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L'ÉNERGIE ATOMIQUE DU CANADA LIMITÉE

RESEARCH AND DEVELOPMENT INTO POWER REACTOR FUEL PERFORMANCE

Travaux de R&D touchant la performance du combustible des réacteurs de puissance

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Chalk River, Ontario

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Résumé

Le combustible nucléaire d'un réacteur de puissance doit fonctionner fiablement en service normal et les conséquences de phénomènes anormaux doivent être étudiées et évaluées. Le fonctionnement actuellement très fiable de l'UO_2 naturel dans les réacteurs de puissance de type CANDU a réduit la nécessité de poursuivre les travaux de R&D dans ce domaine. Cependant, un noyau de compétence doit être conservé, par exemple pour former le nouveau personnel, pour être en mesure de répondre à des circonstances imprévues et pour pouvoir participer au développement commercial de nouvelles idées. L'évaluation de la performance du combustible dans des conditions accidentelles, exige que des recherches soient faites sur de nombreux aspects du comportement des matériaux, du combustible et des produits de fission. Par ailleurs, les connaissances ainsi acquises doivent être introduites dans les codes machine employés pour évaluer les conséquences de n'importe quel accident hypothétique. Ces travaux prennent actuellement de l'ampleur. On a beaucoup de données provenant d'études faites hors réacteur à des températures allant jusqu'à environ 1500°C. Cependant, le besoin de vérifications en réacteur et la nécessité d'enquêter sur les accidents à haute température ont rendu nécessaire la construction d'une nouvelle grande boucle d'essais en réacteur et la mise sur pied de programmes associés hors réacteur.

Etant donné que les programmes relatifs à la performance normale et au fonctionnement dans des conditions accidentelles sont de nature générique, ils s'appliqueront aux cycles de combustibles avancés. Les travaux de R&D passeront donc graduellement du cycle actuel des réacteurs de puissance en service au cycle à base de thorium prévu pour la prochaine génération de réacteurs.

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ABSTRACT

The nuclear fuel in a power reactor must perform reliably during normal operation, and the consequences of abnormal events must be researched and assessed. The present highly reliable operation of the natural UO_2 in the CANDU Power reactors has reduced the need for further work in this area; however a core of expertise must be retained for purposes such as training of new staff, retaining the capability of reacting to unforeseen circumstances, and participating in the commercial development of new ideas. The assessment of fuel performance during accidents requires research into many aspects of materials, fuel and fission product behaviour, and the consolidation of that knowledge into computer codes used to evaluate the consequences of any particular accident. This work is growing in scope, much is known from out-reactor work at temperatures up to about 1500°C, but the need for in-reactor verification and investigation of higher-temperature accidents has necessitated the construction of a major new in-reactor test loop and the initiation of the associated out-reactor support programs.

Since many of the programs on normal and accident-related performance are generic in nature, they will be applicable to advanced fuel cycles. Work will therefore be gradually transferred from the present, committed power reactor system to support the next generation of thorium-based reactor cycles.

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Research and Development Into Power Reactor Fuel Performance

1. INTRODUCTION

The power developed by a nuclear reactor is produced in nuclear fuel elements by fission. This thermal energy is transferred by the surrounding coolant to steam generators, and thence to turbines where the energy is converted to electricity.

The nuclear fuel is a very compact energy source. A single CANDU* fuel bundle weighs about 25 kg, yet operates at up to 800 kW heat output and produces in its lifetime of about one year as much thermal energy as would 600 tons of coal. Because of the intense heat generation, the fuel has to be extremely well cooled to prevent it operating at excessive temperatures. Therefore, in a CANDU reactor (Figure 1), the fuel is contained in an array of pressure tubes, through which heavy water at about 10 MPa pressure, 280°C, flows at velocities of up to 25 m/s.

This report summarizes the evolution of the CANDU fuel design, and the progress that has been made in understanding and improving the performance of the fuel. First, however, we should summarize the manner in which the design and development process has evolved and how responsibility for various aspects of the work has shifted.

Lewis' perception (1) that economic nuclear power depends on low fuelling cost, which is most easily achieved with natural uranium and by strict attention to neutron economy, set the stage. The decision in 1954, to use natural, rather than enriched, uranium, led to the concept of refuelling while the reactor was at power. Refuelling led to horizontal fuel channels, (pressure tubes) through which short (500 mm long) bundles were fed (Figure 2).

The CANDU reactor, and hence the fuel design, is significantly different from the pressurized water reactor (PWR). The latter is a pressure vessel, where fuel can only be removed by first taking off the lid of the reactor. The fuel is oriented vertically, in full-length rods, for easy removal. Because refuelling necessitates a long reactor shutdown, and because the fuel has to be enriched to operate with a light water moderator, fuel is designed to achieve burnups (total power outputs over the fuel's lifetime) about four times as great as the CANDU fuel, which in contrast can be replaced with no loss of power output.

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* CANadian Deuterium Uranium reactor system

In Canada the development of nuclear fuel began about 25 years ago with the design and manufacture of the first charge for the demonstration power reactor NPD*. Initially Atomic Energy of Canada Limited worked closely with private industry, funding development and passing on know-how developed at Chalk River. In subsequent years as designs improved and more fuel was manufactured, a division of responsibility evolved. The AECL laboratories concentrated on fundamental studies related to fuel performance, while manufacturing and design associated with production became the responsibility of private industry, primarily Canadian General Electric Co. (CGE) and Westinghouse Canada Inc. (WECAN). The AECL Engineering Company took the responsibility for design of the reactor, with associated fuel handling and product specification, and for assessment of the performance of the reactor and the fuel under abnormal operating conditions. At the same time, utilities, initially Ontario Hydro, developed fuel management, procurement, and design and assessment capabilities, but again relied on the laboratories for R&D work. It is this R&D work that we address in this paper.

2. *FUEL DESIGN*

For economical power reactor operation, fuel has to be designed to operate at as high a power output as possible. This constraint necessitates considerable work on optimizing the design of the CANDU fuel bundle from a thermalhydraulics viewpoint. This aspect of the work will be described in a separate report on R&D in thermalhydraulics (2); the remainder of the present document will be concerned with the other factors that influence the performance of the fuel.

The original fuel charge for the first CANDU, a 22 MW(e) Nuclear Power Demonstrator (NPD) reactor, consisted of "wire-wrapped", 7-element bundles (3) in the outer zone and 19-element, wire-wrapped bundles in the centre of the core (Figure 3). Using the performance knowledge obtained on NPD, the 19-element, wire-wrapped bundle was modified for the 200 MW(e) Douglas Point Reactor to improve the bundles's thermal performance. The wire wrap, which was included to prevent inter-element contact (and local coolant starvation) was replaced by brazed-on spacer and bearing pads, which facilitate high volume production with no effect on thermal efficiency.

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* Nuclear Power Demonstrator

The Pickering reactors were designed to have a larger-diameter pressure tube and hence fuel bundle, so the same size fuel element was used, but with 28 rather than 19 elements assembled into a bundle. The Bruce reactors employ the same diameter pressure tubes as Pickering, but because of the higher power density a 37-element bundle was developed. With minor changes, the bundle used in the 600 MWe G-II type reactor at Gentilly for Hydro Quebec, Point Lepreau for New Brunswick Electric Power Commission, Embalse, Argentina for Comision National Energia Atomica and Wolsung, Korea for Korea Electric Power Company is similar to that used successfully in Bruce (Figure 2).

Fabrication developments proceeded in parallel with the design evolution (3). The original core load for NPD utilized inert gas fusion welding for both rod closure and bundle assembly. All subsequent fuel was manufactured using resistance welding for both end cap to sheath and end plate to end cap joints. This method of welding is fast, inexpensive, and lends itself to automation. Spacers and bearing pads are brazed in place. The emphasis is on improved productivity and process control through the use of automation and advances in machine control technology, such as micro-processors (4).

3. RESEARCH IN SUPPORT OF FUEL DESIGN

The CANDU fuel design is based on extensive irradiation testing of over 1000 individual rods, some highly instrumented, and 400 bundles, some with replaceable outer rods. The development over a 25 year period involved 650 professional man-years and 180 million dollars. The synergistic association of the engineers, who did the irradiation work, and the scientists, who studied the underlying mechanisms, quickly established a semi-empirical approach that relied on simple models, realistic test results and an understanding of the physical mechanisms; this understanding then enabled us to concentrate on the most important, and most cost-effective ways of improving the fuel design.

Much of CRNL's success has been due to the availability of excellent test facilities in the NRX and NRU reactors. These facilities or 'loops' are individually cooled through tubes in which the fuel experiments can be performed under closely controlled and monitored conditions. Often the experiments will necessitate development of specialized measurement techniques. A typical fuel experiment may be in-reactor for up to one year, but from its inception to final examination usually takes about three years.

AECL has been at the forefront of development of Zircaloy-sheathed UO_2 fuels, and although the CANDU fuel design is unique in many respects, there is a broad base of fundamental information common to all water-cooled reactor fuel types, to which AECL has made significant contributions and from which AECL had drawn much useful information.

The complex interactions that dictate fuel element performance are illustrated in Figure 4. Fuel temperature is the dominant variable. This is governed by the fuel power output, the thermal conductivity of the fuel, the efficiency of the heat transfer from the sheath surface to the coolant, and from the UO_2 fuel itself to the sheath. During irradiation, uranium atoms split to form fission products, some of which (especially the noble gases) may be released from the UO_2. Those that remain cause the UO_2 to swell; those that are released cause an increase in the gas pressure within the sealed fuel element. This gas pressure may cause the sheath to move out of contact with the fuel pellet, but at the same time the release of the fission products will increase fuel pellet swelling, a process which tends to restore fuel to sheath contact and increase heat transfer.

Our understanding of the quantitative significance of the effects illustrated in Figure 4 has been gathered into predictive computer codes, which serve the two purposes of providing best estimates of the effect of any particular operational conditions, and provide a vehicle for assessing whether new experimental information requires us to re-think our understanding of any particular physical process. The following sections detail some of the work done to develop the fundamental understanding, together with particular problems posed by reactor operation. In general, this section deals with fuel performance under normal operating conditions, while Section 4 discusses topics pertinent to fuel performance under abnormal conditions (i.e. during accidents). However some topics are common to both areas, and are discussed wherever is the most appropriate.

3.1 UO_2 Operating Temperature

The dominant variable that affects UO_2 behaviour is temperature, which is a function of thermal conductivity. Early experiments (5) at CRNL helped define the dependence of thermal conductivity on factors such as local temperature, fuel porosity and minor variations in fuel chemical composition (O:U atom ratio). International concensus having been reached, this aspect of the work has been completed.

However fuel-to-sheath heat transfer also affects fuel temperature quite significantly. For many years, most workers depended on relationships developed by Ross and Stoute (6) in a laboratory study at CRNL. Campbell et al. (7) devised an ingenious method of repeating the measurements in the NRX reactor at CRNL and confirmed the earlier results when there was a gap between fuel and sheath, but showed that contact conductance was greater in-reactor than had been found in the laboratory. Further work, which is continuing, is indicating that one reason heat transfer improves in-reactor is because the contacting surfaces become smoother. This is of importance, in that it indicates that as-fabricated pellet surfaces may not have to be ground to the present fine surface finish. Another factor which affects heat transfer is the CANLUB coating applied to the inner surface of the sheath (Section 3.3). Experiments have indicated that heat transfer can be improved by the siloxane coatings, but is not significantly affected by graphite. As new coatings are developed, this aspect of performance will need to be checked.

3.2 Fuel Expansion

CANDU fuel is designed with small clearances and UO_2 diametral expansion is intended to close the pellet-to-sheath gap at partial power so that there is some elastic strain in the Zircaloy on reaching full power. This promotes good fuel-to-sheath heat transfer.

The first mechanistic models for UO_2 expansion recognized that the high temperature core of the pellet behaved plastically and interacted with the sheath through an elastic cracked region (Figure 5). The rate of power increase (ramping) was found to be important (8), since rapid ramps resulted in less plastic flow and hence larger diametral expansions. Furthermore, it was demonstrated that irradiation enhanced the elimination of the fine porosity that remained after fabrication (9), a phenomenon that became recognized as important when fuel densification caused some temporary operating restrictions in Light Water Reactors.

Recently, a successful development of an In-Reactor Diametral Measurement Rig (IRDMR) - Figure 6 - has enabled a detailed study of the factors affecting fuel expansion or shrinkage (densification). This work (10) has shown that densification due to elimination of porosity is more rapid than anticipated, and that power cycling of fuel should not cause additive increases in fuel expansion (and hence sheath strain).

3.3 Fuel Performance During Power Changes (Power Ramps)

Several years ago it was found that an increase in fuel power could cause defects in fuel elements irradiated to moderate burnup at low power. This is thought to be due to fuel expansion caused by the increase in power (and temperature). This expansion stresses the sheath in tension and, in the presence of fission products, gives rise to a phenomenon called "stress corrosion cracking". As a CANDU fuel bundle is moved in the reactor (the channel being intermittently 'charged' with new fuel), it experiences power increases, which, in the case of the Douglas Point reactor, led to a failure rate in excess of the acceptable level of 0.1% of all bundles irradiated (11).

Analysis of the operational data from defective and intact fuel bundles led to FUELOGRAMS (12) which define defect thresholds and bundle defect probabilities in terms of the ramped power, power increase, burnup and dwell time at high power. Several remedies were tested, among the more practical and successful being introduction of a thin graphite or siloxane coating ('CANLUB') on the inside bore of the fuel element sheath. This coating is now applied to all production CANDU fuel and has virtually eliminated power-ramp failures. Research has defined most of the fabrication requirements and operating limits of such coatings, but there remains a certain amount of work, particularly to define the operating limits for ramping higher burnup fuel. This information can also be applied to development of fuel for use in the Advanced Fuel Cycle (13), based on ThO_2 or enriched UO_2 fuel.

Investigations of fuel defect mechanisms, both at CRNL and elsewhere in the world, identified iodine and cesium-cadmium mixtures as being the probable causes of stress-corrosion cracking (14). These elements are produced by fission and can be released to the fuel-to-sheath interface. The mechanism by which the CANLUB coating material reduces the incidence of cracking has been researched extensively but inconclusively; in the absence of any promising leads, such work is in abeyance at CRNL. However, if the CANLUB layer proves to be less-than-satisfactory at higher burnup, there remains a satisfactory, but higher cost alternative developed by US General Electric in the form of thin layer of pure zirconium on the inner surface of the Zircaloy sheath.

As the proportion of the Ontario Hydro grid supplied by nuclear generation grows, the need for load following rather than base load stations increases; this is because nuclear station power may have to be decreased overnight or at weekends as the demand drops. To date there has been little need for fuel to perform in this 'load following' or power cycling regime, but testing is required, to confirm the expectations (based on a few in-reactor experiments) of satisfactory performance. The probability of defecting fuel during power cycling appears to be very low, so any test to evaluate this would require a very large number of bundles. However, a necessary transitional stage between the present single element testing and the power cycling of a complete reactor core requires testing of a few bundles, in part to examine whether differential thermal expansion might damage end plates or welds. This is due to be performed in the NRU reactor, using a rig that moves fuel bundles at power between regions of high and low neutron flux.

3.4 Fission Product Release

Lewis et al. (15) provided many of the initial concepts that present theories encompass. Fission product atoms move (diffuse) through UO_2 grains at a rate dependent on temperature, until they reach a grain boundary.

There they precipitate, with the gaseous elements forming bubbles. When there are sufficient bubbles, these interlink to form tunnels along grain edges (16), and gas atoms can move towards the fuel-to-sheath gap (Figure 7). Gas pressure within a CANDU fuel element is of concern because little space is allowed within the sealed element to contain the gas; a high gas pressure might cause the sheath to 'lift' off the fuel and decrease fuel to sheath heat transfer. Other fission products, particularly Iodine, Cesium and Cadmium, are suspected as being responsible for 'stress corrosion cracking' failures when the fuel power is increased (see Section 3.3).

If the sheath of a fuel element were to rupture, radioactive fission products would be released to the coolant. There is therefore increased emphasis on understanding the factors controlling the release of radioactive species from defected fuel under normal operating conditions (Section 3.5), or under accident conditions (see Section 4.2).

3.5 Performance of Defected Fuel

A certain incidence of leaking (defected) fuel is always to be expected during normal operation, due to manufacturing faults, debris in the coolant, or power-ramp failures. Currently in CANDU reactors this is less than 0.2% of all bundles irradiated, and, since usually only one or two elements of the 28 or 37 in a bundle fails, is less than about 0.01% of all elements.

The questions that have to be answered include

(i) How long can defected fuel operate in-reactor without unduly contaminating the primary coolant circuit? and

(ii) How should defected fuel be handled during transfer from the fuelling machine, after it has been removed from the reactor, to the sotrage bays, and how should it be stored?

There have been many irradiations of defected fuel, both in power reactors and in test loops. However the latter tests used pre-fabricated defect holes 0.5-1.0 mm in diameter, and the hydriding damage* observed in the power reactors was absent. A recent test of very small defects produced in manufacture indicated that restricted access of coolant to the pellet-to-sheath gap was required if hydride blisters (Figure 8) were to form. Further tests of naturally-occurring defects are planned to study sheath deterioration rates. A possible way to identify badly defected bundles (and hence be able to remove them from the reactor) is to measure and compare the

* Water (H_2O or D_2O) reacts with the UO_2 and the Zircaloy sheath to form brittle blisters of zirconium hydride or deuteride (ZrH_x or ZrD_x, where x is between 1.5 and 2).

activity of various released fission products. This seems to be a function of, among other things, defect hole size. Therefore the ultimate target of this program is to correlate damage, operating conditions and activity release. It is hoped to complete this in 1986.

The release of fission products from defected fuel into air during transfer from the reactor to the water bays may provide a feasible way of detecting fuel that has defected during operation. This fuel may then be isolated in containers to avoid excessive contamination of the bay area. Preliminary tests at CRNL have shown that significant quantities of volatile radioactive species can be evolved from a defected element if the bundle heats up in air, and it remains to try to devise apparatus for trial use at a power reactor.

3.6 Fuel Modelling

Fuel modelling is an ongoing activity, since the objective is to have the codes represent our best understanding of the physical processes controlling fuel behaviour. As such, not only are code revisions required from time to time, but a library of closely defined experimental test data must be built up and enlarged, against which code predictions are checked and areas of disagreement investigated.

The fuel modelling code ELESIM (17) has been in general use for several years, for evaluation of performance during normal operation. An improved code, ELESTRES (18), has been written. This uses many of the ELESIM subroutines, but the semi-empirical pellet deformation model in the latter has been replaced by a more detailed stress analysis, which permits calculation of sheath strain (and hence of the conditions which might cause sheath failure), at the ends of the pellets (where most strain is observed to occur) as well as the pellet midplane location. Concurrent with this has been the re-organization of the fuel codes into a series of subroutines stored in a current 'library'. This facilitates revision of any particular topic as required, and reduces duplication.

4. PREDICTION OF FUEL BEHAVIOUR UNDER ABNORMAL (ACCIDENT) CONDITIONS

There is a small possibility that, occasionally, something may go wrong with one of the process or control systems of a nuclear reactor. If the incident subjects the fuel to abnormal conditions, then we must evaluate whether the fuel is damaged in any significant way. Since the number of hypothetical accident scenarios is limited only by the imagination, we attempt to model fuel behaviour and use these computer codes to give us a prediction of the most likely result of any particular sequence of abnormal operating conditions.

Accidents of concern generate higher-than-normal fuel element temperatures, usually by impairment of the cooling of the element. Such undercooling can arise as a result of

i) loss of power regulation, which leads to element overpower,

ii) by reduction of coolant flow (local channel blockage), or

iii) by rupture of the primary heat transport circuit (loss of coolant accident, known as LOCA); in this case the reactor shuts down immediately, but the energy stored in the fuel, the heat due to radioactive decay and the heat from zirconium oxidation all can cause the fuel temperature to rise.

During the temperature excursion the gas pressure within the fuel element may exceed the restraining coolant pressure, and the fuel element sheath may balloon or rupture. This could lead to release of fission products, or to local restrictions of coolant flow within the channel.

Thus, we require to be able to predict at least three events:

i) When does fuel defect, and what is the consequent release of radioactivity?

ii) Will sufficient flow blockage occur that channel cooling is seriously impaired?

iii) Is the pressure tube put at risk (for example by local overheating)?

It is convenient to consider abnormal events in three classifications.

i) Loss of coolant accidents (LOCA) in which fuel overheating is relatively moderate. In these, the sheath remains an important structural member, and its deformation dictates whether or not it defects and releases fission products. These accidents are classified as 'Low Temperature Loss of Coolant Accidents'.

ii) Loss of coolant accidents during which the element gets sufficiently hot to cause sheath oxidation, embrittlement and rupture. In general, the performance of fuel under these conditions has to be analyzed in terms of chemical reaction rates rather than response to stress, since the sheath ceases to have significant mechanical strength or integrity. Here, significant dispersion of fuel and fission products is to be expected. These accidents are classified as 'High Temperature Loss of Coolant Accidents'.

iii) Undercooling events, where steam formed on the sheath surface with the reactor operating at power may impair cooling (cause "dry-out") and the overheated fuel elements may then bow and contact the pressure tube.

All these accidents involve a period of rapidly rising sheath temperature followed by a declining temperature history as the safety systems (e.g. reactor shutdown, emergency coolant injection) operate. The consequences are critically dependent on the temperature history during the event, and must be evaluated numerically, since one cannot otherwise extrapolate reliably from any particular experimental determination. Therefore we need to describe each of the relevant phenomena in mathematical terms, link them together in predictive codes, and verify predictions by comparing with well controlled experimental tests. The individual tasks are described in the following sections.

4.1 Fuel Bundle Code Development

Traditionally, code development has proceeded in two areas. The whole reactor coolant circuit has been described in channel thermalhydraulic codes, which include the nuclear fuel in a simplified manner. Fuel codes describe the performance of single fuel elements, and rely on the separate thermalhydraulics code to give input conditions. However, fuel element temperature can be affected greatly by local thermalhydraulic conditions, which in turn are affected by fuel deformation, etc. For this reason we would like to consider what happens to each fuel element in a single fuel channel, and define the boundary thermalhydraulic conditions as being those at the inlet and outlet headers. Coolant within the channel can then be allowed to vary as a consequence of fuel element distortion or dryout and this will not alter the header conditions.

A code, CANSIM, is under development at CRNL (19) to model the performance of a fuel channel containing CANDU fuel bundles in the above manner. We intend to use this system to do sensitivity studies, to guide us as to the importance of each interacting variable. The code is written in a system of subroutines, with a control code to pass information between each 'block' of calculations (each representing a particular physical process). In this way, three objectives can be met:

i) Code running time and storage requirements are minimized.

ii) Every subroutine (physical process) can be worked on independently, yet can be made compatible with the overall code.

iii) Sensitivity studies and assessments guide the relative importance of each individual physical process, and provide a tool for determining how much more work is needed in each area.

The bundle/channel model will be operating in 1983. From then on, the code will be improved by either allowing for new (and significant) effects, or by improving algorithms (and our knowledge of physical processes).

Adequate representation of the single fuel elements within the fuel bundle presents problems. It is known (Section 4.4) that temperature gradients or 'hot spots' in the sheath will significantly reduce sheath strain and increase the probability of rupture. However it is difficult to write a code that allows for sheath temperature variations and yet is economical in its running time. The fuel element code presently available, ELOCA (20), was developed at CRNL some years ago, and has been used extensively in safety analysis. It is one dimensional axisymetric (i.e., it permits variations in properties, temperature, etc. across the fuel radius only, not along the length or around the circumference). It is possible that ELOCA can be used to describe the behaviour of the fuel pellet, and the sheath be described as a shell with a circumferentially-varying temperature distribution.

A finite element code, FAXMOD (21), has been developed at WNRE to permit two-dimensional evaluation of a complete fuel element. This allowed for the representation of either an axisymmetric fuel pin, with effects varying axially, or an axially uniform pin, with two dimensional variations across a 'slice' normal to the axis. The code structure is complex and it therefore occupies a great deal of computer memory. Additionally, many constituent physical effects, such as pellet cracking, fuel densification and sheath oxidation are not included. Although too large to incorporate easily into a channel code, sensitivity studies using FAXMOD enable us to evaluate the importance of various effects, and thus the need to include them even in an approximate form.

In subsequent sections in this report, work done to evaluate discrete physical processes, together with their numerical simulation, will be described. As previously mentioned, many subroutines are common to both normal and accident codes, and all are stored in a library and called as required. Those particular to accident evaluation codes are discussed here; others are discussed under work on fuel behaviour under normal operating conditions (Section 3).

4.2 Release of Radioactive Fission Products

As discussed in Section 3.4, fission events break fissile atoms into lighter atoms called fission products; these are usually radioactive. Some of the fission product atoms remain fixed in the UO_2 lattice in which they were born, but others migrate and may eventually be released. When the sheath ruptures, the fission product inventory within the fuel-to-sheath gap (the 'gap inventory') will be released to the channel. If the fuel heats up, if it shatters, or if it reacts with oxygen or the Zircaloy components, then more fission products may be released. Some of the fission products (the noble gases, xenon and krypton) are inert, but others such as iodine, cesium and tellurium are chemically reactive and can adhere to surfaces, remain in solution in water, or form aerosols or gaseous components, depending on the system chemistry. Therefore our objectives are

(a) to predict the amount of each isotope released from a ruptured fuel element, and

(b) to determine the chemical form of the released species.

Research on the subsequent behaviour of the fission products within the fuel channel and the reactor containment is part of the System Chemistry R&D program (22) at CRNL and the Thermalhydraulics R&D program (26) at WNRE.

To date, experiments at CRNL have concentrated on the releases from fuel elements under normal operating conditions, to assess the 'gap inventory'. A 'sweep gas' technique (Figure 9) has been developed in which a stream of inert gas is passed through an operating fuel element to act as a carrier for radioactive species. It is then passed past a gamma spectrometer to analyze the amounts of the radioactive isotopes. The results (23) have shown that less than 1% of the gap inventory of iodine (the most radiologically significant of all the fission products) is present in a volatile form (e.g., CH_3I). When liquid water (not steam) was present, as occurred when a test defected, the iodines were transported from the gap very much more readily.

During a temperature transient other phenomena can augment release. Fuel microcracking can release a significant fraction of the fission products retained within the fuel pellet. The growth of fission product-filled bubbles can form interlinked tunnels which increase release, and the swelling of the fuel by formation of the bubbles can affect fuel performance by, for example, increasing the contact between the fuel and the sheath. The 'sweep gas' experiments have been extended to transient situations (24), where fuel element temperature is altered by controlling the coolant flow past the element. This results in transient increases in release of radioactive species when temperature is increased and, to a much greater extent, when the fuel cools rapidly following reactor shutdown. This data is being used to evaluate models which are based on those which had been successfully used to predict the release of stable fission product gases in UO_2 fuel elements.

When the sheath defects, the UO_2 fuel will oxidize. The amount and rate of this oxidation will depend on many factors, for example, coolant chemistry, temperature and defect size. Although there have been studies of UO_2 oxidation, there is little relevant information on the effect of oxidation on fission product release. A program to study this area is currently being formulated. A minicell is being built, in which measurements of the weight gain (oxidation) of UO_2 samples, and the concurrent release of fission products, will be measured. Chemical species will be identified by mass spectrometer. In addition, experiments using the new in-reactor blowdown test facility (Section 4.6) will extend this work to an in-reactor environment, and, probably, to higher temperatures than can readily be achieved in the laboratory.

4.3 Sheath Mechanical Properties

If the gas pressure within a fuel element is greater than the coolant pressure, the sheath is stressed in tension. This may cause sheath strain (ballooning) especially at the elevated temperatures anticipated in some LOC accidents. In order to determine fuel element performance, we must be able to calculate such deformation and the possibility of rupture. A program to determine the mechanical properties (strength) of Zircaloy fuel sheathing has therefore been under way for about ten years, mainly at CRNL (25), but with significant effort from WECAN and CGE (both funded by AECL), and at WNRE.

The attachment of appendages by brazing is one of the more significant differences between CANDU and PWR fuel. It results in three zones of different crystalline structures and mechanical properties in the sheath. Over 700 mechanical tests have provided data from which a microstructurally-based deformation model has been derived, which takes into account the metallurgical structure of the as-fabricated tube, stress, temperature and microstructural changes occurring during the test.

At high temperatures, the sheath will oxidize in an air or steam environment. A ZrO_2 layer is formed on the oxidizing surface, and a layer rich in dissolved oxygen forms between the ZrO_2 and the unoxidized Zircaloy substrate. Both the oxidized layers are stronger than the unoxidized Zircaloy, so ballooning of a fuel element will be reduced by oxidation. However, the ZrO_2 is brittle and cracks; when it does so, strain is concentrated in the sheath beneath the crack and may cause localized overstrain failure (rupture) - Figure 10.

To model these effects, the rate of formation of the layers has to be predicted, and the strength of the layers has to be measured as a function of temperature and oxygen content. We are currently developing a code which calculates the amount of oxidation and estimates strengths and tube rupture using the three-layer model outlined above. Work on the strengthening effects of high oxygen contents is under way at McGill University, to ensure that extrapolation from presently available low-oxygen data is satisfactory. Work at CRNI is concerned with the strain of ballooning tubes, their rupture, and verification of the predictive code.

4.4 Sheath Failure Criteria

During a LOCA transient, the sheath may fail for a number of causes;

(a) Localized overstraining.

(b) Cracking due to penetration of the Beryllium braze material used to attach the sheath appendages.

(c) Oxidation induced strain failures.

(d) Oxygen embrittlement.

The predictive codes (e.g. ELOCA) have to have criteria in order to predict when, during the simulated accident, conditions are such that sheath rupture is probable. It has been customary to use a conservative criterion that when the element balloons by more than 5%, rupture is assumed, but a more realistic approach is desirable. It has already been stated that we can expect some temperature non-uniformity around the sheath of the fuel element. Since Zircaloy is, in general, weaker at higher temperatures, rupture can be expected preferentially at a 'hot spot'. Adoption of criteria based on the local stress or strain in the sheath exceeding some critical value appear satisfactory. The effect of sheath oxidation has been included in the code, (Section 4.3) and, since the oxide layer cracks and localizes strain, the limiting local strain criterion can also be used for oxidized sheathing. Similarly, since the amount and distribution of oxygen is calculated, a criterion for the amount required to embrittle the sheath has been devised.

4.5 High-Temperature Fuel Performance

Our program was originally based on the concept that no single failure in a CANDU reactor could lead to significant fuel failures. In general terms, conditions were such that sheath temperatures would not exceed about 1200°C and would be cooled to lower temperatures within 100 seconds. However, it has become necessary to consider conditions where higher temperatures can be hypothesized. Accordingly, effort is being redirected towards the less likely high temperature accidents involving the loss of emergency cooling or where poor cooling occurs over a long period of time during loss of cooling accidents.

A fuel channel undergoing a high temperature accident might expand into contact with its moderator-cooled calandria tube, thus maintaining the integrity of the pressure tube/calandria tube assembly (26). The fuel bundle temperature could be somewhere between 1200 and 2400°C, depending on the decay heat and the surface area exposed to the exothermic Zircaloy steam reaction. At these temperatures it is expected that the fuel bundles will slump and settle on the bottom of the fuel channel. The deformation of the settled bundle is of importance since it will affect the heat transfer to the pressure tube and hence the moderator, reduce the surface area for the exothermic Zircaloy-steam reaction and thus reduce the temperature to which the pressure tube is subjected. This process of fuel settling to the bottom of the pressure tube is complex, involving the interaction of the Zircaloy fuel sheaths, the UO_2 fuel pellets, the various bearing and spacer pads, the welds between the end plate and end caps, the end plate and the steam supply. At these very high temperatures, rates of chemical reactions are more likely to be important than rates of mechanical deformation. The research and the resulting computer models are therefore likely to emphasize different aspects than those in support of the lower-temperature accidents discussed previously.

We have initiated a program of work at WNRE to look at the fuel behaviour during these high temperature accidents. The principal objectives of this Transient Fuel Element/Bundle Settling Program are:

(a) To investigate the relevant physical and chemical phenomena which may lead to the fuel bundle settling to the bottom of the fuel channel.

(b) To develop models for determining the rate of fuel bundle settling.

(c) To provide a basis for a decision on whether settling tests are necessary with irradiated fuel bundles.

The main phenomena of interest are presently defined as:

- The high temperature oxidation and embrittlement of Zircaloy fuel cladding in steam at temperatures greater than 1200°C, including the possible effects of steam starvation.

- The resultant hydrogen production.

- The mechanical and chemical interaction between UO_2 pellets and Zircaloy cladding.

- The coolability of the settled bundles, i.e. the change in transfer of heat from the settled fuel bundle to the pressure tube.

- The effect of bearing and spacer pads, and end plate restraints on the geometry of the settled bundles.

- The fission product release from settled bundle (in-cell).

This program got under way in 1982, and is likely to gain more emphasis as work in the lower-temperature regions is completed and phased out.

4.6 "All-Effects" Experiments

Up to this point, we have been discussing work to develop computer codes that predict fuel performance under abnormal operating conditions. However, this approach of researching the constituent parts of a multi-variable problem and then putting them together, eventually has to be checked against 'all effects' experiments. These 'all effects' experiments are being done both in- and out-reactor. Although it would be desirable to do all the confirmatory tests in reactor, there are practical difficulties. Existing facilities in the research reactors are vertically oriented, and thus do not permit investigation of the gravitational effects that can occur in horizontal channels. Furthermore, experimental difficulty is vastly increased by testing in-reactor, due to limitations to experimental measurement techniques imposed by radiation fields. Experiments in power reactors are impractical. The compromise we therefore adopted is to study those effects where gravity is likely to be important, namely fuel element sag and coolant stratification, in horizontal out-reactor experiments using electrically-heated simulated fuel elements, and to use vertical in-reactor tests to research fuel behaviour and fuel-sheath mechanical interactions.

AECL and Ontario Hydro are funding out-reactor testing at WECAN to study the behaviour of seven-element clusters of electrically-heated elements under high temperature conditions. The objectives are to assess whether distortion is likely to be severe enough to affect heat removal from the fuel channel, and to investigate how single element behaviour is modified by its interaction with other elements and the pressure tube. Special electrically heated elements have been developed which contain UO_2 annular pellets to approximate the correct weight and thermal response of a fuel element.

In-reactor tests of instrumented fuel elements are presently being done in the X-2 loop of NRX. This is a loop that was originally designed to run tests of defected elements during normal operating conditions, but it was modified so that the coolant in the test section could be allowed to flow (blow-down) into a tank. However the facility has not proved to be sufficiently versatile. There is a lot of coolant in the pipes leading to the test section; when the test section is depressurized (blow-down into the tank) there is excessive cooling and it is difficult to achieve fuel sheath temperatures above about 1000°C. One further shortcoming of the loop is that we must ensure that the fuel element does not defect during blowdown. For these reasons we have decided to build a steam-cooled accident loop.

The tests that have just been completed used stainless steel sheathed elements. For this series, an advanced data collecting system with an information display module - a REDNET satellite has been installed (27). This system and the fuel instrumentation have worked well through 24 blowdowns. However, even with a ten-second delay between initiation of blowdown and reactor trip the maximum sheath temperature observed was only 1065°C, and furthermore it was localized and lasted for only a few seconds. The X-2 loop will continue to be used even though it has limitations, until the new steam cooled loop becomes available in 1986. In the meantime, essential work on evaluating and developing instrumentation is in progress, in collaboration with the researchers of the Reactor Instrumentation, Control and Electronics program (27). Existing sensors for measuring loop coolant pressure, element internal gas pressure, a dryout detector and a non-contacting strain sensor, are all being improved and evaluated.

A further series of in-reactor tests is under way, making use of the Power Burst Facility (PBF) at Idaho Falls (USA). In a collaborative program sponsored by Ontario Hydro, we have built and irradiated instrumented CANDU fuel elements which will be installed in PBF and subjected to a blowdown transient such as could be anticipated in a power reactor, if a header were to rupture. These tests, of four elements with differing configurations, will provide information against which the ELOCA code can be compared. By this means, code verification for low-temperature accidents will be attained earlier than by waiting for completion of a new in-reactor accident test loop at CRNL.

NRX reactor is due to be closed down in the late 1980's. In order to ensure that Canada has a facility available in which to continue to test fuel performance under accident conditions (primarily high-temperature severe fuel damage accidents) we have to construct a new test loop. The blowdown test facility (BTF) is being designed and constructed in NRU reactor, and should be commissioned by 1986. The facility is designed so that fuel elements (initially a trefoil, but possibly a septafoil) could be operated at power in steam or pressurized water coolant, then the test section would be valved off, the test section depressurized to simulate a loss-of-coolant accident, and the reactor be tripped (shut down). If the fuel defects, fission products would travel down the blowdown line to a holding tank. Grab samples will be taken for subsequent analysis, gamma-ray spectrometers will measure the release and movement of fission products down the pipework.

We envisage that the facility will permit studies of high temperature accidents to:

(a) provide information on activity release from failed fuel, deposition of fission products in the test section, and possibly the chemical form of species released from the channel,

(b) provide information on the progression of failure, such as fuel disintegration, reaction with structural components, etc.,

(c) verify any model of fuel behaviour (low or high temperature accidents).

When it is completed, this loop will be the primary facility for in-reactor fuel testing under accident conditions, and much of the fuel program remaining to be completed after 1985 will make use of it.

5. DISCUSSION

Some thirty years of development of CANDU reactor fuel has resulted in a highly reliable, inexpensive source of power, and has been instrumental in setting up a multi-mission dollar fuel industry in Canada. It is inconceivable that fuel development should terminate; to do so would invite competitors to drive the indigenous industry out of business. However, a greater share of the costs of fuel development is, and should be, being borne by the users of the fuel rather than by the taxpayer through the Government of Canada.

As work to perfect the fuel design for normal operating conditions has been completed, it has shifted to the task of predicting fuel performance under abnormal operating conditions, for example, in accidents. In the longer term, work on Advanced fuel cycles (13) will build on, and extend what has been learned for UO_2 fuel in the current CANDU reactors. Since the fuels have similar properties and are required to operate under similar conditions, staff and facilities that have been used for development of UO_2 fuel will spend progressively more time on ThO_2-based fuel. However they will be available to respond to unforeseen problems in the present CANDU reactors, as discussed below.

To this point we have concentrated on the technical work being done on power reactor fuel performance. However, there are other responsibilities that must be met by the dozen scientists and engineers working in this field.

Operational problems cannot be foreseen; an urgent need for a crash program occurs from time to time, for example to investigate why an event occurred or to provide a solution to a problem. The utilities and the AECL fuel designers do not have the full range of expertise nor the facilities of CRNL or WNRE; the research sites can thus provide invaluable assistance. An example of this was the high incidence of fuel defects that led to the development of CANLUB coatings (Section 3.3). In order to be able to respond and cope with this type of unforeseen problem, research facilities and knowledgeable staff must be available. This implies that a minimum level of relevant work, preferrably research or development, must continue. The problem that faces the fuel community as a whole is to decide what skills and level of support is required.

A further obligation is to assist in the marketing of the CANDU reactor system. Although not directly involved, the fuel experts have an important role to play, namely to provide technology transfer to CANDU customers. Often, a significant incentive to buy CANDU lies in the very simple fuel cycle. The CANDU fuel design reduces the complexity of fabrication or fuel handling facilities. It entails less investment and industrial infrastructure to make CANDU rather than PWR fuel, and use of natural uranium offers the chance of indigenous fuel self-sufficiency. A potential customer will wish to learn about the background to CANDU fuel development so that he can resolve problems for himself. He may also want to manufacture fuel, so that he does not have to depend on foreign suppliers. To implement this technology transfer requires a core of knowledgeable specialists, currently working in the field. If customers' staff are to be trained, AECL's fuel-related facilities must be kept operational, doing relevant work. Again there is a requirement for some minimum level of on-going work, funded where possible by contracts, but recognized as a necessary cost of sales and marketing.

6. SUMMARY

The report outlines the research and development work that has gone in to making the natural-uranium CANDU fuel the simple and effective power source that it is today. Effort has been re-directed towards investigation of the consequences of abnormal conditions (i.e. accidents), and in the long run will transfer to research into fuel for Advanced Fuel Cycles, based on thorium.

The role of the AECL research sites in supporting CANDU sales and technology transfer is discussed, as is the need for maintaining a core of expertise and facilities in order to respond to unforeseen circumstances.

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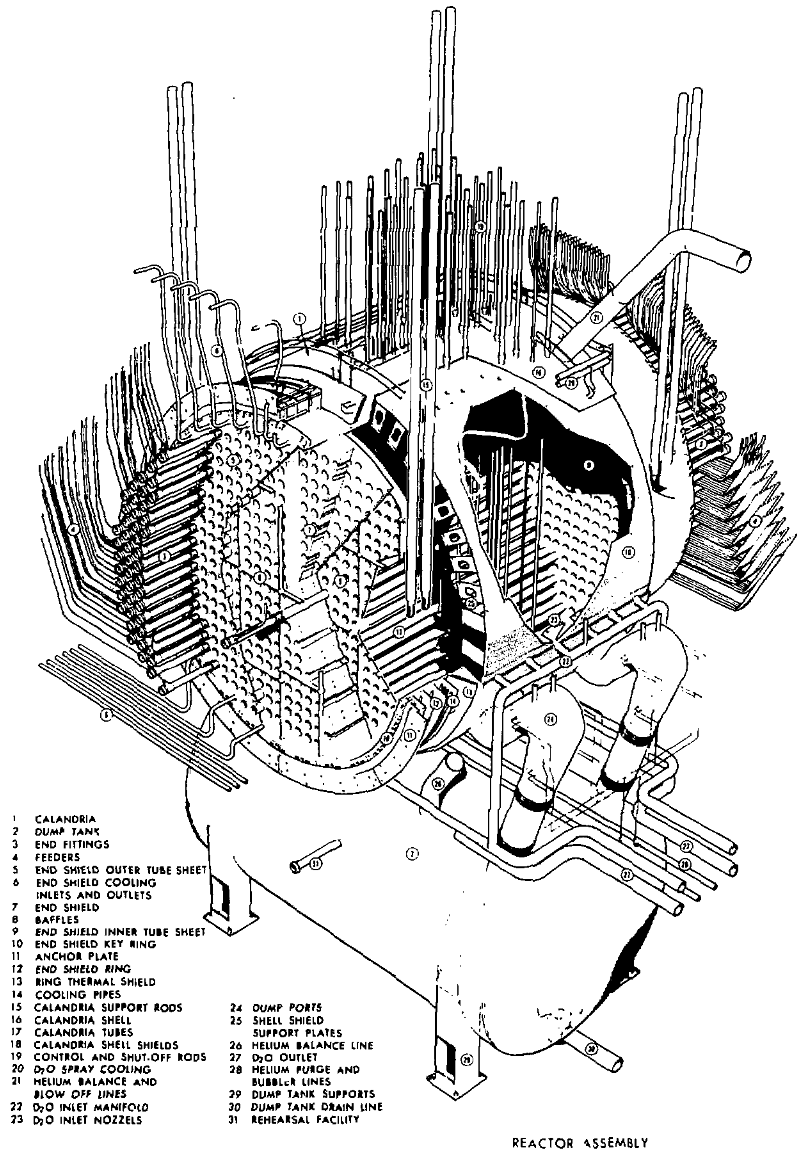

Figure 1 CANDU Reactor, showing the array of horizontal pressure tubes passing through the calandria.

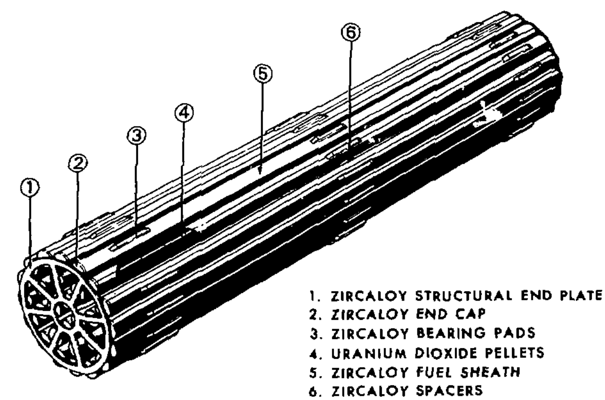

Figure 2 CANDU 37-element fuel bundle.

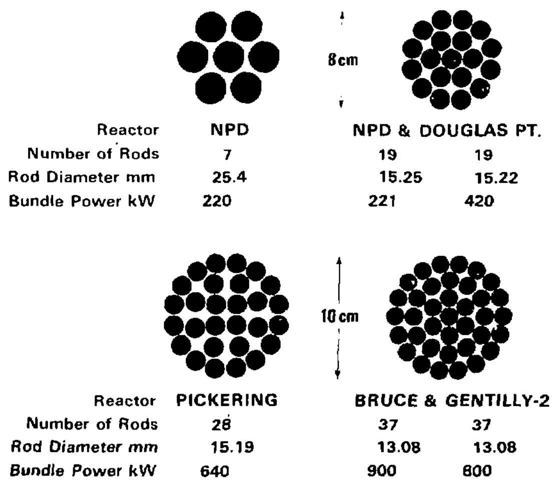

Figure 3 Cross sections of CANDU power reactor fuel bundles.

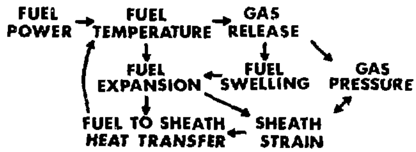

Figure 4 The interactions of the major variables that control fuel element performance.

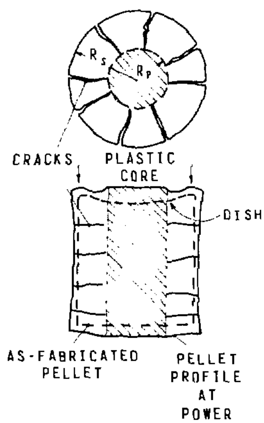

Figure 5 Schematic diagram showir_ how a fuel pellet expands.

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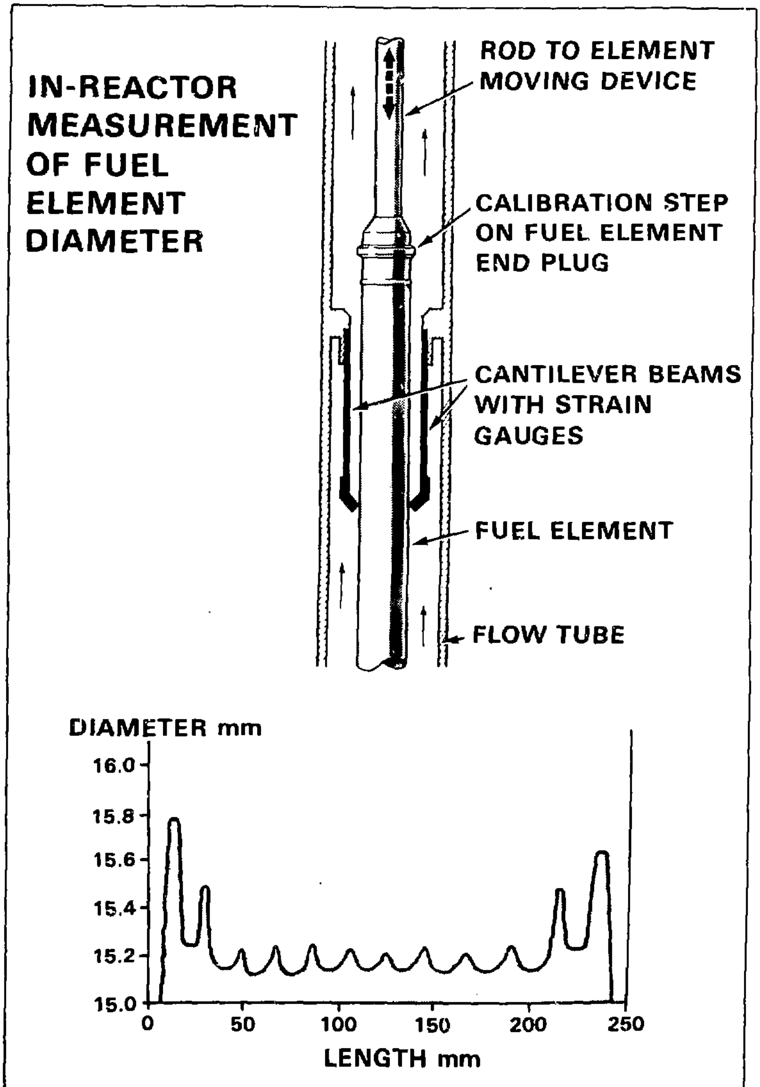

Figure 6 Schematic illustration showing the IRDMR measuring a fuel element diameter in a pressurized water loop (10).

(X1000)

Figure 7 Fractured surface through irradiated UO_2, showing bubbles precipitating at grain boundaries, and tunnels forming at grain edges.

(X75)

Figure 8 'Blister' of zirconium hydride formed by decomposition of the moisture that entered a fuel element through a defect.

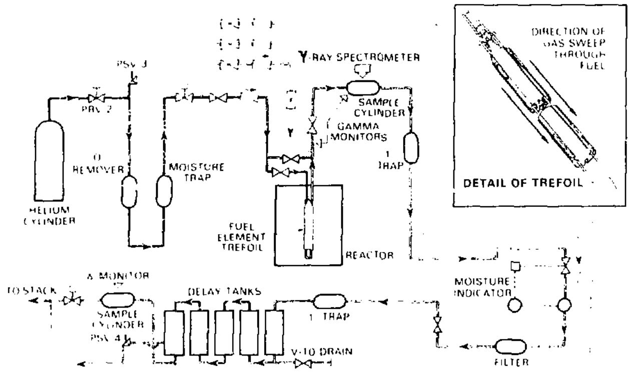

Figure 9 Schematic of the sweep gas system for measuring the on-line release of gaseous radioactive fission products from UO_2 fuel.

(X500)

Figure 10 Localized sheath strain beneath a crack in the zirconium oxide layer.

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