories (in Hamilton, Canada). Although AECB staff are not fully satisfied with the nature and scope of these experiments further action has been deferred until the results are available.

6.2. ANALYSIS OF ECCS EFFECTIVENESS

Background

Some AECB staff continue to be dissatisfied with the existing analyses for major loss-of-coolant accidents. Those performed by Ontario Hydro (e.g., for Darlington) used a limiting approach with very conservative assumptions to determine an upper bound of fuel and fuel channel temperatures. The results were predictions of temperatures higher than those for which there is good experimental data. Without such an experimental basis AECB staff feel that it can not be shown, with confidence, that the assumptions were, actually, conservative, i.e., that the consequences actually are an upper bound. Also, this approach does not give a good insight into the specific performance of the emergency core cooling system.

Ontario Hydro has developed an advanced, two-phase, thermohydraulics computer code **TUF** and is redoing the analyses for the Darlington NGS.

For the Gentilly 2 and Point Lepreau CANDU 6 plants, AECL CANDU did the analyses, initially with the **FIREBIRD** code. FIREBIRD had been validated against tests on the RD-12 and RD-14 loops and the Cold Water Injection Test facility at the Whiteshell laboratories. That code has some limitations, however, particularly in handling two-phase situations. Also, the RD-14 loop had only two "channels" and has been considered as not being truly representative of a CANDU. A modified version, RD-14M, with several channels was built in 1988. With the support of the utilities, AECL CANDU has developed the two-fluid code **CATHENA** which is being validated against tests in the RD-14M loop. (See recent papers referenced in section 4.3. regarding validation of **CATHENA**.)

Status

AECB staff are dissatisfied with the progress by both Ontario Hydro and AECL CANDU (for the other utilities) and have begun to press for action. Whether or not this will lead to a demand for design or operating changes will depend on the ability of the new codes to analyze, to the satisfaction of AECB staff, all of the postulated scenarios for which ECCS is required.

6.3. MOLTEN FUEL - MODERATOR INTERACTION

Background

This topic is related to topic 6.1 above, "Consequences of Pressure Tube Failure". The AECB staff concern arises particularly from their lack of confidence in the analyses for the consequences of a complete blockage of a fuel channel. In such an event the fuel would overheat, melt, and cause failure of both the pressure tube and the calandria tube. Molten fuel would then come into contact with the cool moderator with the potential of a "steam explosion" which could damage the calandria and shutoff rod guide tubes of shutdown system 1.

A report* by a consultant to the AECB concluded that steam explosions were possible, contradicting an earlier report by Ontario Hydro. The latter assumed that any fuel ejected into the moderator would fragment into small particles which would be cooled quickly. Large pieces would increase the likelihood of a steam explosion.

* Molten Fuel-Moderator Interaction: An Investigation of the Potential for Steam Explosion
R. Knystautas, AECB research report INFO-0227, Feb. 1987

Status

AECB staff remain concerned on this issue and are pressing for more information to clarify the existing uncertainties. They wish assurance that a single failure (fuel channel blockage) will not result in severe damage to the reactor and the consequential release of radioactive material. No licensing action has been proposed but AECB staff have indicated that they would require that this issue be resolved before any new plant were licensed.

7. APPLICATION TO WOLSONG 3 & 4

7.1. HYDROGEN BEHAVIOUR IN CONTAINMENT

The Wolsong 3 & 4 PSAR states that, "a number of thermal igniters will be installed throughout the reactor building, with the exact number, distribution and location to be determined by analysis". It is proposed that these be initiated by the "containment triggering signal" (high reactor building pressure / high reactor building activity) and have a manual shutoff. According to AECL CANDU officials the analysis for the distribution and concentration of hydrogen will be done by both the one-dimensional **PRESCON-2** code and the new **PHOENICS** three-dimensional code (which is still being validated).

AECL CANDU officials have informed the author that the Class III electrical system will have sufficient power for up to 70 igniters and the tentative proposal is to have 44 igniters installed. If so, this should meet the concerns identified by AECB staff, i.e., should ensure there are no pockets of hydrogen at an explosive concentration.

The Wolsong 3 & 4 PSAR states that 4 large air coolers in each reactor vault and 8 medium-sized ones in the boiler room will be on Class III power for reliability, but the 19 smaller ones in other rooms will be on Class IV. For analysis purposes, the last group should be considered unavailable.

KINS staff should review carefully the details of the relevant analyses and the evidence for the validation of the **PHOENICS** code over the full range of physical situations and accident conditions anticipated. The specific design and installation of the igniters and of the air coolers should be reviewed, noting that these are, in essence, part of the containment, and should, therefore, meet the appropriate "special safety system" criteria for reliability, effectiveness and testability.

7.2. MERCURY-WETTED RELAYS

AECL officials have informed the author that all mercury-wetted relays will be tin-doped. If so, this issue could be considered resolved.

KINS staff should ensure that any mercury-wetted relays used in safety or safety-related circuits are definitely tin-doped. If any dry contact relays are used, a careful review of their expected reliability, under all expected conditions, should be conducted.

7.3. THERMOSYPHONING

One of the accident scenarios in which credit is taken for thermosyphoning is "complete loss of class IV electrical power". For this case the Wolsong 3 & 4 PSAR states, "after a few minutes a steady thermosyphoning flow of primary coolant is established". The author has not seen any supporting analysis or argument for this statement. AECL CANDU officials have said that these cases will be re-analyzed using the new **CATHENA** two-fluid code. As noted in the review (section 4.3.), both experiment and simulation leave doubts about thermosyphoning if the primary heat transport system is not full.

KINS staff should review carefully the most recent information on the validation of the **CATHENA** code and the relevant tests in the **RD-14M** loop. If uncertainties remain regarding thermosyphoning under any accident condition the applicants should be required to propose methods of precluding or minimizing the possibility of situations requiring thermosyphoning (such as loss of power to the main pumps). Such measures could include design changes, operating restrictions or both.

7.4. POST-ACCIDENT FILTER EFFECTIVENESS

This topic is not addressed specifically in the Wolsong 3 & 4 PSAR. Section 15.2.1.4.A.5.4, dealing with single channel events does note that some activity will be trapped in the heavy water vapour recovery system ducting, but it is a relatively small quantity.

AECL CANDU officials have informed the author that Wolsong 3 & 4, similar to Wolsong 2, will have an arrangement whereby the instrument air would be isolated on an accident that triggers containment closure. When isolated, a special compressor will maintain instrument air by drawing on reactor building air - thus not adding any air to the building. In earlier CANDU 6 plants instrument air added about 170 std. m^3 /hr, which, with the building sealed, would result in a pressure increase of 0.35 kPa/hr. This new arrangement will minimize, if not eliminate, the need to discharge air after an accident.

AECL CANDU officials have stated that venting of the containment will not be needed after any accident but that the system is there if the operators wish to use it.

KINS staff should satisfy themselves about the adequacy and reliability of the "accident" instrument air system, and confirm that, with that system, there is no need to vent the containment after any accident. Looking ahead to operation, there should be strict administrative controls over the use of the stack discharge system after an accident. Venting should not be permitted unless there are good monitoring systems for the containment atmosphere and the air discharge systems, so that proper evaluation can be made at the time regarding the potential release of radioactive material. KINS staff should review carefully these monitoring systems. If they are not fully satisfied with the expected effectiveness and reliability offered by the design, venting should be prohibited.

7.5. SHUTDOWN EFFECTIVENESS WITH TILT

The author has not found any explicit analysis or review of this particular scenario. It is noted that there is a considerable region of reactivity insertion rates where SDS 2 has only one effective parameter.

Unless convincing analyses are presented to cover the scenarios of concern - slow uncontrolled increase in power starting from a tilted flux condition - KINS should impose conditions on the acceptable operating state. At the Construction Permit stage the applicant should be required to show how they intend to deal with this potential situation.

7.6. PRESSURE TUBE INTEGRITY

The Wolsong 3 & 4 PSAR presents extensive information about the design and manufacture of the pressure tubes (section 5.3.) but little on the required (assumed) in-service inspection programs. As noted in section 4.6. of this report AECB staff are still reviewing the 1992 revision of the Canadian standard CAN N285.4 on periodic inspection and the COG "Fitness for Service Guidelines". Both of these deal with operational programs. At the Construction Permit stage the design and manufacturing standards are important and should be reviewed carefully to ensure that the requisite initial quality is achieved and that ALL the initial conditions are measured and recorded to serve as a base-line for in-service inspections.

KINS staff should require and review:

- design measures, material and manufacturing specifications, intended to minimize hydriding, blisters, cracking, corrosion, creep, and other undesirable developments (to ensure their appropriateness, adequacy and completeness);
- methods to ensure garter springs will not move;
- proposed in-service inspection methods (to ensure that these are practicable and reflect all of the assumptions made at the Construction Permit stage).
- proposed "fitness for service" criteria.
- details of initial inspections (to ensure as-installed pressure tubes meet all required specifications and to serve as a base-line for in-service inspections).

7.7. SECONDARY SIDE FAILURES OUTSIDE CONTAINMENT

It is noted that the main steam lines of Wolsong 3 & 4 will be routed around the end of the Service Building and that their support structure will be independent of that of the building.

This would appear to eliminate the concerns of AECB staff related to the Canadian CANDU 6 plants and should be acceptable.

7.8. MANAGEMENT OF AGEING

While this issue applies principally to the operating phase there are a number of factors that should be considered at the Construction Permit stage. Among these are:

- ensuring that the design and specifications of safety and safety-related equipment take into account the likely deterioration due to ageing;
- requiring appropriate environmental qualification testing and evaluation;
- ensuring that the design will permit necessary in-service inspections and testing.

The Wolsong 3 & 4 PSAR does not specifically address this topic.

It is essential that there be good documentation of all the initial conditions of all safety-related equipment to serve as a basis for evaluations of in-service inspections.

KINS should require a reliability analysis, including a FMEA (failure modes and effects analysis) of, at least, all of the "special safety systems" (including all sub-systems and necessary support systems). Descriptions of proposed programs for in-service inspection and testing should be required including explanations of how the design accommodates these proposed programs, for careful review by KINS staff. Clear and specific performance specifications should be required for all safety and safety-related equipment. Documentation of initial testing or evaluation should be required to serve as a base-line for in-service inspections and testing. (A "living PSA" would be desirable.)

7.9. HUMAN FACTORS IN RELIABILITY ANALYSES

The design of the Wolsong 2, 3 & 4 control rooms and the layout of equipment are similar to those of Wolsong 1. They reflect considerable practical feed-back from the operation of earlier plants, even if formal human factors analyses were not performed. Some modifications have been made based on the experience with Wolsong 1 and the equivalent Canadian plants.

The Wolsong 3 & 4 PSAR refers to some use of man-machine interface evaluation in the Wolsong 2 PSA but this appears to be well short of a comprehensive human reliability analysis such as is required by the USNRC.

In the interest of consistency with other plants (PWR) KINS may wish to require a formal human reliability analysis but it may be noted that the AECB is not pressing for this. The operational advantage of similarity with Wolsong 1 and 2 probably outweighs any change based on theoretical analyses.

7.10. CONSEQUENCES OF PRESSURE TUBE FAILURE

This issue stems from AECB staff concerns about possible consequences of a catastrophic failure of a pressure tube and, therefore, is related to Item 7.6. above which deals with "pressure tube integrity". If appropriate measures are taken in the design, manufacture, installation, and operation. and if in-service inspection is carried out properly, the likelihood of a catastrophic failure of a pressure tube should be very low. Experiments are being conducted to provide more information on possible consequences of a catastrophic failure of a pressure tube but the results will not be known for some time.

In the section of the Wolsong 3 & 4 PSAR dealing with the effectiveness of SDS 1 it is argued that even if one row of shut-off rods were incapacitated and one other rod unavailable there would still be sufficient negative reactivity to shutdown and hold-down the reactor in the event of a pressure tube and calandria tube failure.

AECB has accepted, at least for the time being, the existing situation. It would appear appropriate for KINS to do so also.

7.11. ANALYSIS OF ECCS EFFECTIVENESS

The effectiveness of the emergency core cooling systems in CANDU nuclear power plants has been an on-going concern of AECB staff. Despite many experiments in the **RD-14M** multi-pass loop and analyses with the **FIREBIRD** code a number of uncertainties remain. The more advance, two-fluid, **CATHENA** code provides more complete simulation but the code is still being validated for certain conditions.

AECL CANDU officials have said that **CATHENA** will be used in analyses where ECCS is required, but that some "bounding" analyses will also be applied.

KINS specialists should examine closely the reports on the validation of **CATHENA** to ensure themselves that the code does represent all the phenomena involved and is applicable for all of the conditions of all of the critical accident scenarios. Where "bounding" analyses are presented KINS staff should review them critically to ensure that they really do present an upper bound of effects.

(As an aside - a continuing problem in Canadian nuclear power plants has been the reliability of the ECCS. It would appear appropriate to require a good reliability analysis of the Wolsong 3 & 4 ECCS.)

7.12. MOLTEN FUEL - MODERATOR INTERACTION

This issue is related to the topics "pressure tube integrity" (7.6. above) and "consequences of pressure tube failure" (item 7.10. above). It is not likely that this issue will be resolved in Canada for some time. Given that situation and the fact that the AECB staff appear prepared to await the results of further experiments and studies, it would seem appropriate for KINS to ignore this topic at the Construction Permit stage of Wolsong 3 & 4.

8. SUMMARY OF RECOMMENDATIONS

8.1. HYDROGEN BEHAVIOUR IN CONTAINMENT

8.1.1. The Construction Permit should include a condition that acceptable analyses for the number and location of the hydrogen igniters must be submitted (suggest within one year).

8.1.2. Before granting the Construction Permit the design sshould be checked to confirm that Class III power is available for up to 70 igniters, in all areas of the containment.

8.2. MERCURY-WETTED RELAYS

8.2.1. Before granting the Construction Permit KINS should ensure that the specifications for any mercury-wetted relays used in safety-related systems call for them to be tin-doped.

8.3. THERMOSYPHONING

8.3.1. The Construction Permit should contain conditions requiring the submission of acceptable evidence of validation of the **CATHENA** code and acceptable analyses for fuel cooling in the event of a complete loss of Class IV electrical power (suggest within one year). The Construction Permit condition should require design changes if the analyses are not acceptable.

8.4. POST ACCIDENT FILTER EFFECTIVENESS

8.4.1. Before the Construction Permit is issued KINS should ensure that there is no need to vent the containment after any accident. If venting is proposed as an operational option the design of the radioactivity monitoring systems for the containment air and the air discharge system should be reviewed carefully for expected effectiveness, reliability and ease of use.

8.5. SHUTDOWN EFFECTIVENESS WITH TILT

8.5.1. Information should be required on how the Wolsong 3 & 4 shutdown systems will cope with a slow uncontrolled rise in power starting from a tilted-flux condition.

8.6. PRESSURE TUBE INTEGRITY

8.6.1. The Construction Permit should contain a condition requiring the submission of the following at least one year before the Operating Licence:

- details of final inspections prior to operation;
- details of proposed in-service inspection methods;
- proposed criteria for "fitness for service".

8.6.2. Before the Construction Permit is issued KINS should check:

- the design and specifications for the garter springs to ensure they will not move;
- specifications for pressure tube material, with reference to experience on hydriding, corrosion, creep, etc.

8.7. SECONDARY SIDE FAILURES OUTSIDE CONTAINMENT

- no recommendations

8.8. MANAGEMENT OF AGEING

8.8.1. The Construction Permit should require the submission, prior to the Operating Licence, of reliability analyses, including failure mode and effects analyses, of all special safety systems, including sub-systems and necessary support systems.

8.8.2. The Construction Permit should require detailed documentation of the initial condition of all safety-related equipment prior to the Operating Licence.

8.9. HUMAN RELIABILITY ANALYSES

- no recommendation

8.10. CONSEQUENCES OF PRESSURE TUBE FAILURE

- no recommendation

8.11. ECCS EFFECTIVENESS

8.11.1. The Construction Permit should require (suggest prior to the installation of any ECCS equipment) submission of:

- satisfactory validation of **CATHENA** code for all relevant situations;
- complete analyses of ECCS effectiveness for all LOCAs.

8.12. MOLTEN FUEL - MODERATOR INTERACTION

- no recommendation

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9. Bibliography

ATOMIC ENERGY CONTROL BOARD (of Canada) **documents**

BMD 92-169 Report on Generic Safety Issues Affecting Power Reactors
Sept. 1992

BMD 93-107 Annual Report on Generic Safety Issues Affecting Power Reactors
May 1993

BMD 93-181 Management of Generic Safety Issues Affecting Power Reactors
Oct. 1993

AECB - 1045 Reactor Safety and Licensing Requirements, D. G. Hurst, F. C. Boyd
1972

Regulatory Document R - 7 Requirements for Containment Systems for CANDU Nuclear Power Plants, 1991

Regulatory Document R - 8 Requirements for Shutdown Systems for CANDU Nuclear Power Plants, 1991

Regulatory Document R - 9 Requirements for Emergency Core Cooling Systems for CANDU Nuclear Power Plants, 1991

Regulatory Document R - 10 Use of Two Shutdown Systems on Reactors, 1977

Consultative Document C - 6 Requirements for the Safety Analyses of CANDU Nuclear Power Plants, 1980

INFO - 0319 Canadian Programs on Understanding and Managing Aging Degradation of Nuclear Power Plant Components, J. A. Chadha, J. Pachner, (paper) 1989

INFO - 0480 Rapport annuel du personnel de la CCEA sur la centrale nucleaire Gentilly 2, 1992, Sept. 1993

INFO - 0482 AECB Staff Annual Report on the Point Lepreau Nuclear Generating Station for the Year 1992, Sept. 1993

INFO - 0357 Severe Accident Considerations in Canadian Nuclear Power Reactors, A. M. Omar, M. P. Measures, C. K. Scott, M. J. Lewis, (conference paper) Aug. 1990

AECB RESEARCH REPORTS

The following documents are reports on research commissioned by the AECB, but not necessarily endorsed by the AECB.

INFO - 0312 Numerical Computations of Underwater Explosions Due to Fuel-Coolant Interactions
J. H. S. Lee, D. L. Frost, R. Knystautas, A. Teodorczyk, G. Ciccarelli; P. Thibault, J. Penrose, Mar. 1989

INFO - 0327 Dispersion, Mixing and Intentional Ignition of Hydrogen in the Darlington Reactor Vault
J. H. S. Lee, R. Knystautas, Mar. 1989

INFO - 0351 Ballooning of CANDU Pressure Tubes - Model Assessment
R. S. W. Shewfelt, Feb. 1990

INFO -0351-2 Design of Experiments and Equipment to Test the Ballooning Characteristics of CANDU Pressure Tubes
C. F. Forrest, F. Stern, R. G. Hart, Jan. 1992

INFO - 0370 Fuel Channel Refilling - Data Analysis
M. Shoukri, A. Abdul-Razzak, Apr. 1990

INFO - 0382 Propagation Mechanisms of Molten Fuel / Moderator Interactions
D. L. Frost, G. Ciccarelli, June 1991

INFO - 0407 Operational Readiness of EFAD Systems
M. J. Kabat, Feb. 1992

INFO - 0429 Human Performance Monitoring in Nuclear Power Plants
D. C. Donderi, A. Smiley, D. J. Ostry, Mar. 1993

INFO - 0447 Operational Readiness of Filtered Air Discharge Monitoring Systems
J. F. Lafortune, T. J. Jamieson, Aug. 1993