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CANDU SAFETY UNDER SEVERE ACCIDENTS

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Abstract

The characteristics of the CANDU reactor relevant to severe accidents are set first by the inherent properties of the design, and second by the Canadian safety/licensing approach.

The pressure–tube concept allows the separate, low–pressure, heavy–water moderator to act as a backup heat sink even if there is no water in the fuel channels. Should this also fail, the calandria shell itself can contain the debris, with heat being transferred to the water–filled shield tank around the core. Should the severe core damage sequence progress further, the shield tank and the concrete reactor vault significantly delay the challenge to containment. Furthermore, should core melt lead to containment overpressure, the containment behaviour is such that leaks through the concrete containment wall reduce the possibility of catastrophic structural failure.

The Canadian licensing philosophy requires that each accident, **together** with failure of each safety system in turn, be assessed (and specified dose limits met) as part of the design and licensing basis. In response, designers have provided CANDUs with two independent dedicated shutdown systems, and the likelihood of Anticipated Transients Without Scram is negligible.

Probabilistic safety assessment studies have been performed on operating CANDU plants, and on the 4 x 880 MW(e) Darlington station now under construction; furthermore a scoping risk assessment has been done for a CANDU 600 plant. They indicate that the summed severe core damage frequency is of the order of 5 x 10^{-6}/year.

CANDU nuclear plant designers and owner/operators share information and operational experience nationally and internationally through the CANDU Owners' Group

(COG). The research program generally emphasizes the unique aspects of the CANDU concept, such as heat removal through the moderator, but it has also contributed significantly to areas generic to most power reactors such as hydrogen combustion, containment failure modes, fission product chemistry, and high temperature fuel behaviour.

Abnormal plant operating procedures are aimed at first using event–specific emergency operating procedures, in cases where the event can be diagnosed. If this is not possible, generic procedures are followed to control Critical Safety Parameters and manage the accident. Similarly, the on–site contingency plans include a generic plan covering overall plant response strategy, and a specific plan covering each category of contingency.

1. NATIONAL CONTEXT

1.1 STRUCTURE OF THE CANADIAN NUCLEAR INDUSTRY

The CANDU nuclear power concept was realized in 1962 with the first operation of the Nuclear Power Demonstration (NPD) prototype, a co–operative venture between the designer, Atomic Energy of Canada Limited (a federal government Crown Corporation), and the operator, Ontario Hydro (a provincial government utility). Now there are twenty seven operating CANDU reactors worldwide, with eleven more under construction in Canada, Romania, and India. Ontario Hydro owns and operates 18 Canadian reactors (with two under construction), and now performs both plant design and safety assessment. Two more Canadian provincial utilities own and operate CANDUs – Hydro Québec and the New Brunswick Electric Power Commission. Atomic Energy of Canada Limited (AECL) has evolved into a number of related companies, including a design company, AECL-CANDU, and a Research Company (AECL–Research) which performs both fundamental and applied research in support of the CANDU concept.

All of these organizations, plus owners of CANDU plants in Korea and Argentina, coordinate their needs and exchange information through the CANDU Owners' Group, or COG. The safety analysis to support the CANDU is performed both at the owner/utility and at AECL–CANDU.

Nuclear power is regulated in Canada by a federal government agency, the Atomic Energy Control Board (AECB). Each provincial government has the responsibility for offsite emergency planning within its borders.

1.2 DEFINITION OF A SEVERE ACCIDENT

A <u>severe accident</u> is defined as one in which the fuel heat is **not** removed by the coolant in the primary heat transport system (PHTS). In most other reactor designs, this is equivalent to a core melt, and indeed <u>severe core damage</u> (defined as loss of core structural integrity) is one end of the spectrum in CANDU. However the inherent characteristics of CANDU provide a broad spectrum of scenarios where, even if primary and emergency cooling are lost, the fuel does not melt.

1.3 REPORT OUTLINE

The characteristics of the CANDU reactor relevant to severe accidents are set first, by the inherent properties of the design, and second, by the Canadian safety/licensing approach. For a basic introduction to the safety approach, see [Snell, 1985].

In Section 2, the evolution of the licensing approach is described, and in Section 3, the design aspects related to severe accidents are summarized. The characteristics of severe accidents themselves, in terms of frequencies and consequences, are summarized in Section 4.

The Canadian research programme which supports the conclusions reached in severe accident analysis is presented in Section 5. The operating philosophy and procedures relevant to arresting and mitigating severe accidents are described in Section 6.

Hereafter the features of a "typical" CANDU reactor are described. Most of the conclusions reached are generic.

This report is a more detailed version of an invited paper presented at "The International Symposium on Severe Accidents in Nuclear Power Plants", in Sorrento, Italy, March 21–25, 1988 [Snell, 1988].

2. CANADIAN SAFETY AND LICENSING APPROACH

2.1 EARLY PROBABILISTIC DEVELOPMENTS

The safety/licensing approach in Canada has had a probabilistic component going as far back as the 1950s (see [Snell, 1986] for a review). The accident which damaged the NRX research reactor at Chalk River, Ontario, in 1952, was caused by a failure of a normal process system (a process system is defined as any system required for the normal operation of the plant) and a partial failure of the protective system (shutdown) designed to protect against the fault ([Lewis, 1953], [Hurst, 1953]). This spurred an early interest in the frequency of accidents, and in the reliability and independence of the protective systems. These ideas led [Siddall, 1959] to definition of frequency targets for failures of the process systems, and reliability targets for the protective systems, which had to be demonstrated in practice.

A severe accident could only result if a process system failed **and** the appropriate protective system was simultaneously unavailable. With the systems sufficiently independent, the frequency of a severe accident could be made acceptably low. This philosophy of **separation** of process and safety systems, and **verifiable reliability** targets for each, became the hallmark of Canadian safety and licensing [Laurence, 1961].

These ideas were expanded with Canada's first power reactors – the 25 MW(e) Nuclear Power Demonstration reactor and the 208 MW(e) Douglas Point prototype. A safety goal was defined for risk of death to any member of the public. This goal was used to establish the reliability requirements for the process systems, the shutdown systems, the emergency coolant injection system, and containment (the latter three were later called the special safety systems). The possible combinations of process failures with or without safety system action formed an accident **matrix** which included a number of severe accidents; these were analyzed with the tools of the day as part of the licensing submission.

2.2 SINGLE/DUAL FAILURE MATRIX

With the construction of the four–unit Pickering–A nuclear generating station near Toronto, the AECB licensing guidelines, while retaining a probabilistic component, became more deterministic. All currently operating plants up to Darlington have been since licensed under these rules [Hurst, 1972]. The spectrum of possible accidents is divided into two broad categories – **single failures**, or the failure of a process system which requires the intervention of one or more of the special safety systems; and **dual failures**, a much less likely event defined as a single failure coupled with the assumed unavailability of one of the special safety systems. (The single failure is an assumed system failure and is not related to the same term used elsewhere to describe a random component failure additional to the initiating event.) For each class, designers had to demonstrate that specified frequency and consequence targets were met (Table 2.1). For example, the spectrum of dual failures included loss–of–coolant plus failure of the emergency coolant injection system, and loss–of–coolant plus failure of containment ventilation to isolate. For each of these, the maximum individual whole–body dose to the critical individual at the site boundary had to be demonstrated to be less than 0.25 Sv, and the maximum collective dose in the surrounding population had to be shown to be less than 10^4 person–Sv, under pessimistic atmospheric dispersion conditions. There were additional limitations on thyroid dose, as shown in Table 2.1. The frequency targets are not **expected** frequencies – they were chosen large enough that compliance could be demonstrated (from direct observation of single failure frequency, and from safety system reliability in periodic testing) in a few years of actual station operation.

TABLE 2.1 - DOSE/FREQUENCY GUIDELINES

ACCIDENT	MAXIMUM FREQUENCY	INDIVIDUAL DOSE LIMIT	POPULATION DOSE LIMIT
Single	1 per 3 years	0.005 Sv wb	10^2 person–Sv
Failure		0.03 Sv thy	10^2 thy–Sv
Dual	1 per 3000 years	0.25 Sv wb	10^4 person–Sv
Failure		2.5 Sv thy	10^4 thy–Sv

Note: wb=whole body; thy=thyroid.

The special safety systems are designed and operated to:

1) demonstrate during operation, by test, a dormant unavailability no greater than 10^{-3}, or about eight hours per year;
2) be physically and functionally separated from the normal process systems and from one another. For example each shutdown system has its own detectors, amplifiers, logic relays, and actuating mechanisms, which cannot be shared with either the other shutdown system, the other special safety systems, or the reactor control system; which are of generally diverse design and manufacture; and which where practical are located in physically separated areas of the plant. During the design a stringent peer review, design review and QA programme is applied to ensure, among other things, that separation standards are met.

2.3 THE C–6 FIVE–CLASS APPROACH

The latest CANDU in Canada, the 4 x 880 MW(e) Darlington Nuclear Generating Station, was licensed using the approach described in AECB document C–6 [AECB, 1980]. C–6 retains the deterministic elements of its predecessor, the single/dual failure matrix, but recognizes that describing the spectrum of accidents by only two classes is too limiting: the number has been expanded to five, with consequence targets set for each class, and accidents assigned to classes implicitly according to frequency.

C–6 represents another step in the continuous tradition, in the licensing approach in Canada, of designing for a number of classes of severe accidents, and limiting their predicted consequences.

2.4 2.4 TWO–GROUP PHILOSOPHY

For common–cause events such as earthquakes, fires, and missiles, Canada has developed the two–group approach. All important systems in the plant are divided into two spatially separated and independent groups, either of which can, by itself, shut the plant down, remove decay heat, and monitor the plant safety status. Physical protection or environmental qualification is provided so that at least one group will be available when required – e.g., a protected secondary control area is provided in case the main control area becomes uninhabitable due to an earthquake or fire. This approach reduces the chance of severe core damage due to common–cause initiators.

3. DESIGN ASPECTS RELATED TO SEVERE CORE DAMAGE

3.1 THE CANDU CONCEPT

The CANDU is a natural uranium fuelled, heavy water moderated, heavy water cooled reactor. The pressure–boundary in the core consists of several hundred 10 cm. diameter pressure tubes, each containing twelve or thirteen short (0.5m) fuel bundles. Surrounding each 0.44 cm. thick pressure tube (Fig. 3.1) is a 0.14 cm. thick calandria tube; between the calandria tube and the pressure tube is an insulating gas–filled gap, which reduces normal heat loss to the moderator.

3.2 PRESSURE TUBES AS PRESSURE BOUNDARY

All the reactivity devices (control rods, shutoff rods) penetrate the moderator but not the coolant pressure boundary, and so, with one exception, are not subject to hydraulic forces from a loss–of–coolant accident (LOCA). The exception is a channel failure, which is a small break LOCA. The calandria tube may or may not contain the pressure tube rupture, but no credit is taken for this in the design, and relief pipes, sized for channel failure, are provided to protect the calandria vessel (which contains the moderator) from overpressure. It has also been shown ([Ross–Ross, 1963], [Muzumdar, 1987]) that a channel failure will not propagate to other pressure tubes, nor will it damage more than the neighbouring few shutoff rod guide tubes. The shutoff rod shutdown system is therefore provided with enough redundant mechanisms (typically 28) to remain effective without credit for the rods with possibly damaged guide tubes, nor for the most effective rod among those with undamaged guide tubes.

3.3 DECAY HEAT REMOVAL

There are a number of emergency heat sinks for decay heat from the fuel, which can prevent or mitigate a severe accident.

3.3.1 Shutdown Cooling System

A shutdown cooling system is provided in all CANDUs for normal decay heat removal (as an alternative to the boilers) and for cooldown below 100C. It can operate at full heat transport system temperature and pressure, and can therefore be used as an emergency heat sink from hot, shutdown, full–pressure conditions, should the boilers be unavailable.

3.3.2 Emergency Water System

Most CANDUs also have an Emergency Water System (EWS), seismically qualified to remove heat after a Design Basis Earthquake. It provides a supply of water to the boilers independent of the normal and auxiliary feedwater. Since it is sized to remove decay heat shortly after shutdown, it can also be used as an emergency heat sink should normal and auxiliary feedwater, and the shutdown cooling system, all be unavailable.

3.3.3 Emergency Coolant Injection

As in all water–cooled reactors, an emergency coolant injection system refills the fuel channels following a loss–of–coolant accident. To ensure that injection for small breaks is not blocked by high Primary Heat Transport System pressure, the main steam safety valves on the main steam lines are opened on a loss–of–coolant signal. These are sized to cool down the boiler secondary side, thereby depressurizing the PHTS. Emergency coolant injection thereafter is conventional recovery of water from the reactor building sump, and re–injection.

3.3.4 Moderator

In normal operation, about 5% of the thermal energy produced by the fuel is deposited into the moderator, by radiation and direct nuclear heating and, to a much smaller extent, by conduction through the insulating gas gap of the pressure–tube/calandria–tube assembly. This heat is removed by a moderator cooling system, consisting of pumps and heat exchangers. In a severe loss–of–coolant accident, the same system will remove decay heat from the fuel channels, even if they contain no coolant at all. Fuel would be severely damaged, but would not melt, and the channel would remain intact and contain the debris. This capability has been verified by full–diameter channel tests at the Whiteshell Research Laboratory, as described in Section 5. The moderator is thus a distributed low–pressure emergency heat sink surrounding each fuel channel.

3.3.5 Calandria and End–Shield Cooling System

In normal operation, heat is generated in the calandria shell and in the end shields which support the fuel channels and which provide radiation shielding for the reactor vault in front of the reactor faces. This heat, amounting to about 0.3% of the full–power heat generation, is removed by a dedicated shield cooling system. In addition, the cylindrical calandria shell is located inside either a metal shield tank, or a concrete vault, either of which is filled with water to provide both cooling and radiation shielding (Fig. 3.2). The vault floor itself is typically 2 1/2 m. thick. Should the emergency coolant injection system, and the moderator heat sink be lost after a loss–of–coolant accident, the shield cooling system can, depending on the failure sequence, prevent melt–through of the calandria vault or shield tank, or delay it for many hours as the shield water is boiled away. The analysis which supports this conclusion is discussed in Section 4.3.

3.4 3.4 CONTAINMENT

CANDU reactors use two types of containment (Fig. 3.3): single–unit containment, at the CANDU 6 nuclear generating stations, and multi–unit vacuum containment, at the Ontario Hydro nuclear generating stations. Both use a high–rate water spray, called dousing, which condenses steam released in an accident and reduces the containment pressure. In the single–unit containment, the reactor and the dousing system are all located in the same building. In the vacuum system, four or eight reactors, each with its own local containment, are connected by large ducting to a separate common vacuum building kept, as its name implies, at near–zero absolute pressure. Should steam be released from a pipe break in the reactor building, it, and any radioactivity, is sucked along the duct and condensed by dousing in the vacuum building. Long–term pressure control is by local containment air coolers, and by a filtered air discharge system.

The dual failure "loss of coolant plus loss of emergency coolant injection", while it does not lead to a loss of core geometry, nevertheless permits the fuel to reach high temperatures. The Zircaloy fuel sheaths can be highly oxidized, and the hydrogen gas which evolves will make its way through the break to containment. The control of hydrogen following severe accidents depends on the station design. In the vacuum containments, a network of hydrogen ignitors, powered by the most reliable source of electricity (Class I batteries), is engineered to reduce local flammable hydrogen concentrations before they can reach explosive conditions, and to ensure that the energy from combustion is released gradually. Alternatively, the natural circulation through the reactor vault can be accounted for. In particular, for the single unit containment at Point Lepreau, the accident analysis for loss of coolant plus loss of emergency coolant injection (LOCA/LOECI) predicts that flammable concentrations (in excess of 4% hydrogen) of the mixed containment atmosphere are not reached. Should a flammable concentration nevertheless occur, the analysis also shows that the containment would not be damaged by the pressures generated by the burn. The reason for the low hydrogen concentration, even for LOCA/LOECI, is that when the moderator acts as an emergency heat sink, it limits the pressure tube temperatures below the level at which significant oxidation can occur. Thus the pressure tube metal does not contribute significantly to the hydrogen source term. This is discussed further in Section 4.

3.5 REACTIVITY CONTROL AND SHUTDOWN SYSTEMS

The reactivity control devices penetrate the low–pressure moderator but not the coolant pressure boundary, as noted earlier, so they are not subject to pressure–assisted ejection (a channel failure is too small a break to develop enough pressure within the calandria to delay the rods significantly). The maximum rate of reactivity addition from the control devices is set by their inherent mechanical or hydraulic operation – normally this is 0.1 mk/sec (1 mk or "milli–k" is about 1/6 beta or 16 cents), and at most it is about 1 mk/sec during shutoff rod withdrawal from a shutdown state. The total reactivity holdup in the movable control devices is about 15 mk. This low value is set not by the need to compensate excess reactivity, but by operational requirements on decision and action time after a reactor trip. The pressure–tube, natural uranium concept permits on–power refuelling as the longer–term means of reactivity control.

CANDU stations control reactor power automatically over the entire range from 6 or 7 decades below full power up to full power. At low powers, up to about 10% full power, power measurement is based on ion chambers, while at high powers, in–core flux detectors are used. Both types of measurement are sufficiently prompt for all practical purposes.

Reactivity control at all power levels, both for bulk and for spatial purposes (spatial control is needed only above 25% power), is based on water–filled zone controllers. If their worth is inadequate, mechanical control absorber and adjuster rods are available for both positive and negative reactivity addition, again under totally automatic control. There is also poison addition to, and removal from, the D_2O moderator, both of which are very slow and relatively rarely required.

Protection against reactivity insertion accidents is provided both by the control system itself, via power stepbacks on high rate log and high flux, and also by powerful, rapid shutdown – see [Snell, 1986 December], [Snell, 1987], and [Howieson, 1987] for more detail. In the CANDU 6 for example, shutdown system 1 consists of 28 gravity–operated, spring–assisted absorber (shutoff) rods, and shutdown system 2 consists of 6 nozzles which inject liquid gadolinium nitrate, at high pressure, into the moderator. Each system is, independently, fully capable of shutting down the reactor for all accidents. Each system has its own detectors, amplifiers, relays, logic, and actuating mechanisms, and is independent of the control system and of the other shutdown system. Because the shutoff units are inserted into the low–pressure liquid moderator, they can respond very quickly to an accident.

In particular, the trip parameters on each system are chosen to provide redundant coverage, where practical, for every accident in the design/licensing set. These trips have been studied extensively in terms of their trip coverage (i.e., the range of initial power level and process conditions for which the trips are effective), and the combined reactor trip coverage is found to be fully comprehensive.

The fastest means of reactivity insertion is through the large loss of coolant accident. Thermohydraulic effects limit the rate to less than 4 mk a second for the worst break. For a large LOCA, the initial rate of rise of reactor power is 50–100%/second. These rates determine the speed of the shutdown safety systems, which must therefore act within about two seconds for the most severe LOCA, a rate achievable with mechanical or hydraulic devices. For hypothetical reactivity insertions even **above** prompt critical, the rate of power increase is set by the longer prompt neutron lifetime of CANDU (at least ten times that of light water power reactors). Thus the rate of rise of power is not very sensitive to going beyond prompt criticality in CANDU.

All recent CANDU reactors have two fully independent shutdown systems, either of which can, by itself, terminate any reactivity insertion accident or LOCA. The provision of dual, fast, independent shutdown systems means that, for these reactors, Anticipated Transients Without Scram, including LOCA, are low enough in probability that they can be ignored for design purposes, as they are a negligible contribution to total risk.

4. SEVERE ACCIDENTS

4.1 PROBABILISTIC SAFETY ASSESSMENTS

In Canada, the nuclear industry (designers and plant owners/operators) has taken the lead in performing probabilistic safety assessments (PSAs). From the early use of risk to define design requirements, the PSAs evolved in the 1970s to fault tree/event tree analyses, which were used to confirm the reliability and separation goals specified for the design.

The first such study was performed by AECL in 1975 on the service water system at Ontario Hydro's Bruce–A Generating Station. The benefits from this study were:

a. a comprehensive identification of crosslinks, since service water has interfaces with many systems;

b. identification of which support functions needed backup cooling water; and

c. definition of the necessary operator actions to mitigate the loss of service water.

PSA studies (called Safety Design Matrices at the time) were performed from 1978 to 1983 on the CANDU plants in the design and construction phase during that period ([Gumley, 1985], [Shapiro, 1986]).

The regulatory agency also recognized the usefulness of the studies for the Bruce–A Nuclear Generating Station, and the PSA studies became part of the licensing process for all CANDU stations constructed in Canada after Bruce–A.

These PSAs went beyond the assessment of support systems, and were used to systematically identify and quantify all major sequences that could release radiation from the plant at credible event frequencies [Rennick, 1987].

4.1.1 Darlington Probabilistic Safety Evaluation

The Darlington Probabilistic Safety Evaluation (DPSE) [Ontario Hydro, 1987] was carried out by Ontario Hydro as part of the design verification process for the 4 x 880 MW(e) Darlington station, during the final stages of station design and construction. The study is essentially equivalent to a Level 3 Probabilistic Risk Assessment (PRA); however, no detailed off–site consequence analysis was performed for certain low probability severe accidents. The scope of the study in addition did not include initiating events external to the station nor those arising from internal fires and flooding, which were judged unlikely to contribute significantly to risk because of rigorous, deterministic design criteria, as discussed in Section 2.4.

Important features of the DPSE include a very comprehensive set of initiating events internal to the plant and a high degree of detail in the fault tree analysis of all major process, safety and support systems, including specific modelling of potential instrumentation and control failures, e.g., [Raina, 1986]. Human reliability modelling was likewise comprehensive, including detailed identification of human error opportunities both prior to and after the initiating event. Preliminary human reliability quantification was performed using prepared data tables [Iwasa–Madge, 1985]. Final quantification of important human error probabilities was obtained by using a structured expert judgment procedure involving plant operations staff.

The spectrum of potential core damage was divided into ten fuel damage categories (FDCs), labelled FDC0 to FDC9. Of these, FDC0 to FDC3 cover the range of events considered to meet the severe accident definition in this report. FDC1 to FDC3 deal largely with loss–of–coolant initiating events accompanied by failure of emergency core cooling, either on demand or during the mission time, in which the moderator is called upon to act as the heat sink. FDC0 contains all events with the potential to cause a loss of core structural integrity. This can occur due to the failure of the moderator to act as a heat sink when required, failure to shut down (if such failure would result in fuel damage), or severe over–stressing of the calandria structure.

The magnitude of fuel damage associated with FDC3 is quite small and largely represents an economic, rather than public health, risk. FDC2 results in significant fuel damage, and FDC1 is conservatively estimated to result in 15–30% of the core equilibrium fission product inventory being released from the fuel. FDC0 could result in a greater or smaller release, depending on the nature of the mechanism causing loss of core structural integrity.

The frequency estimates are the result of complete computer–assisted integration of the event trees and fault trees, fully accounting for system crosslinks. Due to the level of detail, the development of special methods and procedures was needed in order to simplify the fault trees and structure the integration process to make it computationally feasible ([Chan, 1987], [King, 1987]). Table 4.1 contains the DPSE mean frequency estimates for the severe accident categories.

TABLE 4.1: DARLINGTON SEVERE ACCIDENT FREQUENCY ESTIMATES

Category	Description	Mean Frequency (/reactor–yr)
FDC3	Moderator required as a heat sink more than 1 hour after reactor trip	6×10^{-4}
FDC2	Moderator required as a heat sink from 200s to 1 hour after reactor trip	8×10^{-5}
FDC1	Moderator required as a heat sink less than 200s after reactor trip	2×10^{-6}
FDC0	Potential for loss of core structural integrity	4×10^{-6}

The severe core damage frequency is bounded by the frequency for FDC0, and at 4×10^{-6}/reactor–year, it is very low indeed.

The complex multi–unit CANDU containment includes, as described earlier, a negative–pressure vacuum system and an emergency filtered air discharge system. The containment event trees include failures of: overpressure suppression, envelope integrity, long–term pressure control and filtration [Dinnie, 1986]. Fuel damage and containment failure logic were fully integrated to search for potential crosslinks. Consequences were estimated representative of a wide range of fuel damage category and containment subsystem failure combinations. The results include an estimate of the frequency of a large release from the core accompanied by the potential for loss of the containment function, leading to the possibility of a large, offsite release. The mean frequency was estimated to be 8 x 10^{-7}/reactor–year.

The overall conclusion is that the calculated health risk is very low.

4.1.2 Probabilistic Risk Assessment of CANDU 6

In 1986 and 1987, AECL performed a CANDU Level 2 Probabilistic Risk Assessment study of CANDU 6, because some countries have expressed a need to be able to compare the overall safety of reactors as a factor in making a choice of a reactor option. This involved a probabilistic evaluation of events of a frequency of less than 10^{-7}/year, and a consequence analysis of severe core damage events and related releases [Howieson, 1988].

In performing the probabilistic evaluation, the following groundrules were used:

1. The reference plant was an existing Canadian CANDU 6 unit, licensed for operation in 1983, with the addition of automatic cooldown of the heat transport system on high end–shield temperature. The required licensing, operating, and design information was readily available.
2. As assumed in water reactor probabilistic risk assessments, the initial plant state was 100% full power operation. External events such as earthquakes and fires were not assessed in this study, although they of course are covered deterministically in the design, as described in Section 2.4.
3. The reactor core contains equilibrium fuel.

Previous probabilistic safety assessments of CANDU proved to be valuable input to the CANDU 6 PRA.

Fault trees were used to determine the frequency of the initiating events, and the failure probability of the mitigating systems. Event trees were used to assess the plant response following the initiating event, and incorporated the possibility of failures of the required mitigating systems.

In the preparation of the event trees, crosslinks were identified between systems, up to the level of major components and electrical power supplies; however, crosslinks between systems via control components (e.g. contacts, relays, and fuses) were not examined.

The operator model used was the same as was used in the previous CANDU 6 Probabilistic Safety Assessment studies. The PSA operator model is a post–initiating–event model, in which operator actions are shown explicitly in the PSA event sequence diagrams. This approach was carried over, and the operator actions were explicitly shown in the event trees. A brief comparison was made, where appropriate, to other operator models.

This preliminary CANDU 6 study was sufficient to identify the major risk contributions with a high degree of confidence because:

1. The detailed design is "known" and has already been subject to probabilistic safety analyses; in addition, the reference plant has been running successfully for the last five years.
2. The CANDU design philosophy calls for independence of safety systems and process systems. This approach minimizes the potential for crosslinks between the two types of systems. As discussed in Section 2.1, separation has always been a key issue in design and licensing of CANDU, and is verified on each plant by an exhaustive review during the design and construction, by the designer, the utility, and the regulatory board.

The study analyzed a total of thirty–two "internal" initiating events, with detailed event tree analysis to estimate frequencies of release categories. As shown in Fig. 4.1, the severe core damage frequency for the reference CANDU 6 plant is of the order of 5 x 10^{-6} per year. This low frequency, comparable to that found for the Darlington plant as discussed above, reflects the presence of the moderator as an emergency heat sink. The major contributor to severe core damage is loss of service water, as this affects cooling water for systems such as the moderator, calandria vault, secondary heat transport system and emergency coolant injection system heat exchangers. The study also found that the initiating event leading to a power excursion coupled with a failure to shutdown is very unlikely: 3 x 10^{-8}/year. This low value results from the use of two independent, rigorously tested shutdown systems in CANDU.

The analysis of severe accident releases drew upon existing Canadian safety analysis for predictions of the containment behaviour for many events. For severe core damage sequences, the significant CANDU design features are the moderator water surrounding the reactor fuel channels, and the shielding/cooling water surrounding the calandria. This water (even without cooling) provides an inherent heat sink for many hours, limiting the progression of a severe core damage sequence. Finally, if the sequence should progress further, the CANDU 6 prestressed concrete containment may crack (but is unlikely to fail) due to overpressure, reducing potential releases from a severe accident.

Further work is underway to refine the operator and consequence models, as described in [Howieson, 1988].

4.2 SEVERE ACCIDENTS SETTING DESIGN REQUIREMENTS

4.2.1 LOCA with Coincident LOECI

As discussed in Section 2, one of the classes of postulated events considered in CANDU licensing analysis is a loss of coolant accident (LOCA) coincident with the failure of the Emergency Coolant Injection (ECI) system to operate on demand. Given the high reliability (99.9 percent) of the ECI system, such a combination of events is extremely unlikely to occur.

Analyses of LOCA/LOECI sequences focus on demonstrating that the regulatory dose limit is met and on verifying that the safety design target of maintaining the integrity of all the fuel channels is also met. The maintenance of channel integrity provides assurance that the fuel bundles remain within their respective channels throughout the accident. Thus, the geometry of the reactor core is well defined and can be analyzed on a channel–by–channel basis, to provide estimates of the timing and extent of fission product release and hydrogen generation.

Over the past few years, considerable effort has been devoted by the Canadian utilities and AECL to further characterize the phenomena relevant to this class of accidents. Conservative analytical models, supported by a large body of experimental data (Section 5), are now in place and are used to assess the consequences of such accidents. Highlights of the recent developments in these models are now discussed, along with the mechanisms and factors that impact on channel integrity, fission product release, and hydrogen evolution during these accidents. Analytical models and methods vary somewhat throughout the Canadian nuclear industry; the models and methods presented here are typical, but the specifics are those currently used by Ontario Hydro.

4.2.1.1 Large LOCA With Coincident LOECI

Postulated large LOCAs are characterized by rapid coolant voiding in the fuel channels which induces an overpower transient. Reactor shutdown systems are activated by one of several redundant trip signals and consequently reactor power is reduced to decay power levels within seconds. The PHTS depressurizes at a rate determined by the break size. For most large breaks there is sufficient convective cooling throughout the transient to avoid severe fuel temperature excursions and consequent large fission product releases, unless the ECI system is assumed unavailable. Severely degraded fuel cooling and significant activity release result if it is assumed that the ECI system fails to operate on demand.

The overall methodology used to assess CANDU core behaviour during large LOCAs with ECI unavailable is shown schematically in Fig. 4.2. System thermohydraulic codes are used to determine the pressure transient and to provide an estimate of the range of transient heat removal conditions in the core. Variations in individual fuel channel characteristics such as initial power, elevation and feeder hydraulic characteristics, in conjunction with possible variable conditions in the reactor headers (e.g. coolant phase separation), produce a wide range of channel conditions which may co–exist in the reactor core during a degraded cooling accident. In the assessment of the thermal and mechanical behaviour of the fuel and fuel channels, channel flow transients are derived from a conservative estimate of the system behaviour. Various conservative channel flow transients are applied to groups of channels with similar characteristics. The results of these single channel analyses are then combined to provide a bounding assessment of the core behaviour under severely degraded cooling conditions.

Under such conditions, fuel heatup leads to fuel deformation and may cause pressure tube yielding ([Howieson, 1986], [Brown, 1984], [Muzumdar, 1983 January], [Muzumdar, 1983 May]). The coolant pressure at the time of overheating determines the mode of pressure tube deformation. Higher pressures (greater than approximately 1 MPa) lead to pressure tube ballooning and, at 16% pressure tube strain, to complete circumferential contact between the pressure tube and the calandria tube in the overheated region (Fig. 4.3). At lower pressures (near atmospheric), pressure tube sag is more prevalent and leads to more localized contact between the pressure tube and the calandria tube. At intermediate pressures (between atmospheric and 1 MPa) a combination of sag and ballooning can result in localized contact followed by complete circumferential contact between the pressure tube and the calandria tube. In all cases – sag, strain, or no contact (for regions of lower power) – a heat removal path to the moderator is established which is effective in limiting the fuel temperature excursions, and consequently limiting fission product release and hydrogen production ([Gordon, 1982], [Lau, 1981]). The detailed assessment shows that:

Over the entire range of large break LOCAs with ECI unavailable, the fuel temperatures do not reach the melting point of UO_2.

An assessment is performed to ensure that channel integrity is maintained throughout large break LOCAs with coincident loss of ECI. The potential for pressure tube failure prior to uniform pressure tube contact with the calandria tube is examined. The potential failure mechanism under these conditions is local overheating of the pressure tube followed by rapid local strain to failure. It is found that for the expected range of contact conditions between the fuel and a pressure tube, local pressure tube overheating is **not** severe enough to cause localized over–strain and failure prior to pressure tube/calandria tube contact [Gulshani, 1987a].

The prevention of sustained, calandria tube dryout following pressure tube contact is a sufficient condition for pressure tube integrity, since it ensures that the fuel channel will not strain further. The factors which affect the potential for calandria tube dryout are the pressure tube contact temperature (i.e. the stored energy available), the contact heat conductance between the pressure tube and calandria tube (i.e. the ease with which heat can be transferred from the pressure tube to the calandria tube), and the subcooling of the surrounding moderator (since this determines the magnitude of the critical heat flux). The experimentally–determined relationship between these factors is presented in Section 5.2 (see Fig. 5.1, [Gillespie, 1981] and [Gillespie, 1982]) and its application is discussed in [Archinoff, 1984], [Brown, 1984], and [Muzumdar, 1982 March]. An assessment is performed of the transient, local moderator subcooling required to prevent calandria tube dryout anywhere in the core. This required moderator subcooling is then compared to the predictions of the available moderator subcooling during the accident. The assessment of the transient spatial variation of the moderator subcooling throughout the accident includes the effect of the additional heat load due to pressure tube/calandria tube contact in a number of channels in the core.

The thermal and mechanical behaviour of a fuel channel under degraded convective cooling conditions is assessed ([Lau, 1986], [Akalin, 1982], [Reeves, 1985], [Reeves, 1982]), including the feedback effect of pressure tube and fuel deformation on the distribution of steam flow in a channel, and consequently on the fuel and pressure tube thermal behaviour (Fig. 4.2). Pressure tube ballooning and/or fuel bundle slumping promotes the bypass of steam flow around the interior of the fuel bundles in a channel ([Reeves, 1983], [Akalin, 1983 February], [Akalin, 1983 September]). Thus, the extent of both the exothermic Zircaloy/steam reaction and the convective heat removal may be reduced in the central region of the fuel bundles due to a limited steam supply. Depending on the rate of fuel heatup, the extent of the exothermic Zircaloy/steam reaction may be further reduced due to relocation of the molten Zircaloy–4 sheath material [Lau, 1987], as now discussed.

If the fuel sheath is not completely oxidized when the Zircaloy–4 melting temperature is attained, then there is potential for the molten Zircaloy to react with the UO_2 fuel, form a low melting point eutectic, and relocate ([Akalin, 1985], [Rosinger, 1985 June], [Rosinger, 1985 September]). If the oxygen content of the molten Zircaloy is high, then the melt wets the UO_2 easily and tends to relocate into pellet cracks and dishes. If the oxygen content of the molten Zircaloy is low, the melt does not easily wet the UO_2 and tends to relocate along the outer surface of the element (Fig. 4.4). The results of experiments on CANDU fuel bundle behaviour at temperatures in excess of 1900C demonstrate this type of melt relocation behaviour ([Wadsworth, 1986], [Hadaller, 1984 September], [Kohn, 1985]). Contact of this eutectic with the pressure–tube has been shown not to threaten pressure–tube integrity – an experimental programme to confirm the models is underway.

Flow bypass due to pressure tube and fuel bundle deformation, and molten Zircaloy relocation, are both mechanisms which effectively reduce the overall rate of the Zircaloy/steam reaction in a channel. This exothermic reaction is an important source of heat to increase fuel temperatures under severely degraded cooling conditions, and also determines the timing and extent of hydrogen evolution from a channel. An example of the effect of fuel and pressure tube deformation on the cumulative hydrogen production is shown in Fig. 4.5, for a large LOCA/LOECI wherein a constant steam flow of 10 g/s is assumed to occur in **all** channels after 40 seconds [Blahnik, 1984]. Such a steam flow is chosen because it maximizes hydrogen production and heat addition due to the exothermic Zircaloy/steam reaction, while providing a minimum amount of convective heat removal.

The fuel temperature transients generated are used in the assessment of fission product release (Fig. 4.2). The distribution of active fission products within the fuel under normal operating conditions [Muzumdar, 1982], the timing and extent of sheath failure, and the transient release of fission products from the fuel are assessed [Archinoff, 1983]. The transient release mechanisms considered include pressure–driven release of the free inventory, rewet and/or high temperature release of the grain boundary inventory, temperature–driven diffusion release from the fuel grains, steam–enhanced grain growth and consequent grain boundary sweeping release, release from the fuel grains due to the reaction with molten Zircaloy, and long–term leaching release from the failed fuel in water [Lau, 1985].

At present it is conservatively assumed that any fission products released from the fuel are transported out of the channels, through the long, relatively cool feeders to the break, with no retention. Future research (see Section 5.5) will address the extent to which fission product retention in the PHTS delays and/or precludes the release to containment of various fission product species.

4.2.1.2 Small LOCA With Coincident LOECI

Small LOCAs are characterized by continued forward flow through the reactor core during most of the transient and relatively slow system depressurization. For these breaks, the reactor power regulating system can compensate for most, if not all, of the void–induced reactivity. Therefore, reactor power is maintained in the operating range until a reactor trip occurs.

In small LOCAs with an assumed coincident failure of emergency coolant injection, the fuel channels would receive adequate single–phase liquid or two–phase coolant (Fig. 4.6a) until well after reactor trip. Eventually, due to the unabated loss of coolant inventory from the PHTS, feeder connections at the supply header would be uncovered. The uncovered inlet feeders still contain low quality coolant, which must drain into the channel before single–phase steam cooling commences (Fig. 4.6b). There is also a substantial liquid level in the channel which contributes to effective cooling. Eventually, as the liquid in the channel boils off (Fig. 4.6c), the fuel is cooled by a decreasing flow of steam and fuel temperature excursions commence. The thermohydraulic response of a fuel channel under these conditions has been assessed using models that are well–verified against experiments ([Luxat, 1987], [Archinoff, 1986], [Hussein, 1985], [Gulshani, 1986]).

Slow boil–off in a fuel channel may result in temperature variations around the circumference of the pressure tube. If the pressure tube is locally hot enough to deform, then these temperature variations could result in localized over–strain and pressure tube failure prior to uniform pressure tube/calandria tube contact. Transient thermohydraulic information is used to assess the transient fuel and pressure tube temperature distributions at any axial location in a channel ([Locke, 1987 March], [Locke, 1987 June], [Locke, 1987 April], [Gulshani, 1987b], [Locke, 1985 June], [Lowe, 1986]). The results of analyses indicate that localized over–strain failure, due to thermohydraulic–induced circumferential temperature gradients on the pressure tube, is not expected for the range of conditions of interest in this accident.

Fission product release and hydrogen generation are bounded by the results for large break LOCA/LOECI scenarios. As in that case, there is **no fuel melting** in any small LOCA/LOECI.

4.2.2 Containment Impairments

The containment system is comprised of several active and passive subsystems. Active subsystems are those which are not normally operating prior to an assumed process system or equipment failure and are then required to operate. Passive subsystems are those which are in operation prior to a failure and continue to operate. The requirement to design for dual failures means that combinations of process system and containment failures must be analyzed; however, the independence which is designed into containment subsystems permits impairments of containment **sub**systems to be considered rather than total containment system failure.

The active subsystems which could be impaired are containment isolation and dousing. Isolation impairments include failure of the ventilation inlet or outlet dampers to close, and failure of isolation logic which implies that <u>both</u> inlet and outlet dampers fail to close. Due to separation of dousing into two subsystems in some CANDUs, a failure of both dousing subsystems is highly unlikely but has been examined in certain cases. The personnel and airlock door seals are normally inflated when the doors are closed. However, to account for the possibility that the airlocks have been recently opened and closed and the seals have not been re–inflated, deflated door seals are considered.

Therefore, the containment impairments considered for the dual failure analyses are:

1. failure of an isolation damper,
2. failure of containment isolation logic,
3. failure of dousing, and
4. deflated airlock door seals.

The containment response to an accident is determined by: the containment design (either multi–unit/vacuum building containment or single unit containment), the magnitude of break discharge, containment heat sources and sinks, and the particular containment impairment being considered.

In addition to the phenomenological response, discussed below, having to meet licensing requirements for dual failures has led to a significant emphasis on the reliability of containment isolation. A programme of tests during plant operation is established, so that the reliability of isolation on demand can be established at greater than 99.9%. Provision of test logic and the arrangement of components in order to demonstrate the availability of active containment subsystems are requirements of the conceptual design, and form an integrated part of containment.

4.2.2.1 Multi–Unit Containment

Following a postulated large LOCA, operation of the multi–unit negative pressure containment system would be as follows [Morison, 1984]. There is a large release of steam into the Reactor Building which results in a rapid increase in containment pressure. The self–actuated Pressure Relief Valves open automatically at 7 kPa(g). The steam enters the Vacuum Building where the pressure increase actuates the dousing system. The resultant spray condenses the steam, thereby reducing the pressure. A typical pressure transient is shown in Fig. 4.7. For the largest break size, the pressure peaks within the Reactor Building accident vault in 3 seconds, and returns to subatmospheric in about 60 seconds.

Assuming no containment impairments, the pressure slowly returns to atmospheric pressure as a result of the small inleakage of air from outside, and because of the consumption of instrument air by equipment located within the containment.

When containment pressure returns to atmospheric, the Emergency Filtered Air Discharge System (EFADS) is brought into operation to control the pressure at slightly sub–atmospheric. EFADS is equipped with particulate filters and charcoal filters for iodine removal, and radiation monitoring equipment to measure any radioactive releases from the station.

Envelope impairments reduce the time scale of repressurization, but the phenomena are basically similar. The vacuum containment is powerful enough to hold the reactor building subatmospheric for significant periods of time even **with** an impairment.

4.2.2.2 Single Unit Containment

A typical single unit CANDU containment pressure response with an impaired containment is shown in Fig. 4.8. The containment pressure increases due to the accident, then falls due to the combination of:

- reduced break discharge rates as the coolant pressure falls;
- dousing in containment which acts to condense the steam; and
- leakage through the impairment itself.

4.2.3 Anticipated Transients Without Scram

As discussed in Section 3.5, a hypothetical unterminated accident in a CANDU 6 would be an extremely improbable event, because many independent systems would have to simultaneously fail, namely:

* failure of a normal control system,
* **plus** failure or incapability of stepback,
* **plus** failure of shutdown system #1
* **plus** failure of shutdown system #2.

Such an accident has an estimated frequency of less than 1 in 10 million years per reactor in CANDU 6 [Howieson, 1988], and, in common with world practice on very low frequency events, requires no further design provision. Nevertheless the containment response to such a sequence is discussed in Section 4.3.2.

4.3 SEVERE ACCIDENTS WHICH CHALLENGE DESIGN

4.3.1 Loss of the Moderator Emergency Heat Sink

4.3.1.1 Background

In Section 4.2, it was shown that the moderator can provide an emergency heat sink for the fuel in the absence of normal coolant and emergency coolant injection. Studies for the Atomic Energy Control Board have examined the more severe consequences that would follow if even this emergency heat sink were to fail – namely, the effects of a loss of moderator heat sink capability in a Bruce–A NGS unit occurring **simultaneously** with the dual–failure accident of a large LOCA accompanied by complete unavailability of emergency coolant injection (LOECI) ([Rogers, 1984 June], [Rogers, 1984 August], [Rogers, 1984 September]). These studies were deterministic in nature in that they did not select the failure mode from a probabilistic analysis. In addition, they concentrated on the calandria thermal/mechanical response, rather than on the full spectrum of events accompanying such a severe accident – e.g., hydrogen production and transport. They nevertheless show instructive trends, as they reveal a further inherent heat sink beyond the moderator.

4.3.1.2 Hypothetical Accident Sequence

The study focussed on a hypothetical accident sequence following a large LOCA/LOECI in which the moderator cooling system fails, with the result that the moderator heats up, begins boiling and is eventually expelled from the calandria. As fuel channels are uncovered by moderator expulsion, they overheat and fall to the bottom of the calandria where they are quenched by the remaining moderator. Eventually, all moderator is lost from the calandria and the core debris heats up and melts. The study showed that the molten core material would be contained in the calandria which, because of the separately cooled water–shield heat sink, maintains its integrity throughout the accident sequence. This sequence is now described in detail.

4.3.1.3 Analytical Approach

The behavior of core components under the extreme conditions of this accident sequence is, in general, quite speculative since little experimental information is available. Thus, the approach used required that the essential features of the behavior of core components and the moderator be modelled on a physically sound basis; that sensitivity studies be undertaken; and that bounding analyses be used in certain cases.

4.3.1.4 Results

Typical thermal behavior of the fuel in different rows of fuel channels in this accident sequence is shown in Fig. 4.9, for a case where the mode of pressure tube deformation is assumed to be by sagging onto the calandria tube. The first temperature peak for any fuel channel row occurs while the channel is still covered by liquid moderator, while the second peak is predicted after channel uncovering. Actually, the channels would be expected to fail before the second peak is reached, when the pressure tube and calandria tube temperatures reach about 1750C. Fig. 4.9 shows that fuel in uncovered channels would be well below the UO_2 melting point up to the times of channel failure.

The amount of moderator remaining in the calandria, as a function of time, for the reference conditions assumed, is shown in Fig. 4.10 [Rogers et al, 1984 August]. As bulk boiling initiates and propagates downward through the calandria, at about 16 minutes, it is predicted that more than half the moderator is expelled. The moderator would be expelled in surges in this period but pressure peaks in the calandria are not severe (<220 kPa abs.) because of the low steam qualities (<4%) in the relief ducts. Subsequent rapid expulsions of moderator seen in Fig. 4.10 are caused by groups of fuel channels failing and dropping into the remaining moderator. Clearly the simplification of the model is responsible for the fine structure – in reality the pressure transient would be smoother. Pressure peaks during these subsequent flow surges are again low (<470 kPa abs.) and the integrity of the calandria is not threatened. For the reference conditions, Fig. 4.10 indicates that all the moderator is expelled from the calandria in about 50 minutes.

The study also showed that just before the last of the moderator is expelled from the calandria, almost all of the core debris in the bottom of the calandria is in the solid state and has been quenched to quite low temperatures (about 150C).

Typical results of the analysis of the subsequent heat–up of the solid core debris in the bottom of the calandria are given in Fig. 4.11. For a wide range of bed porosity, Fig. 4.11 shows the maximum and upper and lower surface temperatures of the bed as a function of time after reheating starts, following the loss of moderator from the calandria. The maximum temperature in the bed reaches the melting point for oxidized core material, about 2700C, about 80 minutes after the start of reheating, or about 130 minutes after the start of the accident. The upper and lower temperatures are well below the bed melting point at this time. The lower surface temperature is also well below the melting temperature of the stainless steel calandria wall and the lower surface heat flux into the shield–tank water, 15 W/cm^2, is well below the critical heat flux, about 280 W/cm^2, at the time that melting begins within the bed.

Fig. 4.11 shows that the thermal behavior of the debris bed is very insensitive to bed porosity. Other analysis showed that the thermal behavior was also very insensitive to pore size, material thermal conductivities and contact conductance between pieces of debris. It was concluded that the integrity of the calandria would be maintained during this stage of the accident sequence.

Once melting begins in the debris bed, some time will be required for the transition to a completely molten pool. No attempt was made to develop an analytical model of the debris bed for this transition stage. Instead, the time required for this period and the accompanying heat source decay were ignored, so that the analysis predictions are conservative.

Results for the analysis of a molten pool were inconclusive as to whether the pool would boil or not, depending on the property values used. Nevertheless, the maximum predicted heat flux into the shield tank water for all conditions (about 20 W/cm^2) is well below the critical heat flux. The interaction of the molten pool with the calandria wall, as illustrated in Fig. 4.12, indicates that there will be no melting of the calandria wall and that a solidified crust, over 2 cm. thick, would form on the wall, thus providing a protective shield. Analysis also showed that the heat flux into the wall would have to be about 100 W/cm^2, about five times the maximum predicted, before melting of the calandria wall would begin. Similarly, for conditions that would result in boiling of the molten pool, analysis showed that the calandria wall would be well–removed from melting conditions and the heat flux into the shield–tank water would be well below critical heat flux under both the boiling pool and the condensing vapor film above the pool.

These analyses indicate that the core material debris, whether solidified or molten, will not jeopardize the integrity of the calandria vessel, irrespective of whether the molten pool boils or not. Further calculations show that the molten pool would solidify at a time between 10 and 50 hours, depending on property values. Thus, it is concluded that the entire mass of core material would be retained within the calandria, as long as the shield–tank cooling water system continues to function. The consequences of failure of heat removal from the shield tank have also been assessed, as described in detail in [Howieson, 1988].

4.3.1.5 Conclusions

a. The moderator would be completely expelled from the calandria in about an hour.
b. No gross fuel melting would occur even when fuel channels are uncovered, and the core debris would not begin to melt until more than 2 hours after the accident begins.
c. The calandria would retain its integrity provided that the shield–tank water cooling system remains operational.
d. Core debris would be contained within the calandria and would begin to re–solidify in the period of 10 to 50 hours after accident initiation.
e. The shield tank system provides an additional heat sink to stop the progression of a severe core damage sequence.

4.3.2 Containment Response to Severe Accidents

The containment is designed for the challenges that could result from rupture of the largest main primary cooling pipe. For the typical example of the CANDU 6 reactor, the maximum pressure inside containment for this accident is predicted to be less than 70 kPa(g), well below the design pressure.

Consequential failures in the containment structure could occur due to severe core damage accidents which lead to addition of a large amount of energy to the containment atmosphere. A key feature of the CANDU containment structure is that it is a pre–stressed concrete building. Experiments at the University of Alberta ([Asmis, 1983], [MacGregor, 1980]) have demonstrated that at 330 kPa(g) internal pressure (2.3 times the proof test pressure), cracks would penetrate through the wall. Leakage through cracks is negligible at pressures below 345 kPa(g), and increases exponentially as the pressure is increased beyond that. At pressures approaching but still below the predicted failure load of around 530 kPa(g), the experiments suggest a leakage rate sufficiently high that the internal pressure is relieved; so it is difficult to have a condition in which the containment fails due to internal pressure loading.

This has a significant advantage – the structure would be unlikely to fail in a catastrophic way, and hence fission products would be largely retained inside a containment structure. The "wet" atmosphere therein will immobilize them further.

5. SAFETY RESEARCH

5.1 PROGRAMME GOALS

The aim of the safety research programme is to ensure a firm technical understanding of the various phenomena that could occur during an accident. The work of the safety program has covered both the loss–of–coolant accident, by now well understood, and, more recently, the lower probability severe accident conditions.

There exists a large amount of interaction with the technical community, both nationally and internationally. Much of the research in Canada is supported and coordinated by the CANDU Owners Group (COG); most of the rest is performed by AECL–Research. Internationally there is interaction through participation with formal agencies such as the Organization for Economic Cooperation and Development Principal Working Groups and the International Atomic Energy Agency, and also through various technical associations.

The main focus of the research is on aspects that are unique to the CANDU system. However, sufficient generic and underlying research is also performed to ensure contribution to, and an ongoing interaction with, international programmes.

5.2 FUEL BEHAVIOUR

An ongoing research programme on high temperature fuel behaviour has provided a solid database and verified codes to describe fuel behaviour under LOCA conditions. This has been achieved by separate effects experiments to evaluate properties of the UO_2 and cladding, development of computer models to describe sheath deformation and gas release processes, and in–reactor tests to provide all–effects verification of the behaviour of fuel and fission products. Using this methodology, current work is extending the database and models to the high temperatures associated with severe accidents. An advantage of the short (50 cm.) CANDU fuel bundles is that full–scale tests are relatively easy to do.

Work on fuel oxidation and consequent fission product release is being performed with both fresh and irradiated fuel. The oxidation kinetics of unirradiated UO_2 in air and steam has been studied at temperatures up to 1650C in steam and up to 1200C in air ([Cox, 1986 September], [Cox, 1986 October]) and is being extended to higher temperatures. To date, an extensive database [Iglesias, 1987] for oxidative release of fission products has been built from 60 experiments performed in the temperature range 400 to 1700C.

Severe fuel damage phenomena such as molten Zircaloy relocation, collapse of individual fuel elements, bundle deformation and UO_2–Zircaloy interaction are also being studied in the laboratory. Much of the work is performed with horizontal fuel element simulators.

The laboratory work to date has shown that the fuel cladding, oxidized by steam, has sufficient structural strength to maintain a coolable horizontal geometry, even during very high temperature transients. Other high temperature transient experiments with entire fuel bundles heated in steam to temperatures in excess of 1900C [Wadsworth, 1986] have shown that the relocation of molten un–oxidized Zircaloy to inter–element spaces, as discussed in Section 4.2.1.1, can lessen the effective oxidation rate by reducing the exposed surface area. Small–scale experiments are in progress to quantify this relocation phenomenon [Wren, 1986].

The interaction between the Zircaloy fuel cladding and the UO_2 fuel may be affected by the presence of a very thin layer of a graphite–based lubricant (CANLUB^{tm}) applied to CANDU fuel elements during manufacture. Experiments have shown that CANLUB^{tm} reduces the interaction at temperatures below 1500C (which is relatively slow), but above this temperature it has no effect [Lim, 1986].

5.3 FUEL CHANNEL BEHAVIOUR

The horizontal pressure tube/calandria tube fuel channel has led to an extensive research programme, which has been underway for a number of years, on fuel channel behaviour during postulated loss of coolant accidents involving coincident loss of emergency coolant injection. The work has established the conditions under which the residual heat in the fuel channel can be transferred to the moderator radially while maintaining fuel channel integrity. This heat removal may be accompanied by deformation of the pressure tube at high temperature, as it can contact the calandria tube either by ballooning at a high coolant pressure, or by sagging under its own weight when the coolant pressure is low.

Understanding the phenomena involved in the integrated tests on moderator heat sink effectiveness requires understanding of various single effects and the capability of modelling them. Such tests include studies of high temperature pressure tube sag and ballooning, development of high temperature creep deformation models, studies of

Zircaloy–steam reactions, measurements of critical heat flux in horizontal tube banks, and measurements of contact heat conductance and of high temperature Zircaloy emissivities.

Fig. 5.1 shows the results of a large number of integrated tests where the pressure tubes ballooned into contact with the calandria tube ([Gillespie, 1981], [Gillespie, 1982]). Shown are the boiling regimes on the outside surface of the calandria tube after contact with the hot pressure tube. If the surface of the calandria tube can be maintained in nucleate boiling or patchy dryout, it will be sufficiently cool that significant deformation will not occur. These data are used in accident analyses to assess fuel channel integrity, as discussed in Section 4.

Current experiments and model development are focused on the effect of temperature gradients on the deformation of pressure tubes, as discussed in Section 4.2.1.2. These experiments measure the temperature gradients and pressure tube deformation under various power, coolant pressure and coolant flow rate conditions.

5.4 BLOWDOWN TEST FACILITY

Single– and three–element in–reactor high–temperature fuel tests have been performed at AECL–RC for many years. In–reactor blowdown tests on CANDU fuel, at temperatures around 1000C, were performed in the U.S. Power Burst Facility reactor, and confirmed models of fuel behaviour in LOCA.

Now, a series of in–reactor severe fuel damage tests are planned. These will be performed in the new Blowdown Test Facility (BTF) ([Fehrenbach, 1987], [Wood, 1986]) in the NRU reactor. The purpose is to confirm fission product release fractions and chemical behaviour for overheated fuel, under depressurizing conditions. The focus will be on the release of **active species** fission products from fuel operating at temperatures in the range of 1500 to 2500C, and the subsequent transport and deposition of fission products in the primary heat transport system. The specific test objectives of this programme are:

- to measure the amount and timing of fission product activity release to the coolant during depressurizing conditions, during high–temperature post–depressurization, and during subsequent rewet, and to correlate the measured releases with the stages of fuel element behaviour;
- to measure the rate of fission product transport and deposition in carbon steel and stainless steel pipes, and determine the partition of fission product isotopes between liquid, solid, and vapour phases, and the chemical form of fission product species in the blowdown tank;
- to demonstrate techniques and procedures for decontamination of system components experiencing extensive fission product deposition and transport of irradiated fuel debris.

This programme will provide information on fission product behaviour that will be used to assess and refine the predictive ability of accident analysis codes.

The Blowdown Test Facility is shown schematically in Fig. 5.2 and its main design parameters are included in Table 5.1. The in–reactor test section of BTF is a vertical, re–entrant, pressure tube arrangement which will accommodate assemblies of three fuel elements plus a thermal shroud up to 70 mm in diameter and up to 3 m in length.

5.5 AEROSOL TRANSPORT

In contrast with other water reactors where there are substantial sources of aerosol material due to the presence of borated water, stainless steel structural materials and silver–cadmium control rods, CANDU aerosol source materials are limited to the Zircaloy fuel cladding, the UO_2 fuel and the fission products themselves. The smaller amount of low–melting–point material results in much lower aerosol densities for severe accidents in a CANDU. Currently the attenuation of these aerosol–borne fission products in the Primary Heat Transport System is not credited in CANDU safety analyses.

TABLE 5.1

NRU Blowdown Test Facility (BTF) Design Parameters

	Reactor and Test Section
Reactor power	130 MW
Cosine flux length	3.6 m
Mid–core (maximum) flux – thermal	1.7×10^{18} $n.m^{-2}.s^{-1}$
at cell boundary – fast	0.3×10^{15} $n.m^{-2}.s^{-1}$
Maximum fission heat in BTF	2 MW with pressurized water cooling 200 kW with superheated steam cooling
	Normal Operation **Coolant Conditions**
Coolant type	Recirculating pressurized water or superheated steam
Pressure (maximum)	10.5 MPa
Temperature (maximum)	water 300°C, steam 350°C
Flow (maximum)	water 10 kg/s; steam 1 kg/s
	Blowdown Conditions
Delay from loop isolation to reactor trip	0.1 – 60 s
Blowdown time to 1 MPa	10 – 300 s
Blowdown time to 0.3 MPa	30 – 500s
	(Post–Blowdown Stagnation)
Coolant type	Saturated steam or helium
Flow (steam)	2 – 20 g/s
Flow (inert gas)	variable
	(Rewet)
Coolant	25°C water
Rewet flow	0.04 – 4.8 kg/s
	(Post–Rewet)
Coolant	Once–through de–ionized water
Pressure	atmospheric
Temperature (inlet)	25°C
Flow	0.01 to 0.05 kg/s

The current focus of the research programme is to model the production and transport of aerosols which may be created in the Primary Heat Transport System [Mulpuru, 1987]. The laboratory experiments will be augmented by results from in–reactor BTF experiments. Development of a model coupling aerosol transport to thermohydraulics is also underway. This code development effort is supported by theoretical analysis of the validity of key assumptions which are used in aerosol physics models [McDonald, 1987 September a,b].

Key work on containment aerosols focuses on the two–phase jet at a break, in order to characterize the water droplet aerosols and their size distribution, and to determine the extent of fission product washout by the droplets.

5.6 FISSION PRODUCT CHEMISTRY

In parallel with the aerosol programme, there is a research programme to develop a fundamental understanding of fission product chemistry. The main focus is the chemistry of iodine in the containment building, although the chemistry of iodine and other fission products (Cs, Ru, Te) in the Primary Heat Transport System is also being addressed ([Wren, 1983], [Garisto, 1986]). The behaviour of iodine in the containment building depends on parameters such as pH, Eh, temperature, radiation fields, impurities in the sump water, and the presence of chemical additives that could be added to suppress iodine volatility. The work shows that, at equilibrium and at the low iodine concentrations ($<10^{-5}$ $mol.dm^{-3}$) of an accident, the dominant forms of iodine would be $I^-(aq)$ and $IO_3^-(aq)$ [Lemire, 1981].

The chemical kinetics data has been used in the formulation of the code LIRIC (Long–term Iodine Release Integrated Code) which predicts the iodine chemical forms and distribution between the aqueous and gas phase in containment [Wren, 1985].

Integrated tests are starting in the Radioiodine Test Facility (RTF) to validate the LIRIC model. The RTF is an intermediate–scale facility designed to provide radiation fields and chemical conditions appropriate to the reactor containment building conditions after an accident [Kupferschmidt, 1986]. Shown in Fig. 5.3, the main reaction vessel is a 400 dm^3 cylindrical vessel capable of containing a Co–60 radiation source of variable strength (0–10 $kGy.h^{-1}$) to simulate radiation fields encountered in an accident. The inner surface lining of the vessel can be altered to include painted and bare steel and concrete, and the temperature can be varied up to 80C. It includes extensive on–line instrumentation to monitor the various process variables and sampling systems for off–line chemical analyses.

5.7 HYDROGEN COMBUSTION

A study of hydrogen combustion has been underway for a number of years. The research addresses the hydrogen produced both by the Zircaloy/steam reaction and by the radiolysis of water. The experimental programs use both bench–scale tests as well as tests on intermediate scale vessels housed in the Containment Test Facility (CTF) [Kumar, 1984]. Important combustion phenomena which have been studied include flammability limits [Kumar, 1985], laminar burning velocities [Liu, 1983], and flame acceleration by venting [Kumar, 1987]. Also, the effectiveness of hot surface ignitors, such as glow plugs, has been examined for both lean and rich hydrogen–air–steam mixtures ([Tamm, 1985], [Tamm, 1987]) in a cooperative program between AECL, COG and EPRI (Electric Power Research Institute).

Current programmes focus on hot surface ignition limits, flame acceleration mechanisms, turbulent burning velocities, detonation limits, and transition from deflagration to detonation, to provide the basic information needed to develop and verify models for localized hydrogen combustion.

6. OPERATIONAL ASPECTS

Under the Canadian regulatory process, the licensee of a nuclear power plant is responsible to ensure that the plant staff and the general public are adequately protected from the consequences of plant accidents. Comprehensive studies are undertaken to ensure that, following accidents, essential features of safe plant operation are maintained. These include:

* habitability of control room(s),
* means to ensure reactor subcriticality after shutdown
* containment integrity,
* assured heat sink, and
* monitoring of plant safety status.

As part of this, operating procedures are developed, and staff trained, with the focus on stabilizing the plant Critical Safety Parameters (CSPs). Contingency response procedures are also developed to mitigate the consequences of an emergency and to provide assurance that all reasonable measures are undertaken to ensure human safety and to minimize property damage.

In the following, a fairly typical approach by a Canadian utility is described, which ensures post-accident operational safety. The specific example is the Lepreau I CANDU 6 plant in New Brunswick, owned and operated by the New Brunswick Electric Power Commission.

6.1 POST-ACCIDENT OPERATIONS REVIEW

All nuclear power plants are required to perform a plant-wide review of operation following a worst case loss of coolant accident. A detailed review of the relevant safety assessment documents, emergency plant operating procedures, and contingency plans is conducted in order to identify operator actions required to maintain essential plant safety functions. Each operator action is assessed for feasibility of execution based on its location, duration, access and expected radiation field.

The methodology for performing a radiation field study is described in [Natalizio, 1983] and is based on an improbable accident sequence which involves a break in the primary coolant circuit and a failure of a safety system (ECI or containment). The estimated worst case source term for a case of LOCA with impaired ECI is typically 10% halogens and noble gases and 3% particulates (of total core inventory).

The CANDU 6 moderator and primary heat transport system components are located inside containment, so the potentially hazardous high radiation fields following a LOCA are limited to: parts of the Emergency Coolant Injection System located outside containment; the reactor building ventilation; piping penetrations through the containment wall; and the airlocks.

To date, several post-LOCA operations reviews have been performed, covering short-term procedures and long-term equipment reliability. Although these studies generally confirmed the adequacy of the original design and operating procedures, a detailed assessment led to several recommendations to improve access to specific locations, to address the need for remote operability, and to confine contamination.

Examples of recommended changes included:

- Installation of additional shielding to improve control room habitability. At the Gentilly-2 CANDU 6 sister plant, a shielding wall was installed alongside the main airlock; and at Point Lepreau, a shielding wall was installed beside the reactor building exhaust filters.
- Provision of remotely operated isolating valves on instrument air lines to the reactor building at Gentilly-2 and Point Lepreau.
- Relocation of the ECI pump switchgear from the vicinity of the ECI pump pit to an accessible area, to permit operability and maintainability.
- Re-routing of the ECI system leakage to the reactor building to avoid excessive contamination of active drainage.

6.2 EMERGENCY OPERATING PROCEDURES

The day-to-day operation of a Canadian nuclear generating station is governed by the AECB-approved Station Operating Policies and Principles which define the envelope within which that station must be operated.

Besides the system-specific operating manuals covering normal operation, Emergency Operating Procedures (EOPs) are produced to cover abnormal or accident situations.

Where the cause of the upset can be recognized, an event specific EOP is produced. The ability to predict the anticipated plant response is perceived as a major advantage of these procedures, as they permit optimization of the corrective action. Typical event–specific EOPs include: dual computer failure; loss of services such as instruments or cooling water supplies; loss of main electrical power; loss of feedwater; LOCAs; and boiler tube failures. A generic EOP is also produced to cater to situations where the upset cannot be clearly diagnosed or identified; or the initial response by the operator proves inadequate; or the status of a Critical Safety Parameter is unsatisfactory.

An unsatisfactory status of a Critical Safety Parameter (CSP) indicates a threat to the integrity of the fuel sheath, or the Primary Heat Transport System, or the containment. Typical CSPs include the primary coolant sub–cooling margin, primary coolant inventory, reactor power, boiler pressure and level, containment pressure and radioactivity levels. Fig. 6.1a ([Kelly, 1986], [Kelly, 1987]) shows a typical response strategy to a plant upset, and Fig. 6.1b shows a guideline for controlling the CSPs.

Each EOP is developed, or is being developed, to meet the requirements prepared by a joint utilities task force [Kelly, 1987]. This document covers the complete life cycle of the EOP program which include its generation, verification, validation, issuance, training requirements and revision.

The operating staff are provided with comprehensive training to develop the necessary knowledge and skills to identify and respond to a plant upset. Training methods are normally a combination of classroom and field sessions, with the former providing the technical and procedural understanding and the latter developing the operating skills. This training may include control room training, plant walks–through or simulator training.

6.3 CONTINGENCY PLANNING

The starting point for all emergency plans that cope with severe radiation contingencies is the assumption that an accident causing highly radioactive releases to the environment can occur even if its occurrence is extremely unlikely. The emergency planning and preparedness systems must define the necessary countermeasures required to mitigate the consequences of severe accidents to protect public health and minimize damage to property. The Atomic Energy Control Board [AECB, 1984] requires the licensee to:

- develop on–site contingency plans for coping with emergencies within the facility boundary, and
- participate with federal, provincial and municipal governments to develop an off–site contingency plan to deal with those events which result in release of radioactivity beyond the station boundary.

Although detailed procedures differ in scope and methodology, the general approach followed at all CANDU stations is to define two sets of complementary plans: a general plan defining overall plant response strategy and a specific plan covering each category of contingency, namely: Radiation, Medical, Chemical, Security and Fire contingencies [Weeks, 1987].

The severity and extent of any particular contingency will dictate the degree of response required. Normally incidents are classified as Alerts or Emergencies. An Alert is declared for localized events which can be controlled by station staff and on–site resources, while an Emergency is declared when the contingency threatens more severe harm to site personnel, public, or the continued safe operation of the plant.

The key to the success of any Contingency Plan is of course the organizational framework, the efficiency of communications and level of expertise and training of the response groups involved. Fig. 6.2 shows a typical organization chart for the purpose of contingency planning. The Shift Supervisor on duty retains the overall responsibility for all response duties as head of the Command Unit. The Response Team consists of a group of designated and specially trained shift staff as part of the normal shift complement. The Assistance group is assembled from the non–shift staff to provide senior–level technical and operational advice to the Shift Supervisor in case of an emergency.

A comprehensive training program is developed to provide staff with the necessary knowledge and skills to support the contingency–related activities, the extent of training being commensurate with the individual's role in the overall plan. Thus the Response Team members, because of their key function, receive extensive advanced training in a variety of topics which include fire–fighting, first aid, chemical protection, and all the specific contingency plans.

7. CONCLUSIONS

Inherent CANDU properties, namely:

- a moderator which acts as a dispersed emergency heat sink for fuel heat;
- the presence of a water–filled shield tank which can prevent melt–through of the calandria; and
- a containment which exhibits forgiving behaviour under hypothetical overpressure conditions;

all contribute to a design for which the probability of severe core damage is low, of the order of 5 x 10^{-7} per reactor–year, and the consequences of core damage are limited.

The licensing philosophy of examining dual failures as part of the design basis set, has led to redundancy of shutdown which makes an unterminated accident a negligible contribution to total risk, and to a design which will accommodate impairments in the containment and emergency coolant injection systems.

Furthermore, these same characteristics mean that the plant response to increasingly severe accidents is gradual – there is no sudden change in behaviour.

The design characteristics and the licensing approach have also resulted in:

- a research programme which supports models for the predicted behaviour of CANDU for both loss of coolant and severe accidents, and
- a flexible approach to severe accident management on site.

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