5. SAFETY ASPECTS OF HWRs

5.1. INTRODUCTION

The purpose of this section is to:

- Summarize the status of the safety of current HWRs at an overview level,
- List major improvements in current designs,
- Indicate the directions that safety is likely to take in future designs.

In order to address increasing uncertainty versus time, this section discusses four time periods, each referring to when a plant could come into service:

- Safety of operating plants and current designs (Sections 5.2, 5.3, 5.4),
- Projected developments over the next ten years (Section 5.5),
- Projected developments between ten and twenty years from the present day (Section 5.6),
- Safety characteristics beyond twenty years from the present day (Section 5.7).

Next generation reactors are sometimes equated with passively safe concepts. However, although passive safety is a desirable characteristic in some circumstances, it is not necessarily a goal in itself. Accordingly, although this section deals with evolution towards passively safe aspects of HWRs, where these make sense, it does not automatically assume passive safety, nor does it cover passive safety aspects exclusively. An overview of IAEA activities on passive safety systems and international developments has recently been published [63].

5.2. DESIGN CHARACTERISTICS OF CURRENT HWRs RELATED TO SAFETY

There are a number of different HWR concepts, including PHWRs, pressure vessel HWRs, GCHWRs, etc. These are described in Section 3, where a brief summary of their safety characteristics is given. This section focuses on the safety characteristics of the most widely implemented concept, the PHWR.

The unique design concepts of this system offer certain intrinsic advantages with respect to safety, during both normal operation and accident conditions. Some of these key safety features are inherent to this type of reactor, while others are specifically engineered. The inherent safety features are elaborated in Section 5.2.1, while the engineered safety features are described in Section 5.2.2.

5.2.1. Inherent safety features

5.2.1.1. Fuel

The use of natural UO_2 fuel, with its low content of fissile material, precludes the possibility of a reactivity accident occurring during fuel handling outside the core or during storage. Also, there would be no significant increase in core reactivity in the event of any severe core damage accidents occurring which caused redistribution of the fuel by lattice distortion or otherwise. In fact, the latter remains true even if enriched fuels are used. As a result of the neutron economy of HWRs, the core reactivity at equilibrium is relatively independent of the fuel enrichment.

The thermal characteristics, namely, the low thermal conductivity and high specific heat of UO_2, permit the heat generated in a fast power transient to be initially absorbed in the fuel. Furthermore, the high melting point of UO_2 permits several full power seconds of heat to be safely absorbed above that contained at normal power.

Most of the fission products remain bound in the UO_2 matrix and may be released slowly only at temperatures considerably higher than the normal operating temperatures. Also, because the UO_2 is chemically inert to the water coolant, any defected fuel releases limited amounts of radioactivity into the coolant.

In an HWR, the use of 12 or 13 short length fuel bundles per channel, rather than full length elements extending the whole length of the core, subdivides the radioactive inventory that could be released to the coolant in the case of a fuel defect. The on-load refuelling facility in HWRs also has the singular advantage of allowing any defected fuel to be replaced with fresh fuel at any time.

The thin cladding used in fuel elements is designed to collapse under coolant pressure onto the fuel pellets. This feature permits high pellet–cladding gap conductance, resulting in lower fuel temperatures and consequently lower fission gas release from the UO_2 matrix into the pellet–cladding gap than if the cladding were free standing.

5.2.1.2. Reactivity and regulation

The neutron generation time for a natural uranium fuelled and heavy water moderated reactor is about 10^{-3} s. Thus, for a given reactivity, neutron transients are relatively slow, making the reactor easier to control, and there is no sudden decrease in reactor period as the reactor becomes prompt critical.

The excess reactivity reserve in the core for an on-power fuelled HWR is kept to a minimum. The reactivity worth invested in the regulating system is therefore low, typically ~15 mk. Thus, the individual rod worth and the total reactivity change possible due to any malfunction in the control system is limited.

In current HWRs, after equilibrium is reached through steady refuelling with natural uranium, there is no change in core configuration with respect to reactivity changes. This would also apply to other fuels such as SEU once equilibrium was reached. Thus, for 97% of its life, the core configuration of an HWR remains unchanged.

During certain postulated accidents, such as a LOCA or a loss of HTS circulation, voiding in the core could lead to positive reactivity effects. These effects are considered in the design of reactor shutdown systems. Each of the two fast acting shutdown systems can independently cope with the worst reactivity induced by a LOCA.

A heavy water moderated core provides a relatively spacious gap between fuel channels, which is occupied by the moderator. The reactivity control devices are installed in this low pressure, low temperature moderator environment. In the case of pressure tube reactors, there is no possibility for the fast ejection of any of the control rods and the environment is benign (low temperature, low pressure, low corrosion). The same is true in the case of accidents, except for localized effects due to single channel failures. Furthermore, the spacious core lattice allows complete separation between regulating and protective functions and devices. The space is also sufficient to provide two, independent, fast acting shutdown systems working on diverse principles. These features provide a high degree of assurance that reactor shutdown will occur when required.

The initial inherent response of an HWR to a power increase is a prompt and negative reactivity feedback; the increase in fuel temperature gives a rapid negative reactivity feedback through the Doppler effect. This prompt negative feedback is, however, relatively small in HWRs. Reactor control in normal operation is achieved by the movement of mechanical and hydraulic devices; it does not require, nor make much use of, the prompt reactivity feedback. A power increase can also lead to a positive reactivity feedback through the coolant density change; the time-scale is longer than for the fuel temperature feedback since it is controlled by heat transfer through the fuel to the coolant. It is less important in normal control since it is compensated for by control devices, but it becomes important in accidents and sets the rate requirements of the shutdown systems.

5.2.1.3. Core cooling

There is no large pressure vessel enclosing the core of pressure tube reactors. The largest rupture that can occur in the HTS is limited to the size of the reactor headers. In addition, the layout of the HTS is such that large diameter piping, including headers, is located above the top of the core. Thus, any single break in the system at or below the elevation of any fuel channels would be limited in size to that corresponding to one feeder pipe or pressure tube. With this arrangement, the core can be kept flooded with the emergency core cooling water in the event of a LOCA anywhere in the system.

The HTS design, wherein the steam generators are located at an elevation above the reactor, permits removal of decay heat following reactor shutdown, even when the HTS pumps are not operating. On loss of electrical power to the HTS pumps, the reactor will trip. Initially, coolant circulation is provided by the rotational inertia of the pumps, while subsequently natural circulation of the coolant is adequate for transferring the decay heat to the secondary coolant in the steam generators.

Even if there is a moderate depletion of inventory in the HTS, thermosyphoning will still continue to be effective under voided conditions. As the void increases further, unidirectional thermosyphoning may cease but fuel cooling will still be provided by other modes of buoyancy induced flow (e.g. intermittent flow), as long as the steam generators are available as a heat sink, and the HTS inlet headers are liquid filled.

The figure-of-eight circuit layout used in the HTS requires the coolant in each circuit to make two passes through the core. This arrangement slows down the rate of core voiding during a LOCA because for any typical break location, one pass through the core has a much longer communication path to the break than the other pass. This helps to limit the rate of reactivity transients associated with core voiding. In HWRs where a two loop concept is employed (e.g. CANDU 6), with isolatable interconnections between the two loops, the core voiding is limited mainly to one loop. However, as the reactor core size increases significantly, the reactivity benefit of subdividing the HTS into two loops, one on each side of the core, decreases. Voiding of one of the two loops introduces more than half the total core void reactivity, with the proportion increasing as the core gets larger, because the reactivity of the side of the core connected to the broken loop is enhanced by the flux tilt. Thus, large HWRs (~900 MW(e)) either have two loops with *no* loop isolation, or they use a single loop.

The fuel and the high enthalpy coolant are confined to pressure tubes which are surrounded by a large mass of relatively cold moderator heavy water in the calandria, about 130–300 Mg (depending on the reactor size). The calandria is further surrounded by cold shield water in the calandria vault, having a light water inventory of 350–600 Mg. In the case of certain combined failures, such as a LOCA coincident with the failure of the ECCS, the cold mass of moderator serves as an important heat sink which can preserve the fuel channel integrity and prevent gross melting of the UO_2. In severe accident sequences and assuming total failure of active heat sinks, including moderator heat removal, this inventory of water will dramatically slow down the progression of the accident, allowing time for remedial action to be taken by operators and time for emergency planning. In addition, the presence of the shield water around the calandria enables the calandria shell to be cooled and thereby serve for many hours as a core catcher in severe accident scenarios. The role of these cold water inventories is further discussed in Section 5.3.2.

5.2.2. Engineered safety features

The inherent safety features available in the HWR concept are complemented by various engineered safety features built into the design. This section discusses the principles followed for the design of engineered safety features in HWRs and provides a description of some of the salient features.

5.2.2.1. Safety design principles

In the engineering of the systems, well-established safety design principles and guidelines are followed, some of which are uniquely applied to HWRs. These are highlighted below.

(a) Defence in depth

The defence in depth pyramid is shown in Fig. 84. It is, of course, not unique to HWRs [64].

In following the principles of defence in depth, emphasis is placed on minimizing the challenges to the lower echelons in Fig. 84 [65]. Accident prevention is similar to that employed for other reactor types and, with one exception, will not be described in detail here. In fact, many of the events of moderate frequency can be handled by normal operating or process systems without needing to invoke any of the safety systems, for

FIG. 84. Defence in depth.

example, many deviations and transients can be safely handled by the reactor set-back or step-back, which reduce reactor power at a predetermined rate through the reactor regulating system. Small leaks from the HTS, such as those from breaks in instrument tubing, which are within the capacity of the D_2O feed pumps, can be handled by operator actions to shut down the reactor and actuate the small leak handling system.

One aspect of defence in depth particular to pressure tube HWRs is the application of 'leak before break' to the pressure boundary of the core (the pressure tubes). The critical crack length is well above the crack length at which leaks occur; such leaks are detected through an annulus gas system which monitors the gas between the calandria tube and the pressure tube in each channel. The operator can then shut down and depressurize the reactor long before a rupture occurs.

Leak before break can be defeated by the occurrence of excessive local zirconium hydride concentrations in the pressure tube, resulting from, for example, pressure tube/calandria tube contact. The sudden rupture of tube G-16 in Pickering NGS-A Unit 2 on 1 August 1983 highlighted this failure mechanism. The calandria tube did not fail. The reactor and its three sister units were subsequently retubed with zirconium–niobium pressure tubes which were used in all CANDU reactors after Pickering A. Zirconium–niobium does not hydride as fast as Zircaloy 2 but separation must still be maintained between the pressure tube and the calandria tube. In current HWRs, the pressure tube/calandria tube gap and pressure tube hydride concentration are carefully monitored to ensure that all known pressure tube failure mechanisms would be signalled beforehand by a leak.

(b) Two group concept

In order to protect the plant against common mode incidents such as fires and missiles, the safety systems have been divided into two groups that are functionally and physically independent of each other and which use diverse and independent support systems such as electrical power and service water. Each group has the capability to:

- Shut down the reactor,
- Remove decay heat from the fuel,
- Minimize the escape of radioactivity,
- Monitor the safety status of the plant.

Group 1 comprises:

- Normal plant operating systems, including the reactor regulating system, all process systems (except for the moderator cooling) and the main control room;
- The SDS1 and the ECCS.

Group 2 comprises:

- An emergency power supply,
- An EWS,
- SDS2 and containment systems,
- Emergency moderator cooling and circulation,
- A secondary control area or supplementary control room.

The EWS is plant dependent; in most CANDUs it consists of a portion of the dousing tank water and/or an external water reservoir, with pumps powered by the emergency power supply. In Indian HWRs, it consists of 'firewater' with diesel driven pumps.

The required safety functions, and the systems in each of the two groups to satisfy these functions, are shown in Table IX.[3]

TABLE IX. REQUIRED SAFETY FUNCTIONS OF GROUPS 1 AND 2

Required safety function	Mechanism employed	
	Group 1	Group 2
Shutdown of the reactor	SDS1	SDS2
Removal of decay heat	Steam generator cooling or shutdown cooling or ECCS	Cooling by EWS or firewater through the steam generator and moderator cooling
Minimization of radioactivity release	Decay heat removal via the steam generators or shutdown cooling or ECCS	Containment system Hydrogen ignition system (on recent HWRs)
Monitoring of the safety status of the plant	Main control room	Secondary control area and local panels/controls

[3] This allocation of systems to groups is typical of most HWRs. However, in the CANDU 9 design, the ECCS is a Group 2 system and containment systems are Group 1, allowing for a simplified layout.

(c) Redundancy

In line with the usual practice for nuclear power plants, adequate redundancy is provided in safety systems such that the minimum safety functions can be performed even in the event of failure of any single active component in the system. Over and above this ‘single failure criterion’, safety systems in HWRs are also required to meet specified unavailability targets, the evaluation of which takes into account the maximum permissible downtime of the equipment (specified in station technical specifications for operation) due to maintenance, etc., (see Section 5.2.2.1(d)).

Redundancy in the reactor protective systems includes triplicated instrument channels operating on a ‘two out of three’ coincidence principle. Each channel is independent of the other channels, with separate detectors, power supplies, amplifiers and relays. This arrangement ensures that safety functions will be performed reliably and will also allow testing and maintenance of a protection channel without affecting reactor operation.

(d) Availability

The four HWR safety systems, sometimes termed special safety systems (i.e. SDSI, SDS2, ECCS and containment), are designed to meet numerical availability targets, typically 0.999 for the most recent HWRs. This availability must be demonstrated during plant operation by periodic testing. The safety systems must be designed to permit such testing without it having a negative impact on operations.

(e) Separation

Physical and functional separation is ensured between each safety system and any process system, so that a failure in a process system does not impair the safety systems which are intended to protect against it or to mitigate it. As far as is practicable, this separation is also provided between different safety systems, as well as between redundant components within a safety system. These features ensure that a single local event (i.e. fire, missile impact, pipe failure) will not result in multiple component or system failures, and that the functions required for the safety of the reactor will not be impaired.

(f) Fail-safe feature

In order to minimize the probability of unsafe failures, wherever possible, the logic and instrumentation circuits are designed to fail in the safe direction. The limitations of this concept are due to it not always being possible to define a unique fail-safe position (e.g. HTS liquid relief valves), and also because a system which requires powered operation cannot be fail-safe on loss of power. Thus, systems which are mostly passive, such as the shutdown systems and the containment isolation system, are fail-safe on loss of control power, but the ECCS pumps rely on electrical power to operate.

(g) Seismic design

As with all current generation nuclear power plants, HWRs are designed and qualified to withstand earthquakes of defined magnitudes without posing a radiological hazard.

In general, three levels of earthquake are defined, one very infrequent and severe, the second more frequent and less severe, and the third similar to the level used for designing non-nuclear structures in the country. In the case of the first type, the requirement is for one set of systems to survive and perform the essential safety functions of shutdown, heat removal and plant monitoring. In the second type, a few more systems are qualified for operation and/or structural integrity. The third category covers non-safety related structures and equipment, and is provided for both economic protection and worker safety.

The Indian and the Canadian approaches can be taken as examples. In the Indian approach, three levels of earthquake are defined for a site, namely:

— *Safe shutdown earthquake (SSE)*: Earthquake of maximum potential at the site, with a mean recurrence interval of the order of not less than 10 000 years;
— *Operating basis earthquake (OBE)*: Earthquake of magnitude that can reasonably be expected to occur at the site during the life of the plant. Mean recurrence interval of not less than 100 years;
— *Codal*: Level used for earthquake resistant design of non-nuclear structures, as per national standards and practices [66].

All items required for the safe shutdown of the reactor (including decay heat removal) are designed for the SSE level. Items required for ensuring safety during continued operation of the plant are designed for the OBE level. All conventional equipment, and equipment the failure of which does not involve radiological risk, is included in this category. Some items may be assigned a seismic category higher than that warranted by safety considerations, for reasons of economic risk.

The Canadian approach is similar and also defines three levels of earthquake:

— *Design basis earthquake (DBE)*: This is the most severe seismic level and is derived from consideration of several factors, including local and regional seismo-tectonics, and extreme extrapolation of historical earthquake trends and earthquakes occurring elsewhere in the world in similar tectonic structures. The DBE is an engineering representation of the seismic ground motion at the site that represents the potentially severe effects of earthquakes applicable to the site and which has a sufficiently low probability of being exceeded during the lifetime of the plant.

— *Site design earthquake (SDE)*: This is the lower seismic level. The SDE represents a less severe earthquake than the DBE and is used in the design of some systems and components. In contrast to the DBE, the SDE is the representation of an earthquake with a probability of 0.01/a, based upon historical earthquakes only, for the site under consideration. The DBE and SDE are based on different premises and have no correlation or any fixed ratio. The SDE is an engineering representation of the effects at the site of a set of possible earthquakes with an occurrence rate, based on historical records, of not greater than 0.01/a.

— *National Building Code of Canada (NBCC) earthquake*: This is the minimum seismic design requirement. The seismic requirements of the NBCC apply to all other plant structures that are not qualified to DBE or SDE levels.

At least one set of safety systems is qualified to perform the essential safety functions following a DBE. The frequency of the combination ‘LOCA plus simultaneous DBE’ is so low that it need not be accounted for in the design. However, the frequency of the combined failure ‘LOCA followed after 24 hours by SDE’ is within the credible range, and therefore the plant must be designed for this combination. Thus, portions of the ECCS and its support systems are qualified to SDE. In practice, systems required after an SDE are often qualified to the DBE level as the cost differential is small.

(h) Practices relating to multiunit stations

Practices employed with regard to the sharing of safety systems in stations consisting of multiple units differ from country to country.

Multiple single unit CANDU 6 stations (e.g. the four unit station at Wolsong) have single unit modular designs, with very little sharing of safety related systems among the units; the emergency power supply, the EWS and parts of the fire protection system are shared on a twin unit basis.

Indian HWRs use a twin unit modular approach, and although systems important to safety are not shared, provision is made for manual ties between some of the safety support systems of the two units (e.g. in the electric supply system, process water and instrument air).

HWR stations operated by OH (Pickering, Bruce, Darlington) all consist of four or eight (in the case of Pickering) reactors with a common containment envelope, including a single vacuum building. A number of safety related systems are provided on a station rather than on a unit basis, e.g. electrical supplies, ECCS, EWS, emergency power system. However, the design ensures that if:

- The safety systems are required to serve all units, then they are designed accordingly, e.g. sufficient Class III power is provided for a loss of Class IV power to all units;
- An accident in one unit requires emergency core cooling or containment, it does not affect the required capability for an orderly shutdown or cooldown, or for decay heat removal in the other units.

(i) Operator actions during accidents

In general, actions required to handle accidents within a specified time frame are automated. With these provisions, for any operator action to be performed from the control room, the design must be such that at least 15 minutes are available after receipt of an unambiguous signal before such action is credited. In the case of actions to be performed from outside the control room, at least 30 minutes are allowed.

All such diagnoses and actions are defined in abnormal operating manuals at each station.

(j) Testability

Provisions are incorporated to ensure that active components in systems important to safety are testable periodically. Most of these tests can be done with the reactor on-power. Systems such as shutdown systems are designed to be testable from the initiating signal up to actuation of the end device, for example, a partial rod drop test is performed periodically during power operation on each shut-off rod to demonstrate that it is not stuck. The partial drop is designed so that it does not impact power operation.

(k) Environmental qualification

Safety systems are environmentally qualified as required in order to ensure that they will perform their intended functions. For example, components located within the containment, which are required to operate under LOCA conditions, are qualified to function under the high temperature, steam and radiation environment expected during a LOCA.

(l) Safety analysis

The safety assessment of pressure tube HWRs has traditionally followed some unique rules, some arising from the intrinsic aspects of pressure tube design and some arising from regulatory philosophy.

Although the scope of this section does not include licensing, the licensing approach does have an impact on the safety analysis. Licensing HWRs has generally followed the 'proponent propose/regulator dispose' philosophy. That is, the regulator sets overall safety requirements, and the owner proposes the best way to meet these requirements. This is in contrast to the prescriptive approach wherein a regulator sets not only the safety requirements but specifies in detail how to meet them. As a result, HWRs have been licensed under many different regulatory jurisdictions without requiring major changes to design.

The impact on safety analysis is that, apart from specifying the general classes of accident to be analysed, the regulator requires the proponent to identify the accidents to be analysed using a systematic approach. This leads not only to a relatively long list of accidents and accident combinations but also results in a very exhaustive safety review of the plant.

HWR accident analysis requires consideration not only of a set of initiating events, but also of a set of multiple failures, the predictions of which have to satisfy regulatory dose limits. Thus, the traditional 'design basis' set of initiating events is extended in HWRs, in order that protection against multiple failures is reflected in the design. Specifically, each initiating event, or process system failure, must be combined with an assumed unavailability, in turn, of each safety system which is designed to mitigate the event. The impact on design has been profound; effectively the philosophy requires that the safety functions of reactor shutdown and decay heat removal must be capable of being performed by at least two separate and independent systems. Moreover, the regulating system (which in many accidents is fully capable of reactor shutdown) is not credited with reactor shutdown in safety analysis. Thus, the shutdown function must be duplicated by means of another safety shutdown system (in early HWRs, where there was only one shutdown system, the consequences of the failure of that system in an accident had to be shown to be acceptable).

(m) Safety analysis approaches

The traditional or 'conservative' approach to safety analysis has been to combine pessimistic assumptions, data and even physical models to obtain a pessimistic analysis. This approach has been required to a greater or lesser extent by most regulators and for most reactor types. This approach was followed for the licensing analysis of HWRs, although the physical models used were realistic.

The advantage of this approach was that the answer was known to be pessimistic, and in some cases the safety analysis could be simplified by using bounding rather than realistic assumptions (as long as the results were still acceptable).

Recently, however, many jurisdictions, including the IAEA, have been encouraging the use of 'best estimate plus uncertainty' analysis. In this approach, more physically realistic models, assumptions and data are used to make the prediction more representative of expected behaviour. However, in order to ensure that there is no 'cliff edge' effect (i.e. a rapid worsening of consequence with a small change in assumptions or data), an uncertainty analysis is required, which then gives the likely range of consequences regarding the prediction. Even in a best estimate analysis, some conservative assumptions may be retained for simplicity, resulting in a 'better estimate' analysis plus uncertainties.

The approach is being changed because there are a number of disadvantages to the conservative analysis:

- The margin between the expected behaviour and the conservative predicted behaviour is unknown (how 'conservative' is the answer?).
- There is a tendency over time for an increasing number of 'conservatisms' to be added in an unsystematic fashion.
- As the conservative prediction approaches the regulatory acceptance limit, regulators become concerned at the apparent lack of margin, even though they are aware that the results are conservative.
- The predictions of the computer codes can be so severe that the physical conditions are predicted to be in areas where no validation is possible (e.g. very high fuel temperatures).

The best estimate plus uncertainties approach addresses these issues at the price of somewhat increased cost and complexity of analysis [67]. It:

- Quantifies the margin to acceptance criteria,
- Allows rational combination of uncertainties,
- Incorporates cliff edge effects,
- Highlights parameters which are important to safety,
- Focuses safety R&D on areas which really matter,
- Provides the basis for more realistic operator training.

Both approaches are described according to how they are currently applied to HWRs.

(i) Conservative analysis

Key safety analysis assumptions are chosen in a conservative direction. These include fundamental core property parameters, initial plant conditions, system performance measures, and assumptions on the unavailability of portions of mitigating systems. There is often no unique conservative choice of parameter; what is conservative in one application (e.g. minimizing the number of containment air coolers credited, in calculating peak containment pressure) may be non-conservative in another (calculating high containment pressure trip effectiveness). However, many parameters are chosen in a similar way for many accidents and Table X lists the most common choices, along with a simplified rationale.

(ii) Best estimate plus uncertainty analysis

There are three different sources of uncertainty:

- The physical models used, as expressed in the computer codes used for safety analysis;
- The plant model or idealization implemented in the codes;
- The data used for plant parameters.

In the case of HWRs, as noted earlier, the physical models have traditionally been best estimate, the conservatisms being introduced via assumptions made with regard to system behaviour or plant state.

A true best estimate analysis represents a substantial amount of work since it requires development of realistic models of behaviour; it is often less costly to use conservative models. However, the dominant conservatisms used in safety analysis are well known, as indicated in Table X; replacing them with more realistic assumptions is the first step towards achieving a better estimate.

To date, the approach employed has been to perform a best estimate analysis by removing assumptions such as those listed in Table X. Uncertainties in key parameters are then added back in, but rather than stacking independent uncertainties linearly, as is usually done in safety analysis for licensing, they are combined statistically in order to estimate the statistical uncertainty in the answer. The steps are [68]:

- Identification of the output variables which are compared with safety acceptance criteria (e.g. centre line fuel temperature);
- Identification of the key parameters in each analysis to which the output variables are sensitive (e.g. void reactivity, shutdown system delay time);

TABLE X. KEY SAFETY ANALYSIS ASSUMPTIONS

Parameter/system	Conservative condition	Rationale for being conservative
Reactor thermal power	High	Minimizes time to use up cooling water inventory, minimizes margins to critical heat flux, etc.
Reactor regulating system	Normal operation or inactive, whichever is worse; set-back is generally not credited unless it tends to blind the trip	Chosen in order to delay reactor trip
Radionuclide operating load in the HTS	Highest permissible operating iodine burden (and associated noble gases) and end of life tritium concentration	Maximizes radionuclide release from station and public dose
Steam generators	Both clean and fouled cases, whichever is worse in each accident	Reduces reactor trip effectiveness
Steam generator tube leak rate	Maximum permitted during operation, plus assessment of any consequential effects due to the accident	Increases radioactivity release
HTS flow, pressure	Low	Reduces margins to critical heat flux
Instrumented channel flow	High	Reduces low flow trip effectiveness
Coolant void reactivity coefficient:	High Low	Maximizes overpower transient Delays HTS high pressure trip
Fuel loading:	Equilibrium	Maximizes fuel temperatures, radioactivity release
	Fresh	Maximizes overpower transient
Shutdown system	Backup trip on less effective shutdown system using the last two of three instrumentation channels to trip	Delays shutdown system effectiveness

TABLE X. (cont.)

Parameter/system	Conservative condition	Rationale for being conservative
SDS2 injection nozzles	Most effective nozzle unavailable	Reduces shutdown system reactivity depth
SDS1 shut-off rods	Two most effective rods unavailable	Reduces shutdown system reactivity depth
Maximum channel and bundle power	High	Maximizes fuel and sheath temperature
Reactor decay power	High	Minimizes time to use up cooling water inventory
Initial flux tilt	High	Maximizes peak local fuel and sheath temperature
Initial moderator local maximum subcooling	Low	Minimizes margin to critical heat flux on calandria tube
Number of operating containment air coolers and other heat sinks:	Low	Maximizes containment pressure
	High	Delays high pressure trip and maximizes likelihood of hydrogen combustion
Number of dousing spray headers:	Low (typically four out of six)	Maximizes short term containment pressure
	High	Maximizes long term containment pressure and leak rate, maximizes likelihood of long term hydrogen combustion
Containment leak rate:	High (typically 2–10 times design leak rate)	Maximizes public dose
	Low	Maximizes containment pressure
Containment bypass leakage	Pre-existing steam generator tube leak	Maximizes public dose

TABLE X. (cont.)

Parameter/system	Conservative condition	Rationale for being conservative
Weather	Least dispersive weather occurring >10% of the time	Maximizes public dose
Operator actions	Not credited before 15 minutes have elapsed following a clear indication of the event, for actions that can be done from the control room; and not credited before 30 minutes, for actions that must be done in the field	Maximizes mission time of automatic systems

- — Determination of the sensitivity of the output variables to each key parameter and development of a response surface using multiple analysis code runs and multivariate curve fitting;
- — Determination of the uncertainty distribution for each key parameter;
- — Sampling of the response surface using these distributions to determine the uncertainty distribution for each output variable.

To date, the results of best estimate plus uncertainty analyses have shown much less severe consequences in accidents than have the extreme value analyses usually presented for licensing.

As noted, there are two additional sources of uncertainty beyond plant data: the physical model and the plant idealization. A full best estimate analysis should be accompanied by uncertainty assessments not just of plant parameters, but also of code models and the plant idealization. The former uncertainty can be derived from comparisons of the code to actual plant transients; the latter uncertainty is a product of the formal code validation (comparison with experiments).

5.2.2.2. Reactor shutdown

The following discussion applies to pressure tube reactors. The pressure vessel reactors Atucha 1 and Atucha 2 also have two independent shutdown systems which are functionally similar to the pressure tube reactor shutdown systems.

(a) Reactor shutdown systems

Current HWR designs incorporate two, independent, fast acting shutdown systems of diverse design. Each of these systems is separately and independently capable of safely terminating any reactivity transient from any operating state of the reactor. The reactivity transients considered include those from a large LOCA, which result in the fastest reactivity addition rate in an HWR as a result of coolant voiding in the core. The total required reactivity worth ('depth') of the shutdown system is governed by appropriate combinations of potential positive reactivity effects, which include complete voiding in the core; displacement of moderator poison; cooldown of fuel, coolant and moderator; as well as decay of xenon in the fuel after a shutdown.

The two shutdown systems performing the above functions are SDS1 and SDS2 (Fig. 85).

SDS1 (also termed the primary shutdown system) consists of mechanical shut-off rod mechanisms. Whenever a reactor trip signal is received, an electromagnetic clutch in each mechanism is de-energized, releasing a stainless steel clad cadmium absorber element, which drops into the core under gravity, initially assisted by spring thrust. SDS1 is the primary method of quickly shutting down the reactor in the event of an accident.

SDS2 (also termed the secondary shutdown system) consists of perforated horizontal tubes in the calandria through which liquid poison (gadolinium nitrate solution) is injected into the moderator. When triggered by a trip signal, fast acting valves situated between a high pressure helium tank and the poison tanks open to pressurize and inject the liquid poison into the reactor. The Indian 220 MW(e) HWRs employ a different version of SDS2, consisting of vertical empty tubes in the reactor core which, when required by a trip signal, are filled up with liquid poison (lithium pentaborate solution). India's 500 MW(e) reactors will have a similar shutdown system to CANDU units.

The Indian 220 MW(e) HWRs also employ a slow acting liquid poison injection system for long term subcriticality. The fast acting shutdown systems (SDS1 and SDS2) in these reactors have sufficient reactivity depth to handle all reactivity effects except slow ones such as xenon decay. Xenon decay is catered for by the liquid poison injection system, thus ensuring long term subcriticality. The liquid poison injection system injects poison into the moderator by pressurizing a boric acid solution contained in a poison tank with the help of high pressure gas stored in a helium tank. It is triggered automatically whenever SDS1 or SDS2 are actuated, acting after a specified time delay in the former case, and promptly in the latter.

Each of the two shutdown systems has sufficient capacity to perform its safety function, i.e. to provide the required negative reactivity rate and depth, assuming a specified number of elements (one or two shut-off rods in SDS1 or one poison

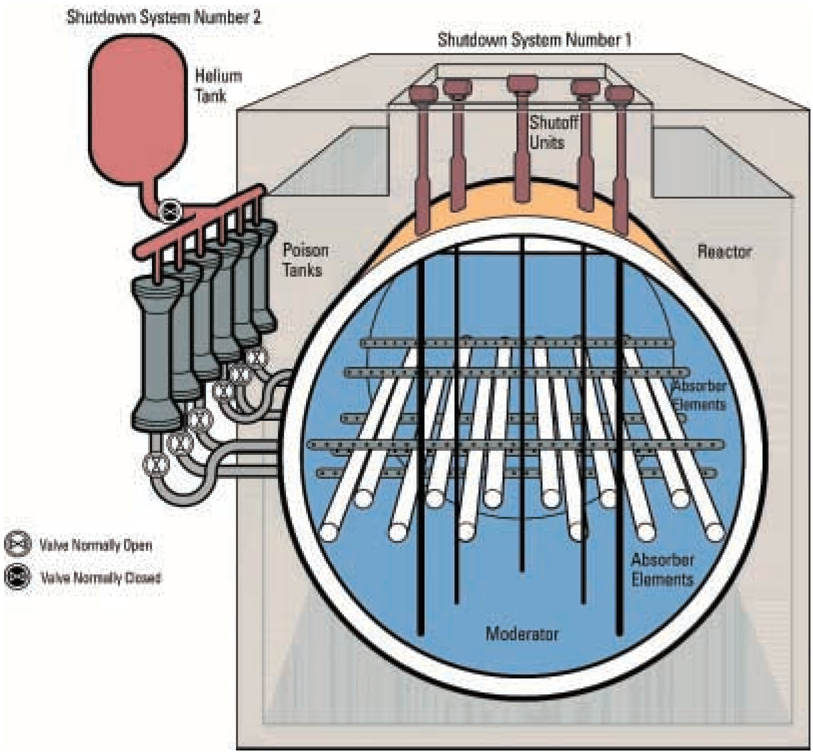


FIG. 85. Shutdown systems.

tube/bank of tubes in SDS2) are inoperable. The system actuation is fail-safe with respect to power or air failure.

(b) Trip parameters and instrumentation

Reactor shutdown is actuated by a number of trip parameters selected to ensure that at least two parameters (except where impractical or counter to the public interest) on each shutdown system are monitored to detect any serious malfunction requiring a reactor shutdown. Triplicated sensors and instrumentation channels are provided for each of the trip parameters. These sensors are independent of those used in regulation, in conformance with the philosophy of separation between regulation and protection. Also, the sensors and instrumentation for the two shutdown systems are separate. The trip parameters are plant specific, but examples include high reactor power, HTS high and low pressure, HTS high temperature, low pressure differential across the reactor core, high containment pressure, normal electric supply failure, low level in the steam generators and low flow in the HTS.

For each system, independent triplicated logic is employed, with two out of three coincidence logic used for generating the trip signal. Each trip channel is testable on-power.

(c) Computer based instrumentation for shutdown systems

Some HWR designs (Wolsong 2, 3 and 4; Qinshan; Darlington) incorporate computer based trip logic for both SDS1 and SDS2. This represents an evolution of the programmable digital comparators used on earlier CANDUs to implement the trip decision logic. Computers enhance safety availability (converting many unsafe failures into safe ones) and improve production reliability. Each channel has its own trip computer, which replaces all analogue trip comparators, all programmable digital comparators and relay logic. Each channel also has a separate test computer for automation of safety system testing. Final, two out of three voting logic is done by relays.

While the possibility of errors is not unique to software based logic, as opposed to relay based logic, in practice the former has developed much more formal and extensive validation and verification, especially when the software is used in reactor safety systems. Standards have been written to codify accepted practices [69, 70]. IAEA activity in this area has been under way for some time, although focused more on the use of software in control and instrumentation rather than in safety systems [71].

(d) Reactor set-back and step-back

The set-back provision reduces reactor power by driving the demand power, in a ramp fashion and at a rate, typically, of 0.5% full power/s, down to a level determined by the parameter causing the set-back. The step-back feature (not provided in smaller Indian 220 MW(e) HWRs) produces a step reduction in power by dropping the mechanical control absorbers (effected via clutch disengagement).

Reactor set-back and step-back act through the reactor regulating system to effect a fast reduction in reactor power in situations which are abnormal but not severe enough to warrant immediate safety system shutdown. The step-back initiation logic in particular is separated from the main control programme to some extent, in order that faults in the regulating system are less likely to also affect step-back.

Many of the less severe reactivity and process transients can be handled by set-back and step-back, thus reducing, as intended, the demand on the reactor shutdown systems.

5.2.2.3. Core cooling

Normally, the core decay heat can be removed through two, alternative, independent and diverse paths: through steam generators, with heat rejected by boiling off feedwater, or through the shutdown cooling system, with heat rejected to process/service water, which ultimately rejects heat into the atmosphere (through cooling towers) or into cooling water from the ultimate heat sink (river, lake, sea).

To deal with certain accident conditions, other paths are also provided. During a LOCA, the ECCS is used to re-flood the core and also to remove at least part of the reactor heat (the remaining part being removed by steam generators). The heat picked up by ECCS water is rejected to the process water in the ECCS heat exchangers. In the case of a LOCA coincident with impairments occurring in the ECCS, an additional heat sink (unique to HWRs) is the cold moderator water, which picks up heat from overheated coolant channels and transfers it to the process water in the moderator heat exchangers. Beyond this, in the case of severe core damage accidents involving subsequent failures of moderator circulation or cooling, the massive water inventory in the calandria vault (the shield tank) would act as an additional heat sink for some hours.

In Atucha 1 and 2, the moderator contributes to core heat removal for LOCA plus LOECC, although in a different manner, as a result of the direct connection between the moderator and the HTS.

Alternative provisions for core cooling during conditions with the HTS intact are discussed in the following sections. Severe accidents are discussed in Section 5.3.2, while the ECCS is described in Section 5.2.2.4.

(a) Steam generators and feedwater system

The normal heat sink for the fuel in the core is provided by forced circulation of HTS coolant, which transfers the core heat to the secondary coolant in the steam generators, with feedwater supplied to the steam generators by steam generator feedwater pumps.

The steam flow from the steam generators, the normal path of which is through the turbine and condenser, can also be directed under transient conditions through one or more of:

— Condenser steam discharge valves (CSDVs) to the condenser directly (provided an adequate vacuum is available in the condenser),
— Atmospheric steam discharge valves (ASDVs) to the atmosphere,
— Main steam safety valves (MSSVs) to the atmosphere.

Typically, the condenser can accept 70% full power steady state steam from the CSDVs; in such plants there is no need for high capacity ASDVs for control purposes, and those provided normally pass ~10% of the steam flow. Instead, for safety purposes, the MSSVs are provided for overpressure protection of the secondary side and for crash cooldown during a LOCA; typically they pass ~115% of the steam flow.

Plants with high capacity ASDVs can likewise pass in excess of 100% of the steam flow for both control and safety purposes.

These provisions allow the steam generators to serve as a heat sink for the core under certain transient/off normal conditions when the turbine is not able to take all the power generated by the reactor.

Under reactor shutdown conditions, cooling of the core in order to remove decay heat can be performed by natural circulation of the HTS coolant, i.e. forced circulation is no longer necessary. On the secondary side, feed flow to the steam generators, at a substantially reduced rate (about 4% of normal feed flow), is provided by auxiliary steam generator feed pumps (ASGFPs). The power source for these pumps depends on the station design; some are electric, powered by Class III (diesel generator backed), some stations use steam driven ASGFPs and some use direct diesel drive. The pipes from the ASGFPs in some HWRs are routed in such a way that auxiliary feedwater can be supplied directly to the steam generators independently of the main feed lines. Thus, in the event of failure of the main feedwater supply (owing to failure of main feedwater pumps or normal electric supply, or a break in the feedwater line), the reactor would be tripped (on low steam generator level or high HTS pressure) and decay heat removal would be ensured by the auxiliary feed flow which starts automatically.

In all HWRs, the steam generators normally contain an inventory of water sufficient for removing decay heat for several tens of minutes. In view of this, a few minutes delay in restoring the auxiliary feed flow would be tolerable.

The de-aerator, from which the main and auxiliary steam generator feed pumps draw their supply, provides a large inventory to maintain the reactor in a hot shutdown condition or to bring it to a cold shutdown condition. Two full capacity condensate extraction pumps and two auxiliary condensate extraction pumps (the latter on Class III power supply) are employed to provide a reliable supply of water to the de-aerator.

To cater for an extreme situation involving total loss of feedwater, HWRs have a backup water supply sourced either from the EWS, or from the firewater system. These sources can be added directly to the steam generators. However, on current

HWRs they represent a low pressure supply and can enter the steam generators only after the latter have been depressurized by blowing down steam through the ASDVs, CSDVs, or MSSVs.

(b) Shutdown cooling system

An alternative heat sink for removing decay heat is the shutdown cooling system, which is normally used for cooldown and during cold shutdown conditions. The system circulates HTS water (by means of shutdown cooling pumps, powered by Class III electric supply) through heat exchangers, which transfer heat to the process water system. Although two shutdown cooling trains are provided, one on either side of the reactor, one train is adequate for removing the heat. The shutdown cooling system is normally intended to be 'valved in' once the HTS coolant temperature has been brought down to around 150–180°C. However, under emergency conditions, it is possible to initiate this system even at full system temperature and pressure. Thus, steam generator blowdown is not required.

5.2.2.4. ECCS

The following description summarizes the functional capabilities of the ECCS for HWRs; the specific details apply to pressure tube HWRs.

(a) Common features

In common with all water cooled reactors, HWRs incorporate an ECCS to cool the core and thereby limit fuel damage in the event of a LOCA. The minimum design objective is to limit the release of fission products from the fuel. Although specific acceptance criteria may differ from country to country, typical requirements in this regard are as follows:

- In LOCAs with break sizes smaller than, and up to, the largest feeder break, there shall be adequate cooling of the core to prevent gross fuel sheath failures. However, in single channel events, fuel failure in the affected channel may not be prevented.
- In LOCAs larger than feeder pipe breaks, fuel failures shall be limited such that the radiological consequences to the public are within limits acceptable to the regulator for this class of event.
- In all LOCAs, the integrity of fuel channels (excluding the affected channel in single channel events) shall be maintained, and fuel geometry shall allow continued cooling of the core by the ECCS.

— Adequate, long term cooling capability of the fuel following the LOCA shall be ensured.

ECCSs provided in various HWRs differ in design details. Usually, the system is designed to inject water into the HTS in three stages: high pressure injection, followed by intermediate/low pressure injection, followed by long term recovery and recirculation.

High pressure injection is supplied either by a system of accumulators containing water and pressurized by nitrogen gas tanks or, in the case of some OH plants, by high pressure pumps. Intermediate pressure injection and recirculation is provided by pumps (powered by Class III electric supply); the water being drawn initially from a tank, and subsequently from a sump (which collects spilled water from the break) via the ECCS heat exchanger.

In the newer HWRs, and in several of the original CANDU 6s, the entire sequence is now automated; in other, older HWRs, operator action is required to switch from intermediate pressure injection to recovery mode.

Emergency core cooling is accompanied by a 'crash cooldown' of the steam generators, which is achieved by venting steam into the atmosphere through the MSSVs, or their equivalent. This ensures that the HTS pressure stays below the ECCS injection pressure, especially for small LOCA, and also for large LOCA in the long term. This approach is preferred to depressurization of the HTS directly via relief valves, since it reduces the possibility of spurious opening of the HTS relief valves. In addition, in HWRs, secondary side cooldown will result in marginal or no reactivity increase. In fact, in HWRs where boiling in the channels occurs, the reactivity actually decreases.

Very small breaks (e.g. instrument line break) within the capacity of the D_2O feed pumps can be handled without actuation of the ECCS. The spilled D_2O is channelled through appropriately located drain lines to a tank, from where it is pumped back to the HTS storage tank after having first been cooled in a heat exchanger.

The above description is general; comments on some specific designs follow.

(b) ECCS in single unit CANDUs

Single unit CANDUs use high pressure H_2O accumulators, followed by medium pressure pumps drawing water from the elevated dousing tank (Fig. 86). As the dousing tank nears depletion, the mode is changed to recirculation: water collected in the basement is injected back to the HTS, having passed through a heat exchanger.

In the case of small breaks, most decay heat is transferred to the steam generators and rejected via the MSSVs. In large breaks, the break flow itself acts as the major heat sink.

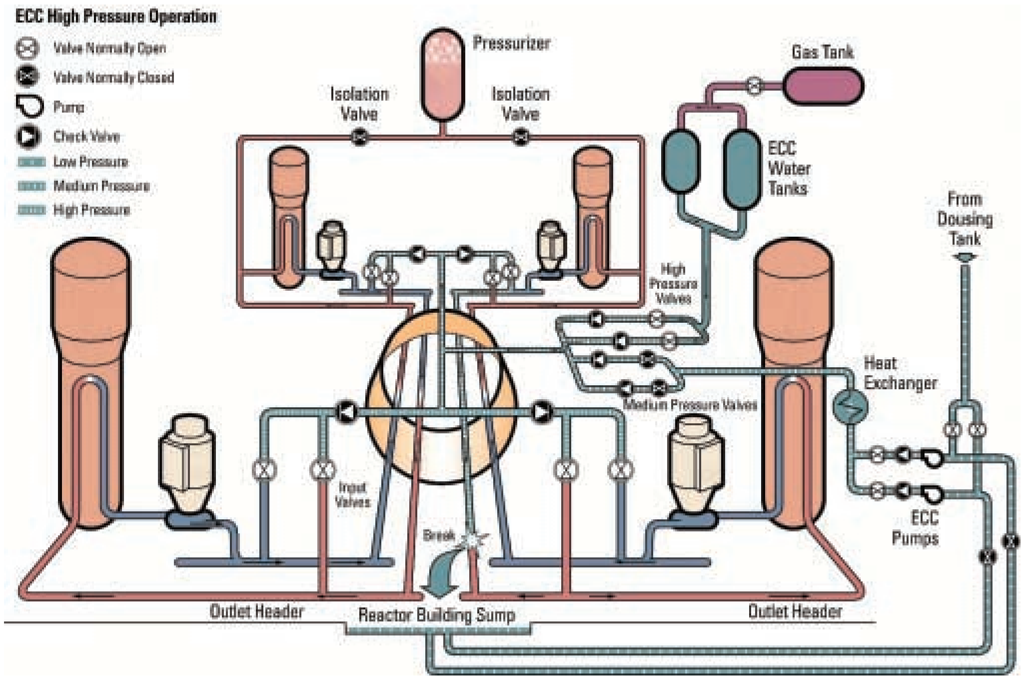


FIG. 86. ECCS of a CANDU 6.

ECCS injection takes place in all the headers in the HTS, irrespective of the location or the size of the break.

In the CANDU 9 design, the ECCS water tanks are sited within the reactor building in order to reduce the likelihood of an interfacing system LOCA occurring outside containment. In addition, the medium pressure emergency core cooling has been eliminated: a portion of the water from the elevated reserve water tank is discharged onto the floor at ECCS initiation and provides water for the ECCS recovery pumps after the accumulator phase has ended.

(c) ECCS in Indian 500 MW(e) HWRs

The ECCS of the Indian 500 MW(e) HWRs is similar except that subsequent to the high pressure injection phase, low pressure injection and recirculation flow is supplied through four 50% pumps from the reactor building suppression pool, which also collects the spilled water. All actions up to and including the establishment of long term recirculation are automatic.

(d) ECCS in Indian 220 MW(e) HWRs

In the Indian 220 MW(e) HWRs (Fig. 87), the ECCS incorporates:

— High pressure heavy water injection,
— Intermediate pressure light water injection,
— Low pressure long term recirculation.

The high pressure heavy water injection is provided by a system of accumulators containing D_2O pressurized by a nitrogen gas tank. Intermediate pressure light water injection is provided by a system of two accumulators and a pressurized nitrogen gas tank. When the HTS pressure falls further, low pressure light water injection occurs, followed by recirculation provided by pumps. The ECCS pumps draw water from the suppression pool and this water is cooled in the ECCS heat exchanger (by active high pressure process water).

The provision of a heavy water accumulator for the initial high pressure injection permits its use during certain non-LOCA transients to make up for fast shrinkage in the HTS, but without downgrading the HTS heavy water.

Injection of ECCS water takes place through two of the four headers, which are selected on the basis of the size and location of the break. In the case of small breaks, or breaks on the outlet header side, in which the flows continue in the normal direction, injection takes place into the reactor inlet headers. In large breaks on the inlet header side, which result in reversal of flow in half of the core, injection is directed to the inlet and outlet headers on the side away from the break so as to assist

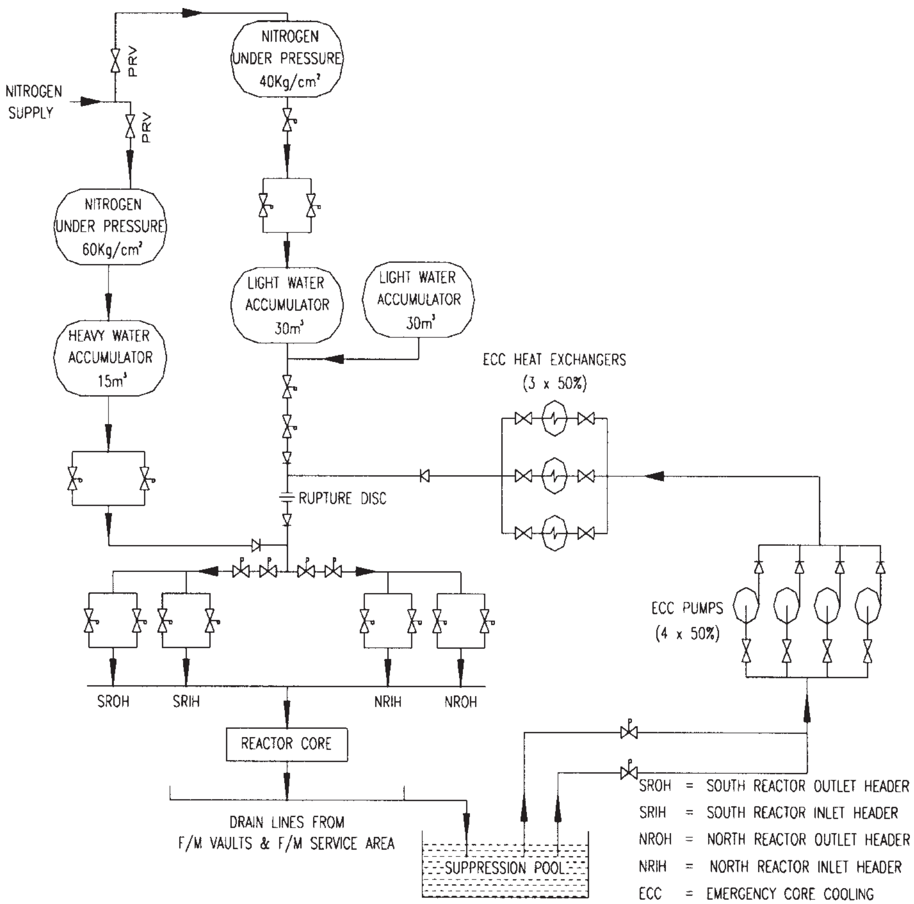


FIG. 87. ECCS of Indian 220 MW(e) HWRs.

the flow direction. Selection of the headers takes place automatically and is based on a pressure difference signal between the headers, indicating the flow direction.

5.2.2.5. Containment system

As with the ECCS, the containment systems for pressure tube HWRs and for pressure vessel HWRs are functionally similar. The actual containment design for pressure vessel HWRs is very similar to the standard Siemens PWR containment. The details in the following description are based on pressure tube HWR containment.

(a) General

The chief function of the containment system is to confine the release of radioactivity into the environment during accidents, to within acceptable limits. The containment system consists of a leaktight envelope around the reactor and associated nuclear systems, and includes a containment isolation system (for the fast closure of valves/dampers in lines penetrating the containment), containment atmosphere energy removal (cooldown) systems and cleanup systems. Hydrogen control is provided in the newer, larger HWRs in order to cater for the long term buildup of hydrogen resulting from radiolysis after a LOCA, and for severe accidents such as LOCA plus LOECC.

Current pressure tube HWR designs employ the following major types of containment:

— Single unit containment employing a dousing system for pressure suppression (as used in CANDU 6);
— Multiple unit containment incorporating a common vacuum building (as used in the Pickering, Darlington and Bruce stations);
— Double containment system incorporating a double envelope and employing pressure suppression (as used in Indian HWRs).

These concepts are described briefly in the following sections. It should be noted that the HWR design does not determine the type of containment; that decision is driven by other factors such as single as opposed to multiple unit philosophy, national regulatory requirements, allowable leak rates, construction cost, etc.

(b) Single unit CANDU containment

The CANDU 6 single unit containment consists of a cylindrical, prestressed, post-tensioned concrete building with a concrete dome (Fig. 88). The concrete

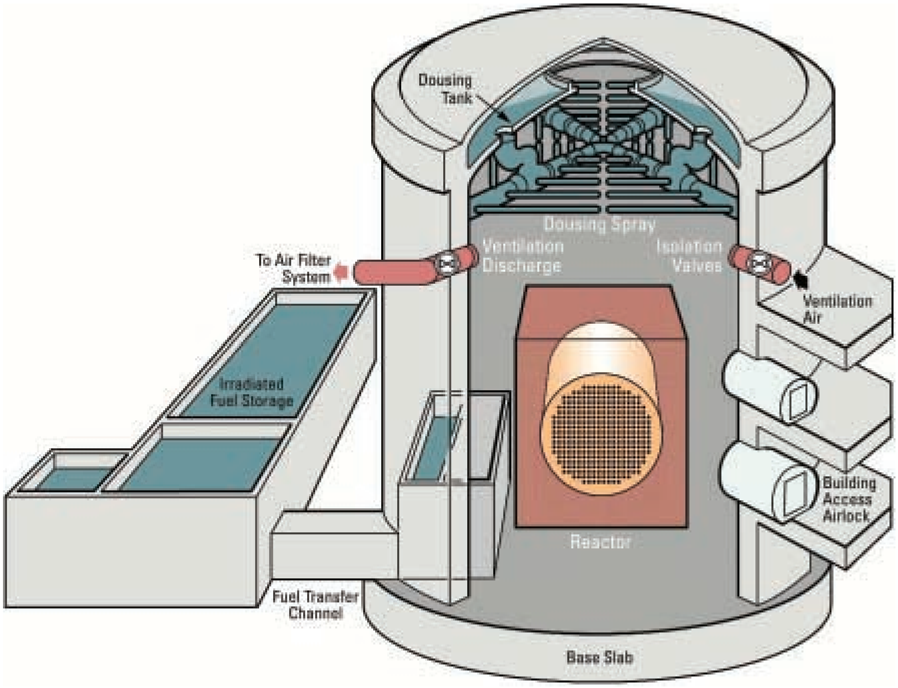


FIG. 88. Single unit containment.

provides strength and shielding; the building is lined with an epoxy coating to improve leaktightness. Beneath the outer dome is an inner one having an opening in the crown. The double dome, together with the perimeter wall, form a container, providing storage at an elevated level for 2550 m^3 of water for dousing and emergency core cooling.

In the event of a rise in pressure or a release of radioactivity to the containment, the containment isolation system would close all penetrations open to the outside atmosphere, mainly the containment ventilation system. This is a subsystem of the containment safety system, and therefore is designed to safety system standards, with triplicated testable logic and an availability target of greater than 0.999 (so that the overall containment system can meet its availability target of 0.999).

A sufficiently large pressure rise (e.g. from a LOCA or steam main break) would trigger the dousing spray system through valves in the dousing spray headers. The purpose of the dousing spray is to suppress the short term pressure rise caused by the accident and thus the flow rate is very high. The dousing spray turns on when the pressure rises above 14 kPa, and turns off when it falls below 7 kPa, resulting in a cyclic operation for small LOCAs. The dousing valves are divided into two sets of six valves (three pairs of two valves in series) each. One set is pneumatically/electrically operated and the other is electric. Operation of any two pairs of valves in any one set is sufficient to keep the pressure following a LOCA to below the containment design pressure.

Long term pressure control and heat removal, after the dousing water is finished, is achieved through local air coolers.

In recent CANDU 6 plants (Wolsong 2, 3 and 4, and Qinshan 1 and 2), a network of hydrogen igniters is provided to burn any local concentrations of hydrogen formed in the long term post-LOCA, and in dual failures (LOCA plus LOECC), before explosive conditions are created.

A filtered air discharge system allows containment depressurization and cleanup in the very long term after an accident.

In the CANDU 9 containment, the dousing system has been eliminated in order to achieve greater plant simplicity. As a consequence, this means that the design pressure is considerably higher than for CANDU 6. To achieve increased leaktightness at the higher design pressure, the building is lined with a steel liner. Also, in addition to hydrogen igniters, passive hydrogen recombiners are used in selected locations to provide increased reliability in the long term post-accident.

The containment design pressure is set by the LOCA, combined with whichever containment impairment maximizes the internal containment pressure, e.g. impairment of air coolers or LOECC. A steam line break inside the containment may reach a higher pressure but does not determine the containment design pressure as there is minimal radioactivity release from a steam line break (the reactivity effect is negative, especially for those HWRs where boiling in the HTS occurs): the requirement is that the containment structural integrity be preserved (no through-wall cracking).

(c) Multiple unit vacuum building containment system

In the multiunit vacuum system, four or eight reactors, each with its own local containment, are connected by large scale ducting to a separate, common vacuum building kept, as its name implies, at near zero absolute pressure (Fig. 89). Should steam be released from a pipe break in the reactor building, the pressure causes banks of self-actuating valves connecting the vacuum building to the ducting to open. The steam and any radioactivity are then drawn along the duct by suction; the steam being

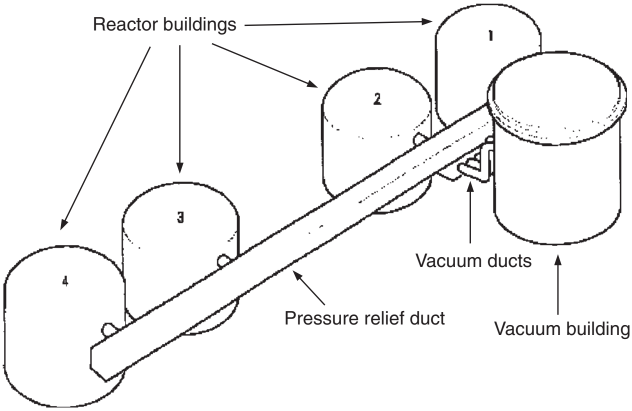


FIG. 89. Multiunit containment.

condensed by dousing in the vacuum building and the soluble fission products such as iodine washed out. The dousing is passively actuated by the difference in pressure between the main body of the vacuum building and the vacuum chamber; it does not require electrical power or compressed air supplies for its operation.

This concept, which was developed as a result of the economic benefits of multiunit sites, has a number of unique safety characteristics:

— After an accident, the entire containment system pressure is subatmospheric for several days; thus leakage is inward, rather than outward. This is also the case (for a shorter duration) even with an impairment to the containment envelope.
— The overpressure period in the reactor building is very short, of the order of a couple of minutes, and therefore the design pressure is reduced and the design leak rate can be increased relative to single unit containment.
— Even if the vacuum building is not available, the large interconnected volume of the four or eight reactor buildings provides an effective containment.
— Several days after an accident, when the vacuum is gradually depleted and the containment pressure rises towards atmospheric, an emergency filtered air discharge system is used to control the pressure and ensure that the leakage is filtered.

(d) Containment system of Indian HWRs

Current Indian HWRs use a double containment principle (Fig. 90) [72]. The annular space between the primary and secondary containment envelopes is provided with a purging arrangement to maintain a negative pressure in the space. This arrangement significantly reduces ground level releases into the environment during accidents in which there are radioactivity releases into the primary containment.

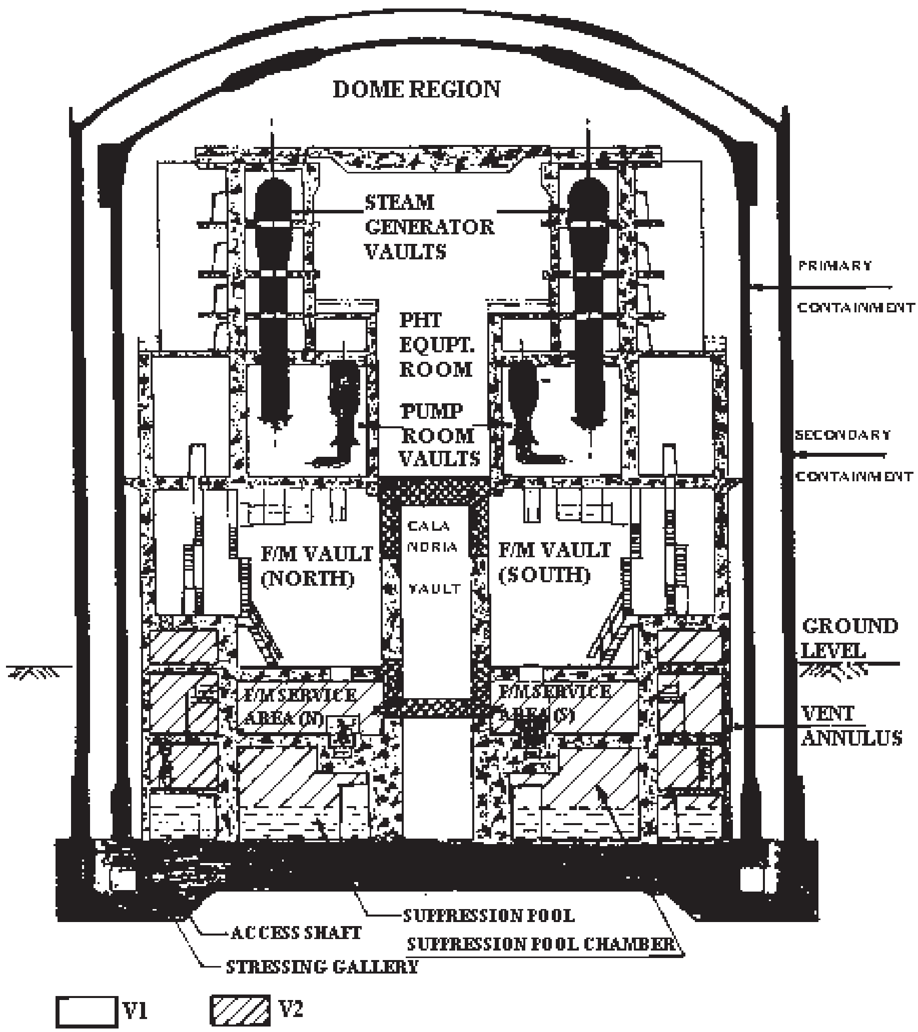


FIG. 90. Double containment employed in Indian HWRs.

The containment structures are of concrete. The primary containment is a prestressed concrete structure, consisting of a perimeter wall topped by a prestressed concrete dome. The outer, or secondary containment envelope is a reinforced, cylindrical concrete wall topped by a reinforced concrete dome. The primary containment uses an epoxy coating to form a liner for added leaktightness and ease of decontamination.

Automatic isolation of the containment is initiated in the event of a pressure rise and/or activity buildup in the containment.

A pressure suppression system incorporating a suppression pool is used for limiting the peak pressure in the containment following a LOCA or a main steam line break (MSLB). The primary containment building is divided into two accident based volumes, V1 (dry well) and V2 (wet well), separated by leaktight walls and floors and connected through a vent system via a suppression pool located in the sub-basement (see Fig. 90). During a LOCA or MSLB, the pressure rise in V1 will cause a steam–air mixture to flow via the vent system to the suppression pool, where the steam will condense and the air escape into V2.

The pressure suppression system is an entirely passive system and does not perform any function during normal operational states. In addition to pressure suppression, the suppression pool water forms part of the long term recirculation mode of emergency core cooling.

To cool down and thereby depressurize the containment following an accident in as short a time as possible, a system of building air coolers is used, distributed at various locations in V1 of the containment building. These coolers are used during normal operation as well. The coolers are supplied from an assured process water supply and their fan motors driven by power supplies backed by diesel generators (Class III electric power supply). As with all other containment related engineered safety features, these coolers are designed to work in post-LOCA containment environment conditions.

In order to achieve further depressurization at low pressures (e.g. below 0.04 kg/cm^2(g)), which may be difficult to achieve by cooling alone, there is a provision for controlled gas discharge to the stack via filters. This system can also be used up to a maximum containment overpressure of 0.4 kg/cm^2(g) for delayed containment venting if warranted by the accumulation of non-condensables during the post-accident phase. Usually, operation of this system is not envisaged in the 48 hours following an accident.

Two systems are used for post-accident cleanup of the atmosphere in the containment:

— *Primary containment filtration and pump back system*: In this system, in order to perform a long term containment atmosphere cleanup operation after an accident, air flow is recirculated within the primary containment through charcoal filters. Significant reduction in the concentration of iodine in the primary containment would be effected over a period of time, so that by the time the controlled gas discharge system is operated (e.g. after 48 h), the associated stack releases would be low.

— *Secondary containment filtration, recirculation and purge system*: This system provides multipass filtration and mixing by recirculation within the secondary containment space, and also maintains negative pressure within it. The negative pressure maintained in the secondary containment space brings the net ground level release down to very low values.

The loads for which the containment structure is designed include normal and construction loads, abnormal loads (including pressure and temperature loads resulting from the design basis LOCA or MSLB) and extreme environmental loads (wind and earthquake). The seismic design is based on a dynamic analysis of the SSE. Combinations of a LOCA and an SSE occurring simultaneously are considered.

5.2.2.6. Station service electrical supply system

The electrical power supply systems for pressure tube and pressure vessel HWRs follow a similar functional approach. Multiple unit pressure tube HWRs, while also functionally similar, have very reliable electrical power as a result of inter-unit ties. The examples in this section refer to single unit pressure tube HWRs.

The station service electrical supply system provides power to all station loads, including safety related loads. The system is designed to have adequate stand-by power sources such that in the event of loss of normal supplies, the essential equipment required for reactor safety can be kept running. Essentially, the power sources employed for satisfying station requirements are:

- Normal supplies provided from two redundant sources, i.e. from the grid (off-site power) and/or from the unit generator;
- Stand-by diesel generators;
- Storage batteries.

The station electrical supply system is divided into four classes depending on the reactor safety importance of the connected loads.

Class IV is the normal AC station service supply and is derived from the unit generator (through the unit service transformer) or from the grid (through the startup transformer/system service transformer). Normally, all loads are shared equally between the two sources. Loads that can tolerate power supply failure without jeopardizing reactor safety are connected to Class IV buses. Class IV is also the normal source for other classes of power. Complete loss of Class IV power will result in a reactor shutdown.

Class III buses receive their supply from Class IV and in the event of its failure, from diesel generators. Redundant, 100% capacity diesel generators capable of starting and taking full load within a couple of minutes are provided for this purpose. Loads which are required for reactor safety but which can tolerate a short interruption are connected to this class (e.g. pumps for auxiliary steam generator feedwater, emergency process water, shutdown cooling, moderator circulation, D_2O feed to the HTS, ECCS, instrument air compressors).

There have been two approaches to grouping and separating the diesel generators. In the Group 1/Group 2 approach, two independent sets of diesel generators are provided, one set in each group. Within each group, there are typically two diesel generators, connected to the 'odd' or 'even' Class III bus respectively. The two groups are physically separated, as discussed in Section 5.2.2.1.(b); individual diesel generators within a group may not be separated.

In the swing diesel approach, there are three diesel generators. One diesel generator is connected to each of the two Class III buses, with a third diesel generator (where provided) serving as auto stand-by for both buses. Physical and functional separation is provided between the diesel generators.

Redundant cables to safety related electrical equipment are connected to separate buses and are led through separate routes to the equipment in order to avoid common cause failure (CCF).

Class II is an uninterruptable AC supply, feeding the most essential loads that cannot sustain any interruption (e.g. fuelling machine supply pump, various valve actuators). Class II normally draws power from the Class III bus and, if this is unavailable, from storage batteries. For this purpose, supplies from Class II come via rectifiers and inverters.

Class I is an uninterruptable DC source, feeding such users as safety related control circuits. It is normally powered from Class III buses through rectifiers and, in the absence of Class III, from storage batteries. All DC control power supply requirements are met from Class I buses.

All the actions required to bring in the stand-by power sources, including load shedding (if necessary) and restoration of power supply to safety related loads, can be achieved automatically.

Those portions of the electrical power supply that are needed to ensure safe shutdown after an earthquake are seismically qualified. Some recent HWRs provide a physically and electrically separated backup power supply for this purpose or for a similar common mode event (e.g. tornado, tsunami). This is called the emergency power supply. It is a Group 2 electrical system typically consisting of two or more seismically qualified diesel generators, remote from the Class III Group 1 diesel generators, which supply Class III Group 2 power to essential safety loads. Similar to Group 1, Group 2 emergency power supply also has a smaller set of batteries which supply qualified DC power (Class I) or AC power (Class II) to control loads. The cable routing follows the usual separation rules between Group 1 and Group 2 up to the end user, in order to reduce the possibility of CCF.

5.2.2.7. Supplementary control centre

In addition to the main control room, a supplementary control centre (also called a secondary control area or emergency control room) is also provided and located away from the main control room. This room is equipped with enough instrumentation to shut the plant down, and to maintain it in a safe shutdown condition and monitor its safety status in the eventuality that the main control room becomes inaccessible for any reason. Generally, the secondary control area controls Group 2 equipment and is qualified for external events such as earthquakes.

5.2.3. Provisions for periodic inspection

Periodic inspection is required “to provide an assurance that the likelihood of a failure that could endanger health and safety has not increased significantly since the plant was put into service”, i.e. to ensure that an unacceptable degradation in quality is not occurring [73]. The scope includes the fluid boundary portions of components, piping and supports that comprise systems:

- Containing fluid that directly transports heat from nuclear fuel,
- Essential for the safe shutdown of the reactor and/or the safe cooling of the fuel in the event of an accident,
- That could jeopardize any of the above systems.

The scope and extent of inspection are chosen commensurate with the likelihood and consequences of failure. Similarly, the frequency of inspection is determined on a logical basis, not by a fixed time interval, and is set by the severity of operating conditions, evidence from earlier inspections, and consideration of any known abnormalities in operation.

Design and manufacturing inspections are intended to provide an adequately defect free product at startup. Since the purpose of periodic inspection is to detect general degradation and not to certify every component as defect free, it is based on a sampling approach, with the samples chosen to include areas subject to the most extreme conditions. To provide a base line, an inaugural inspection is made before startup, of all areas of the plant intended to receive periodic inspection. This is a diagnostic inspection, not a ‘go/no go’ test.

The periodic inspection programme affects the design in that designers must ensure that components which require periodic inspection are easily accessible. Major components in this regard are:

- Pressure tubes,
- Steam generator tubes,
- HTS piping (including headers and feeders),
- D_2O/H_2O heat exchangers.

In the case of pressure tubes, specialized techniques and methods are used to test for different degradation mechanisms:

- Corrosion and hydrogen ingress are monitored by removing microsamples from a representative number of tubes and performing chemical analysis to determine deuterium pick-up since reactor startup. Corrosion processes can also be monitored by measuring the oxide thickness on the inner pressure tube surface using an eddy current method.
- Creep and growth are determined from ultrasonic and/or mechanical measurement of tube diameter and wall thickness. Pressure tube sag (deflection) can be calculated from the measurement of either curvature or inclination.
- Inspections for confirmation of structural integrity include determination of garter spring location using an eddy current technique as well as a calandria tube–pressure tube gap measurement. The latter uses a combination of eddy current and ultrasonic methods.
- High sensitivity volumetric inspection for flaws is done using both ultrasonic and eddy current methods. If found, flaws are characterized using ultrasonic, eddy current and profilometry techniques. These provide details about flaw length, depth, orientation and root tip radius. Such detailed information is essential for flaw disposition and ‘fitness for service’ assessments.
- Material property changes due to long term exposure to stress, elevated temperature and neutron irradiation are determined by periodically removing a pressure tube from a ‘lead unit’. Tests on the removed material include fracture toughness and delayed hydride cracking velocity. The results are used to confirm that material property changes in power reactors conform to predictions based on accelerated laboratory tests and research reactor irradiations, and to show that the leak before break criterion is still met.

5.3. BEHAVIOUR OF CURRENT HWRs IN POSTULATED ACCIDENTS

The purpose of this section is to demonstrate how the inherent and engineered design characteristics of HWRs affect the response to postulated accidents.

5.3.1. Design basis accidents

The basic purposes of safety analysis are to assist in the design of safety related systems, and then to confirm that the radiation dose limits are met. As such, safety analysis requires predictions of the consequences of hypothetical accidents.

In most cases, the approach to HWR licensing has been results driven rather than prescriptive. That is, the regulator sets overall requirements on the classes of accidents to be considered, and on the public dose limits as a function of accident class, but leaves it up to the owner to a large extent to determine how best to meet these limits. In particular, most HWR regulators do not specify the design requirements in great detail, nor do they specify prescriptive assumptions on accident analysis methods. This approach is not intrinsic to HWRs but is nevertheless followed by most HWR owners and regulators.

Thus, in 'conservative' accident analysis, HWR practice has been to use physically realistic models of system behaviour, incorporating conservatisms into assumptions on input parameters, operator response, etc., rather than in the physical models. This requires relatively detailed models of:

- Reactor physics;
- Thermohydraulics of the HTS;
- Thermomechanical behaviour of the fuel and the fuel channels;
- Containment response;
- Fission product transport, chemistry and release;
- Atmospheric dispersion;
- Radiation exposure and dose.

As noted, HWRs include multiple failures in the 'design basis'. In HWRs up to Darlington, the 'single–dual' failure philosophy was used. A single failure (not to be confused with 'single failure criterion', which is a design requirement on safety related systems) is the assumed failure of a process *system*, that is, any system required for power production. A dual failure is defined as the failure of a process system coinciding with the unavailability of a safety system or subsystem. Safety analysis is therefore performed for the failure of each process system in the plant; then for each such failure combined with the unavailability or impairment of each relevant safety system or subsystem in turn. The major impairments are:

- Unavailability of one of the two shutdown systems (this is always assumed regardless of other impairments).
- No emergency coolant injection.
- No ‘crash’ cooldown of the steam generators.
- No HTS loop isolation (for plants with two loops).
- No containment isolation.
- Failure of dousing.
- Deflated airlock door seals.
- Failure of vault coolers.
- In the case of vacuum containment: partial or total loss of vacuum, failure of the instrumented containment pressure relief valves to open or close and failure of one bank of self-actuating containment pressure relief valves.

In addition, mitigating process system actions are not normally credited in demonstrating the effectiveness of the safety systems. In this section, dual failures are treated in the discussion of severe accidents, in order to be consistent with practice for other reactors.

Failures in safety support systems (such as instrument air) are addressed in the PSA.

The description and numbers that follow are specific to CANDU 6, but the general observations and qualitative scenarios are common to all HWRs of similar design unless noted otherwise.

5.3.1.1. LOCA

(a) Introduction

A LOCA constitutes the most severe challenge to all three safety systems (shutdown, emergency core cooling and containment) and sets many of their design requirements.

In the case of CANDU 6, a large LOCA is defined as a pipe break the size of which lies beyond the range of breaks in feeder pipes (above 100 cm^2). A small LOCA is defined as a pipe break within the 2–100 cm^2 size range, which includes the feeder pipe range. A very small LOCA is defined as having a break size of less than 2 cm^2 and one which can be handled by the heavy water make-up system.

A special case of a small LOCA is an in-core break, which introduces additional phenomena such as interaction of the broken channel with neighbouring channels and the reactivity mechanisms. A special case of a large break is an ‘interfacing’ LOCA occurring between the HTS and the ECCS piping.

In Indian HWRs, break sizes that result in overpower and early neutronic trip due to void reactivity are categorized as large breaks [74]. In the case of the 220

MW(e) HWR, a break of 160 cm^2 (10% of the largest header break) or above, would fall into this category. A very small LOCA (leak) would be 1 cm^2 or less.

With both large and small LOCAs, one of the shutdown systems, the emergency core cooling and containment are all required and are initiated either automatically or, at the lower end of the range, manually by the operator. In the case of very small LOCAs, shutdown may be manual or automatic; emergency core cooling and containment are not required and may not be initiated automatically.

(b) Defining the accident

The accident is defined by the size and location of the break, i.e. by the particular pipe which is assumed to break. The pressure tube design means that there are hundreds of small pipes (less than 8 cm diameter) which connect the ends of each channel to the headers above the core. The headers and associated piping around the pumps and steam generators are large, typically 25–50 cm diameter. Steam generator tubing itself is very small in size, around 1 cm in diameter.

In the case of a steam generator tube failure (which equates to a small leak), it is first necessary to identify the accident by detecting the leakage through the use of the D_2O in water detection system. Once the leak is detected, the operator will shut down the reactor, depressurize the HTS and then drain it to below the level of the leaking tube so that it can be repaired. In order to effect this, the shutdown cooling system is used. Normally, the initial cooldown is achieved via the steam generators. However, the shutdown cooling system can remove decay heat at full system pressure, and therefore the operator is not dependent on the secondary side for depressurization/cooldown. Since the D_2O make-up system has the capacity to handle a single steam generator tube failure, fuel cooling is not at risk.

As a result of the total length of feeders and pressure tubes, the chance of a small break occurring is about 100 times more probable than that of a large break. Clearly this range is analysed for both economic and safety reasons.

The small break analysis assumes a break in a pipe of a size up to twice the cross-sectional area of the feeder pipe. In-core breaks (pressure tube breaks) are included, alternately assuming that the calandria tube remains intact and that it fails.

(c) Phenomenology of LOCAs

Depending on national practice and electrical grid reliability, the HWR design basis may assume the availability of Class IV power. Thus, the main heat transport pumps are left running during the blowdown and refill phases of the LOCA and provide an important flow bias.

(i) Small breaks

In the case of small breaks, the requirement on the ECCS is to limit or prevent fuel damage, mainly for economic reasons. In the case of breaks having an area equivalent to the severance of several feeder pipes, the pumps are much more influential in determining flow than the break itself. Thus, the flow is always forward or recirculating, although as steam quality builds up with time, the pump head decreases and the resistance of the circuit increases; therefore, the magnitude of the flow falls with time.

The first requirement is to shut down the reactor before fuel sheaths experience prolonged dry out at high power. Prevention of dry out is sufficient, but not necessary, to prevent fuel sheath failure. Shutdown is triggered by receipt of process parameter signals such as low storage tank level, low flow, low pressure, low core pressure drop or high pressure within the containment. With a feeder size break, shutdown is initiated within the first three or four minutes and is normally set to prevent dry out. As the circuit continues to empty after shutdown, the flow in the headers or channels eventually falls to a low enough level that the coolant phases separate. This could result in steam cooling of some of the upper fuel elements in a channel or of some of the channels connected to the midplane of the header. Prolonged stratification leads to sheath damage within a few minutes, and therefore this defines the time by which emergency core cooling must become effective. An ECCS injection pressure of about 4 MPa limits the duration of stratification to the order of seconds, which is adequate. Once refill has occurred, the pumps keep recirculating flow, and this pattern continues into the long term until the pumps are tripped. Pump trip has been automated on certain HWRs in the longer term, after emergency core coolant refill, in order to avoid pump cavitation once the circuit has refilled with cold water.

The break is a major heat sink for decay heat. Breaks greater than half the size of a feeder break can remove all the decay heat from one HTS loop. If not cooled, the steam generators can, however, hold up HTS pressure. Thus, the steam generators must be cooled down fast enough to ensure that ECCS injection is not blocked. Crash cooldown takes the steam generators' pressure from 4 MPa to near atmospheric in about 15 minutes and this allows continued ECCS make-up flow.

There is a special case of small breaks in which limited fuel damage cannot be prevented, i.e. breaks which occur in the piping connected to one fuel channel only (single channel events). The rest of the HTS experiences just a small break, but the affected channel could suffer both fuel and channel damage. The causes of such accidents include complete flow blockage of one pressure tube, spontaneous pressure tube rupture, feeder stagnation break and end fitting failure. All of these failures can damage fuel, the first and the third by overheating due to reduced flow, the second by mechanical damage following the pressure tube rupture and the last by ejection of the fuel into the calandria vault, followed by mechanical damage and oxidation in the vault atmosphere.

It should be noted that a flow blockage severe enough to cause pressure tube failure due to overheating would have to be greater than 90% of the channel flow area. A single channel event leading to channel failure also requires analysis of the pressure transient within the calandria, in order to show that the calandria vessel itself remains intact, that the shutdown system devices within it can still perform their function and that the break does not propagate to other reactor channels.

(ii) Large breaks

Three break locations are enough to represent the range of HTS behaviour: at the core inlet (inlet header), at the core outlet (outlet header), and upstream of the pump (pump suction pipe) (Fig. 91).

These breaks affect the two core passes in different ways. The core pass upstream of the break (upstream and downstream means with respect to the nominal flow direction) always has the flow accelerated towards the break. The fuel cooling is, if anything, increased and emergency core coolant refill is rapid, in the nominal

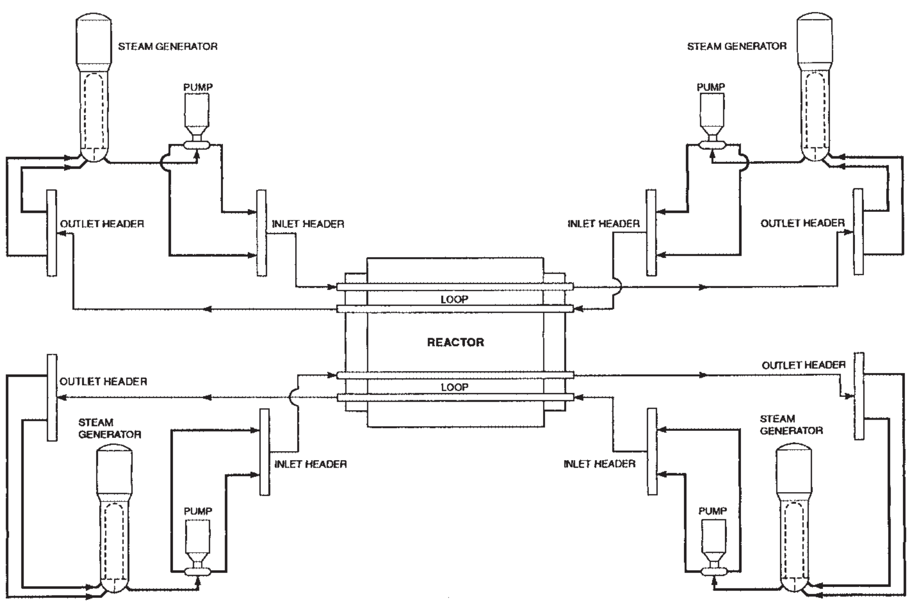


FIG. 91. Figure-of-eight HTS (CANDU 6).

flow direction towards the break. The core pass downstream of the break has its flow reduced by the break and is therefore termed the 'critical core pass'.

A guillotine inlet header break reverses the flow in the downstream core pass. As noted previously, a small break at the inlet header leaves the flow in the normal direction. It is, therefore, possible to choose a break size which leads to a momentary period of zero flow in the downstream core pass. This low flow results from a balance of flow being achieved between the break and the pumps. It is, by nature, transitory; as the circuit depressurizes, the break discharge decreases and at the same time pump force decreases as quality builds up in the pump. The latter effect is faster and therefore after a period of about 10–20 s of zero flow, the break takes over and the flow reverses towards the break. Such 'critical' breaks tend to bound the consequences of other large breaks and are therefore analysed in detail.

Flows in the long term are determined by the balance achieved between the break and the pumps (which may be tripped). In the case of smaller breaks, both refill flows and long term flows will be forward, that is, in the normal recirculating fashion. If the break is larger, refill will be in the reverse direction and this will persist into the long term. Flow in intermediate breaks may reverse when the pumps are tripped.

As the inlet header breaks are the closest to the downstream core pass, they can reduce the flow fastest and therefore lead to the highest fuel temperatures. Sustained low flow does not persist and fuel is cooled down while the coolant pressure is still relatively high. The combination of high temperature and high coolant pressure requires an analysis of the possibility of (i) embrittlement of the fuel sheath resulting from oxygen uptake, and (ii) pressure tube strain due to overheating at high pressure.

Large breaks at the outlet end of the core (outlet header break) may be able to reverse the flow in the downstream core pass. Smaller outlet header breaks allow continued forward flow: refill is in the forward direction and the long term flow pattern is one of recirculation. In the case of the largest outlet header break, the voiding of the downstream core pass is slower than for inlet breaks, since the path from the break to the core is longer and the resistance higher. Thus, when sustained low flow does occur there is less residual heat in the fuel. Fuel temperatures during sustained low flow rise to values slightly lower than in the inlet header case. However, reactor outlet header breaks are limiting for sheath strain failures because the sheath temperature is high when the coolant pressure is low.

At first, the flow goes backwards through the downstream pump, but as the circuit depressurizes, the pump acts more like a check valve. Injection water passing into the inlet header of the downstream core pass is therefore prevented from going through the pump to the break and is forced instead in the forward direction through the core pass. Refilling of both core passes is in the forward direction. The long term flow pattern for the largest reactor outlet header break is one of recirculation until the pumps are tripped: flows are then directed in each pass towards the break. Pump suction breaks are hydraulically similar to reactor outlet header breaks.

It should be noted that very long emergency core cooling interruptions can be tolerated a month or more after the accident, by which time all decay heat from the fuel can be removed through the pressure tube to the calandria tube and then to the moderator, while keeping fuel temperatures below 600°C, without any coolant being present in the channels.

(d) System behaviour

A reactor physics code, which calculates the power transient, coupled with a system thermohydraulics code, are used to predict circuit and individual channel thermohydraulic behaviour. A more detailed analysis of channel behaviour provides data on hydrogen production and pressure tube deformation. Pressure tubes may expand and contact the calandria tubes, in which case the heat transfer on the outside of the calandria tube becomes a consideration, requiring a calculation of moderator circulation and local temperatures. A further level of detail provides sheath temperatures and fission product releases which are used in a containment calculation to determine how much activity has leaked outside the building. The dispersion and dilution of this material before it reaches the public domain is the subject of an atmospheric dispersion/public dose calculation. The public dose represents the end point of the calculation. Figure 92 summarizes the process.

(i) Reactor physics

In the case of a large break, a rapid shutdown is required to reduce the amount of stored heat that the ECCS has to handle. Boiling in the channels following a loss of coolant can introduce positive reactivity at a rate of up to several mk per second, for a couple of seconds; shutdown systems are designed to introduce negative reactivity fast enough to overcome this.

In the case of the larger HWRs, particularly where the voiding is asymmetric (two HTS loops), a three dimensional transient reactor physics calculation is performed. This predicts the power transient at any location in the core (Fig. 93).

(ii) Thermohydraulics

A large break could empty the affected portion of the HTS in about a minute. Fuel heat-up could be predicted to start quite early if the break were chosen appropriately. In critical breaks, the power pulse adds several full power seconds of energy to the fuel. As cooling of the fuel is lost, the stored heat is redistributed within

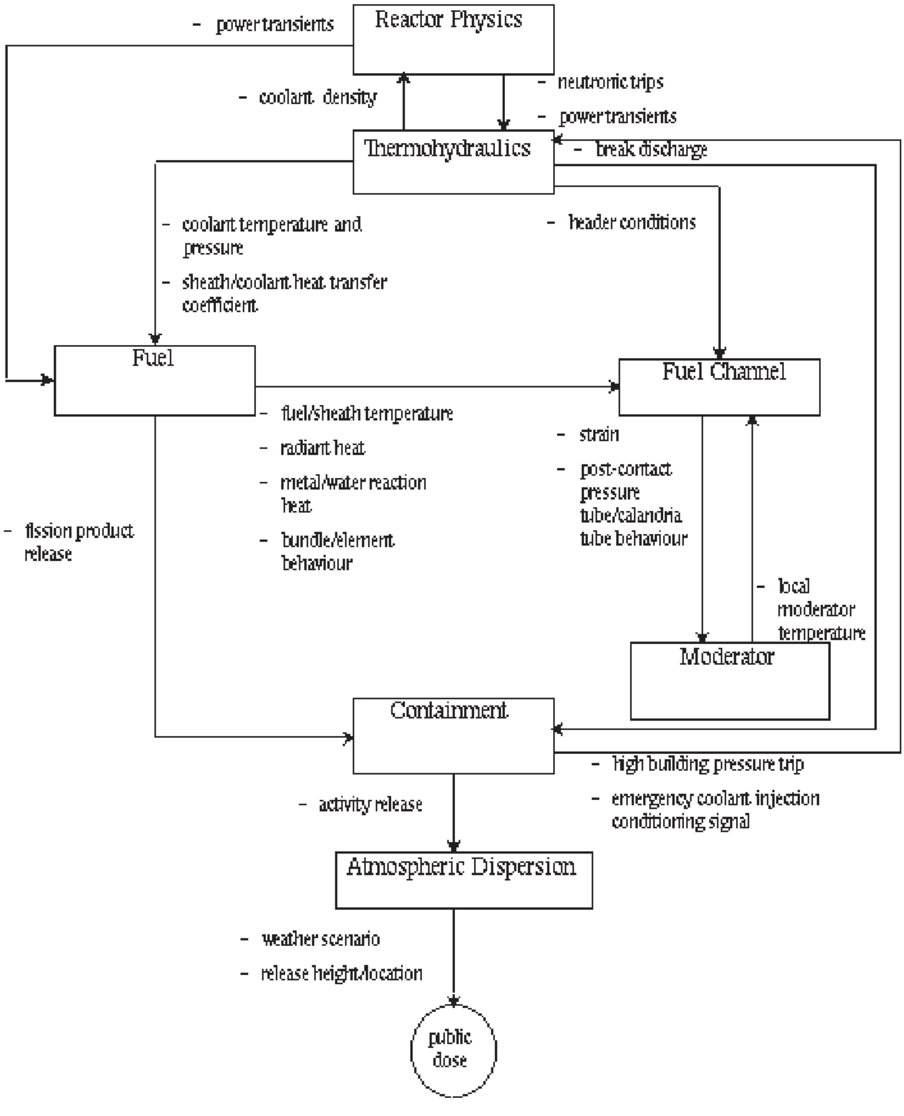


FIG. 92. Safety analysis methodology.

each fuel element within the time constant of the fuel (~7 s). Thus, the sheath heats up at about 100–200°C/s until both fuel and sheath are at the average fuel temperature, which for the hottest element is about 1200°C. Subsequently, the heat-up is limited by the decay power which, ignoring heat removal, will proceed at about 1°C/s for each per cent of full power for the hottest fuel element.

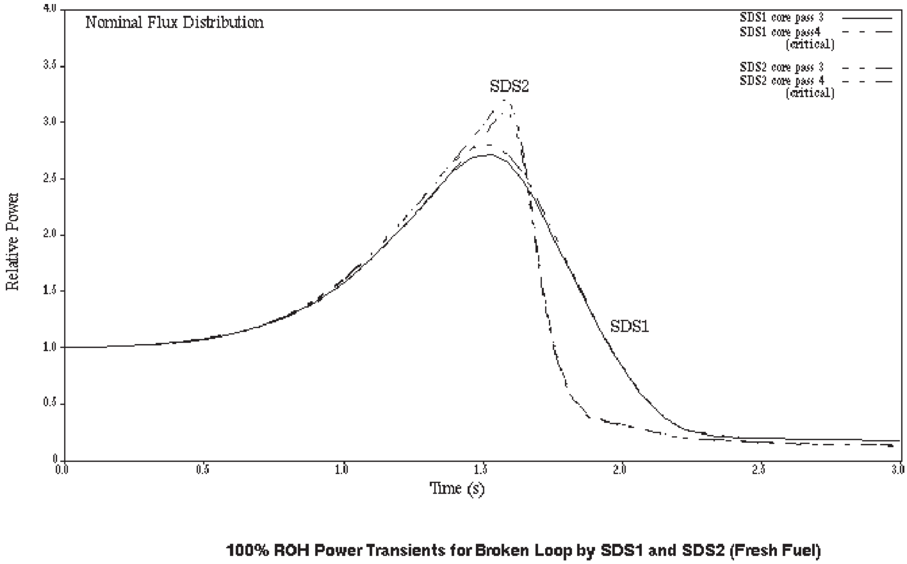


FIG. 93. Typical, large LOCA power transients.

Fuel at a temperature of 800–1200°C will not fail until the coolant pressure falls well below the internal gas pressure inside the sheath. At this point the sheath starts to strain and if the temperature is not reduced and the coolant pressure continues to fall, sheath damage will occur.

These factors determine the pressure, actuation time and flow requirements of the ECCS, namely, high pressure (~4 MPa) core cooling initially, with both flow and pressure sufficient to maintain a pressure of around 1 MPa in the channels during refill. This is coupled with a rapid cooldown of the steam generators in order to ensure continued injection for small breaks.

(iii) Fuel channels

If the pressure tube temperature is greater than about 650°C and the internal pressure is greater than about 1 MPa, the zirconium alloy material will start to deform. This continues either until the pressure tube cools down, or until it expands to contact the calandria tube. At this point its stored heat is rapidly transmitted through the calandria tube to the moderator. This cooldown, which takes place over a period of about 1–2 s, arrests further expansion and provides a direct path by which channel heat is rejected to the moderator, as well as an expanded flow area which assists channel refill.

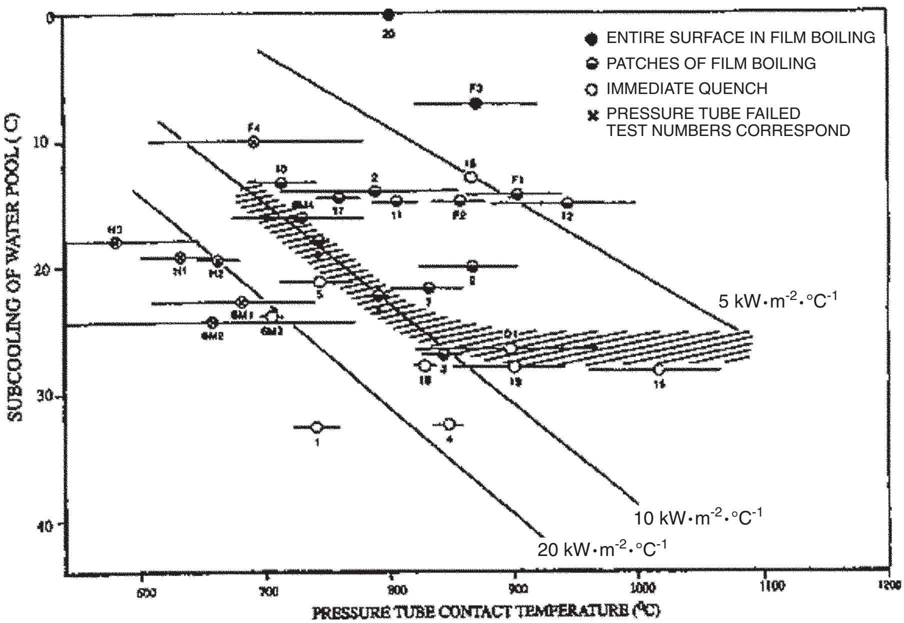


FIG. 94. Moderator subcooling requirements.

The speed of cooldown of the tube assembly depends on the moderator subcooling. At high subcooling, the surface of the calandria tube is in a state of nucleate boiling and cooldown is very rapid. As subcooling decreases towards saturation, patchy film boiling appears in places on the calandria tube, and these patches take some tens of seconds to cool down. At very low subcooling, large areas of the calandria tube are in film boiling and failure of the tube assembly is possible. The design requirement is to have sufficient subcooling in the moderator such that prolonged film boiling is precluded (Fig. 94).

(iv) Fuel

Under the loads produced by a large LOCA, the fuel sheath may be subjected to high temperatures and become stressed owing to the difference between the internal gas pressure and the falling cooling pressure. The failure modes of interest under such conditions are as follows:

— Plastic instability (‘ballooning’),
— Beryllium braze penetration,

— Sheath embrittlement due to oxygen uptake,
— Cracking of oxide layer as a result of stress,
— Volumetric expansion of the fuel as a result of UO_2 melting.

Fuel thermomechanical models incorporate all of these effects when predicting the likelihood and extent of fuel sheath failure. It should be noted that in calculating sheath embrittlement, a mechanistic calculation is performed using a time at temperature acceptance criterion rather than a fixed maximum temperature. The presence of the colder pressure tube near the bundle means that radiation heat losses become significant at elevated sheath temperatures and therefore there is less tendency for the zirconium–water reaction to 'take off' at a fixed temperature threshold.

(v) Containment

Containment concepts have already been discussed in Section 5.2.2.5. Safety analysis models of HWR containment are used to predict the containment pressure transient, integrated leakage rate and fission product release. Besides the conventional thermohydraulic multiphase gas models, they incorporate models of dousing; suppression pool; intercompartmental flow; fission product deposition, transport and leakage; and hydrogen transport. They are not unique to HWRs, except for the emphasis placed on the calculation of actual pressure transients and fission product releases. In general, the HWR containment volume per unit of electrical power is relatively large, because of the size of the calandria and the space needed by the fuelling machines on each end. Thus, the design pressure is relatively low.

Some HWR jurisdictions, while using this approach for accident analysis, adopt a more stylized approach for siting, in which a prescriptive radioactivity source term, containment pressure transient and leak rate are used.

(vi) Releases and doses

The final stage of the analysis involves calculating the dispersion of any radioactivity released from containment and assessing the resulting public dose. Dispersion of a release tends to vary inversely with distance from the source along the cloud centre line, and exponentially away from the centre line.

The dose resulting from a release is highly sensitive to weather. The dose attributable to the worst case weather (a moderate wind speed of 2 m/s with no wind swing) compared with 'average' weather differs by a factor of about 100.

The detailed assumptions are country dependent; many HWR owners use methods based on the Canadian Standards Association recommendations [75].

(e) LOCA analysis: Typical results for Indian HWRs [76]

(i) Analysis methodology used by NPCIL

LOCA analysis has been undertaken using a thermohydraulic neutronic code which simulates the HTS loop, part of the secondary coolant system and the ECCS of HWRs [77]. The thermohydraulics is based on an unequal velocity, equal temperature (UVET) model using three conservation equations and incorporating a drift flux correlation [78]. Conservation equations are solved using a modified version of the semi-implicit scheme, which was developed in order to achieve numerical stability with higher time steps.

LOCA analysis has been undertaken for the entire range of break sizes and locations, ranging from very small breaks, which can be handled by the make-up system, to the double-ended guillotine break of large diameter piping [79]. The analysis covers the plant response to the LOCA with all the safety systems functioning, as well as the response following a LOCA together with a postulated impairment of the safety system. The salient results of large break LOCA analyses (10–100%), which impose the most stringent demands on all the safety systems (i.e. shutdown system, ECCS, containment), are shown in Figs 95–98. Following large breaks, the HTS depressurizes rapidly causing voiding in the reactor core, which in turn causes a positive reactivity addition and consequent power rise. Several trip signals will be activated, any one of which is sufficient to trigger the reactor shutdown system and thereby overcome the void reactivity. Figure 95 shows the variation in actuation time of the shutdown system as a function of break size and covers the spectrum of breaks analysed for large LOCAs in the reactor inlet header. It can be seen that a reactor trip signal is reached within a second in most cases.

The actuation sequence of the trip signals is largely dependent upon break location. In general, for breaks on the reactor inlet header side, high log rate is the first signal followed by high neutron power, while for breaks on the reactor outlet header side, low pressure is usually the first signal followed by high log rate. As the break size decreases, the neutronics signal appears later and is preceded by the process signals (low pressure and reactor building high pressure).

Following the break in HTS, the coolant flow increases on the upstream side and decreases on the downstream side of the break. As the break size increases, the downstream flow ceases and starts reversing. In the case where the break size results in flow reversal, there is forward flow in the upstream pass and reverse flow in the downstream pass. Wherever these two opposite flows interact or meet in the loop there lies a region of flow stagnation or low flow. Dynamic changes occurring during the transient also lead to movement of the region of flow stagnation. If this region of flow stagnation/low flow lies in the core, then this leads to a sharp deterioration in heat transfer from the fuel cladding. This results in the deposition of fuel decay

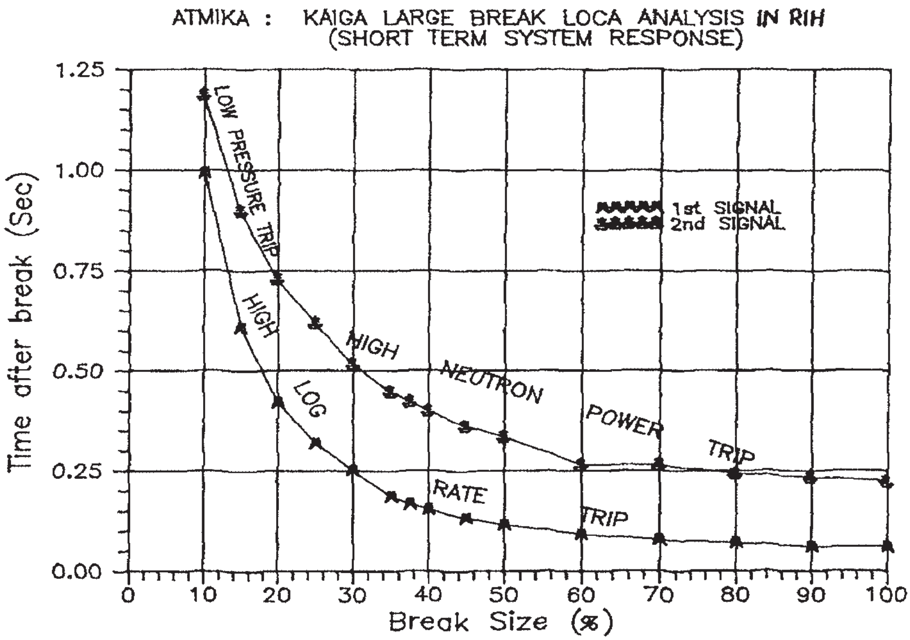


FIG. 95. Variation in time of trip signals as a function of break size.

energy as well as fuel stored heat into the cladding and hence a rise in its temperature. With this rise in cladding temperature, the Zircaloy–water reaction, which is exothermic in nature, may also start contributing towards energy generation and hence result in a rapid rise in the fuel cladding temperature. Early flow stagnation in the core that is associated with the time when reactor power and fuel stored heat are high produces a sharp rise in sheath temperature. Figure 96 shows the variation in peak sheath temperature for a range of break sizes. It identifies the break range within the spectrum of large break LOCAs which produces the most extreme fuel temperature excursions resulting from early flow stagnation. It can be seen that these critical breaks are more severe for the reactor inlet header than for the reactor outlet header. The sheath temperature variations for selected breaks during the initial minute following a break are given in Fig. 97.

The identified limiting (critical) breaks which impose the most stringent demands on the ECCS, along with the guillotine breaks of the reactor inlet and outlet headers which impose equally stringent demands on containment, are analysed for the long term system response over the entire transient period, starting from early blowdown up to the long term quasi-steady-state phase. In general, the period of flow stagnation is brief; by the time sustained flow from the low pressure recirculation system is established in the HTS, an adequate water inventory is still left in the

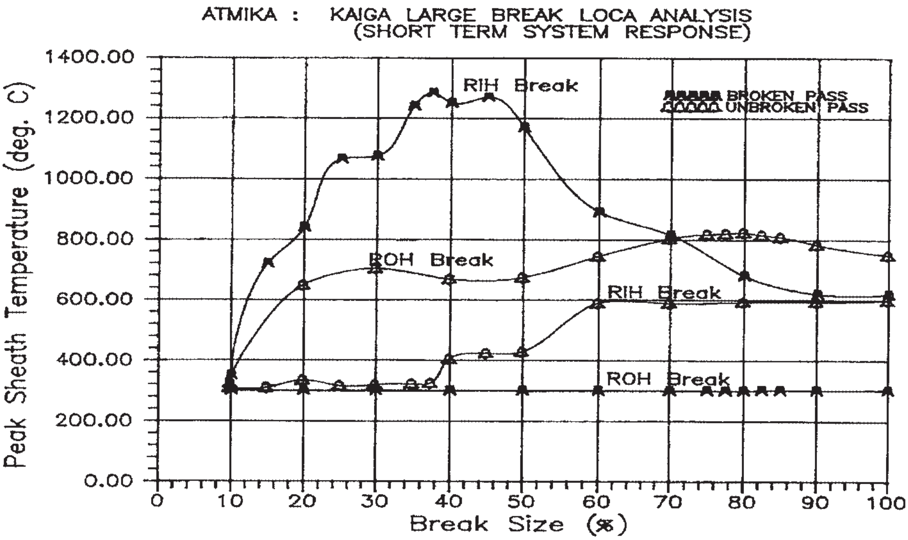


FIG. 96. Variation in sheath temperature with break size.

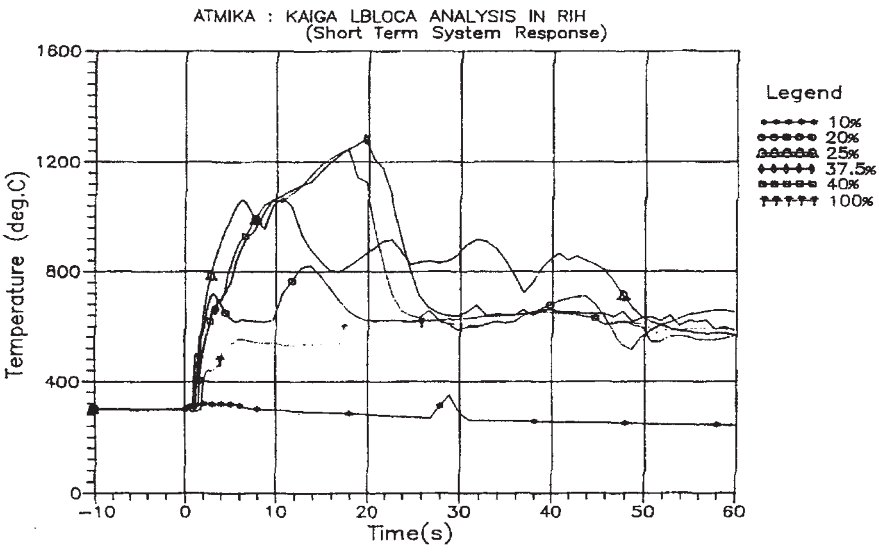


FIG. 97. Sheath temperature variation in core with broken pass.

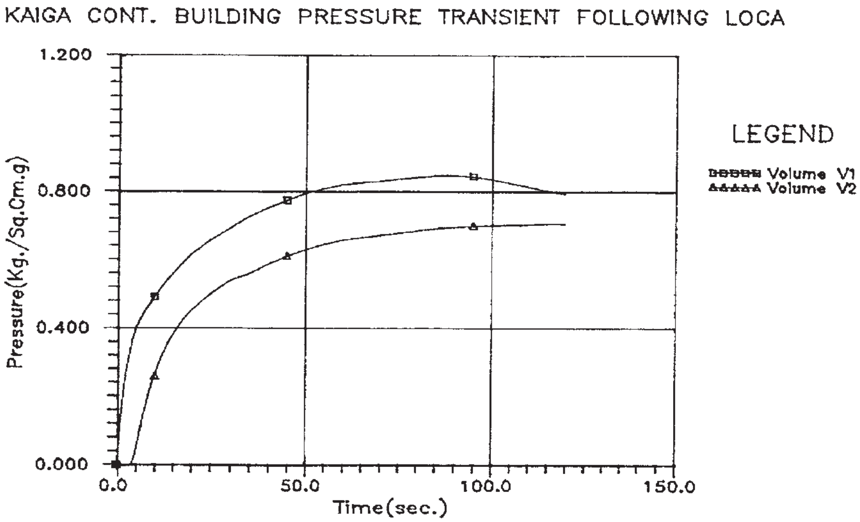


FIG. 98. Containment building pressure transient following a LOCA.

accumulator and complete quenching and refilling of the fuel cladding occurs in less than ten minutes in large break LOCA scenarios. After rewetting of the channels, long term recirculation injection from the ECCS maintains the reactor core in the flooded condition in order to remove decay heat over the long term.

The pressure transient in the containment following a double-ended guillotine rupture of the reactor inlet header, calculated using the NPCIL computer code PACSR, is depicted in Fig. 98 [80, 81]. The predicted peak pressure could attain 0.85 kg/cm^2(g) around 90 s into the accident.

(ii) Approach to LOCA analysis followed by Bhabha Atomic Research Centre (BARC)

A methodology for LOCA analysis has been developed at BARC to study in detail the various phenomena following a LOCA. This methodology involves computations in a number of stages using three sets of computer programs. The first analysis simulates, globally, the HTS using the computer programs COHRA (developed at BARC); RELAP HWR (a modified version of RELAP 4); ATHLET cycles C and D; and RELAP 5 MOD 1 and MOD 3. The first three codes are equilibrium codes that use either slip or drift flux models. The others are multifluid codes. In this thermohydraulic analysis, the reactor channels are usually grouped together. Figure 99 shows the results of one such analysis for a double-ended reactor inlet header break.

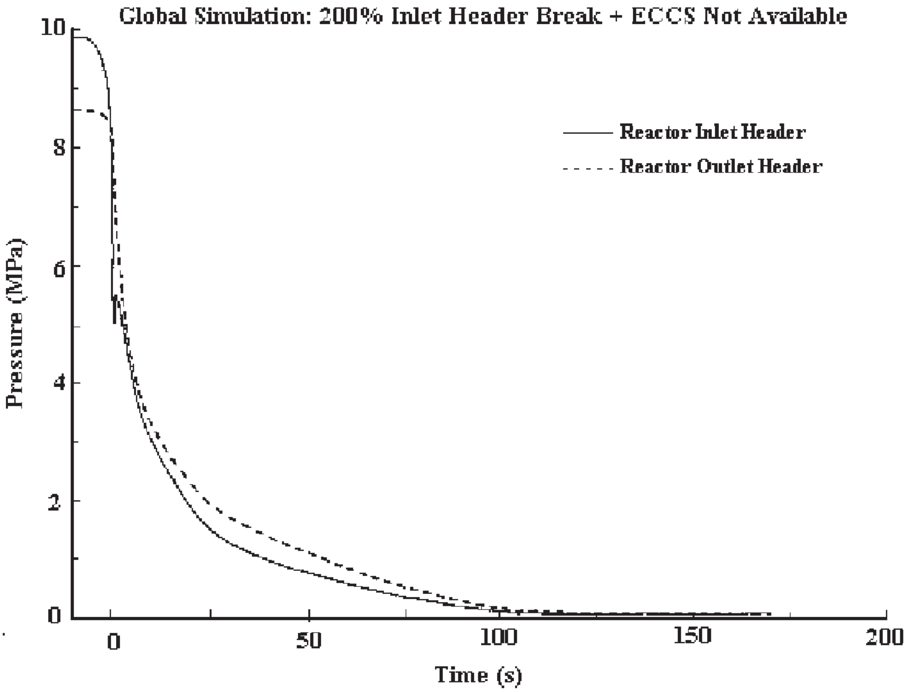


FIG. 99. Reactor header pressure transients.

The reactor header conditions are obtained from this analysis and are used as boundary conditions for the analysis of an individual channel (termed a slave channel). In this analysis, if the header is found to be stratified (using the HFLOWR code), either a vapour pull-through or water entrainment code (BFQ) is used to determine the channel inlet conditions for the channel selected for detailed analysis. Figure 100 shows some of the results of pull-through from the reactor header in the presence of other feeder pipe connections [82].

In the slave channel analysis, the channel is simulated in detail and flow, pressure, quality, etc., are determined at each location in the channel. Figure 101 shows some of the results of the slave channel analysis at the centre of the channel. The flow conditions at each location are fed into the HFLOWR code, which is used to determine flow regimes in channels with fuel bundles [83]. Figure 102 shows the flow regime prediction for an HWR coolant channel, along with the results of an exercise for the validation of this code. Figure 103 shows some of the results obtained using HFLOWR for a reactor channel experiencing a LOCA combined with the loss of the ECCS [84].

In the final stage of calculation, a thermomechanical code (HT/MOD 4) is used to perform detailed 3-D temperature calculations on different components of the

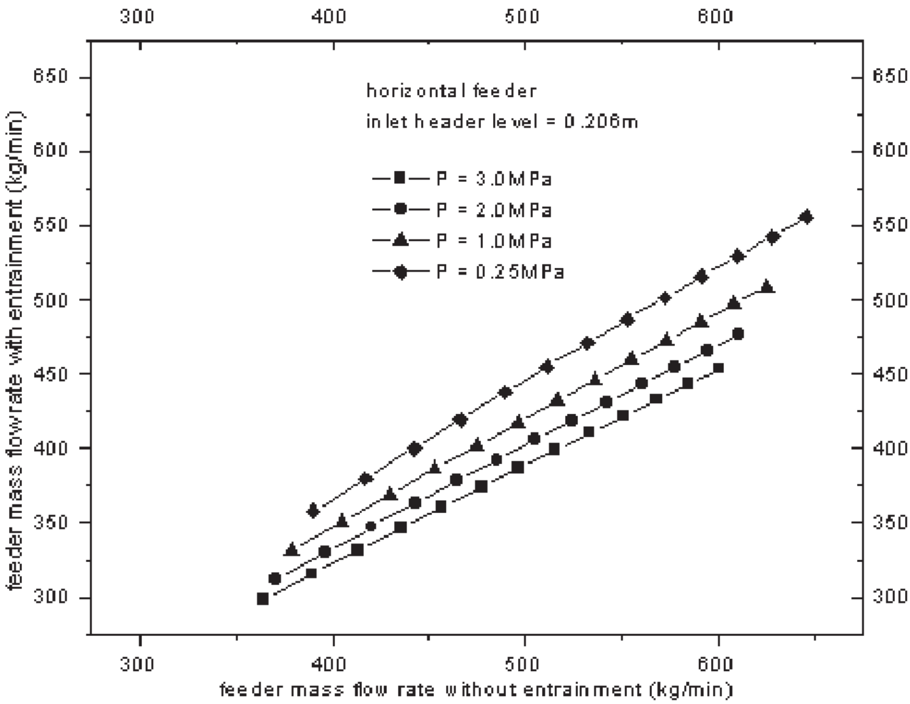


FIG. 100. 220 MW(e) PHWR analysis: Feeder mass flow rate predictions.

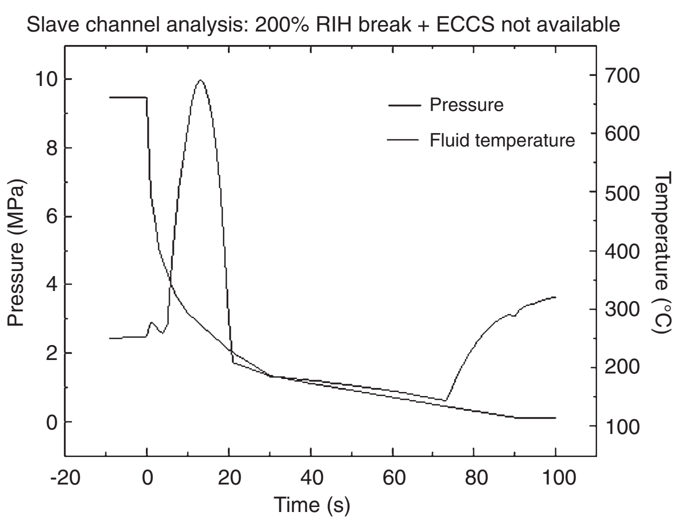


FIG. 101. Variation of thermohydraulic parameters at the centre of a hot channel.

reactor channel taking into account appropriate heat transfer coefficients, metal–water reactions, radiative heat transfer, etc., [85]. It has models for pressure tube degradation due to ballooning or sagging. It also simulates the thermal slumping of fuel bundles. Figure 104 illustrates some of the results obtained by this code for the case of a LOCA combined with the failure of the ECCS.

All these sets of codes have been developed in such a way that they can be integrated with one another.

Following a LOCA, the consequent pressure transients in the containment are calculated using the CONTRAN code (developed at BARC) [86]. This code is also used as an assessment tool to cross-check the containment design parameters. A systematic approach is followed to account for the effect of various parameters (such as break sizes, reactor power level, etc.) in maximizing the containment peak pressure.

(f) Conclusion

A small LOCA in an HWR exhibits continuous recirculating flow and no fuel damage, except for some cases both specific to and limited to a single channel.

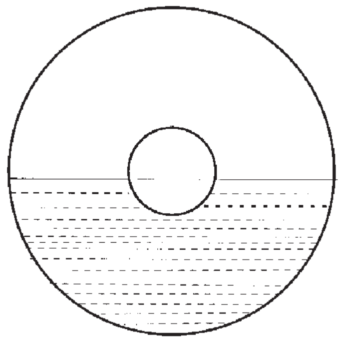
In the case of a large LOCA, the refill direction is determined by the interaction between the running (or tripped) pumps and the break. Refill times are of the order of two minutes or less and sheath damage may occur, especially for the high power, high burnup elements. Pressure tube expansion occurs in some channels for 'critical' large breaks and is arrested by the calandria tube. The resulting doses are well below regulatory guidelines.

5.3.1.2. Loss of flow

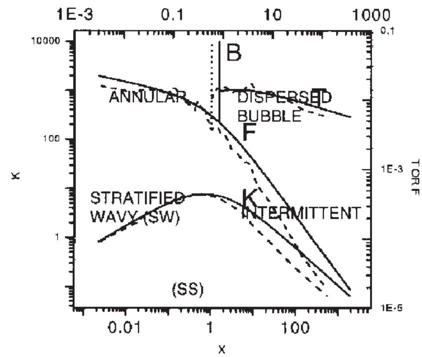
(a) Loss of Class IV power

The behaviour of HWRs following a loss of Class IV power is similar to that of other reactor types and is summarized below.

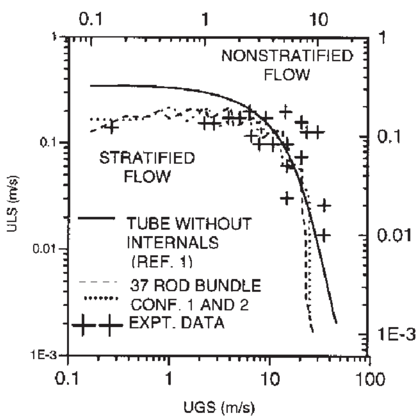
Primary coolant is circulated through the HTS by two pumps per loop, or four pumps in recent single loop plants. The loss of forced circulation of the coolant can be caused by various initiating events, e.g. loss of external electrical supply to all the HTS pumps. Other cases analysed are pump shaft seizure and partial loss of Class IV power (to one or more pumps). The pumps are designed to have a large inertia so as to increase the pump rundown time. This helps match the available cooling to the cooling required after the reactor trip. About two minutes after the loss of electrical power, thermosyphoning provides adequate fuel cooling. Normally, the regulating system itself can terminate these events. When this system is assumed to be unavailable, either of the two shutdown systems is effective. In Indian 220 MW(e) HWRs, the reactor trips after about two seconds of the 'No



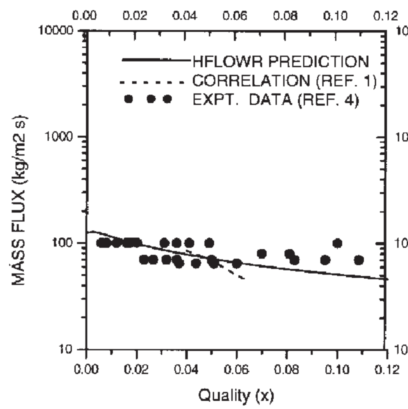
A Flow Channel With Internals



Flow Regime Map - Dimensionless Plot
for PHWR Coolant Channel



Code Validation Using Steam-Water Experimental
Data (37 Rod Bundle)



Validation of HFLOWR with Osamusali's
Experimental Data (37 Rod Bundle)

FIG. 102. Flow regime prediction in horizontal channel with internals.

primary coolant pump operating' trip signal. The objective of the analysis is to show that:

- The reactor is shut down to prevent fuel sheath failures and to ensure that the HTS stays intact;
- The thermosyphoning flows are sufficient to remove decay heat in the long term, without fuel sheath failures.

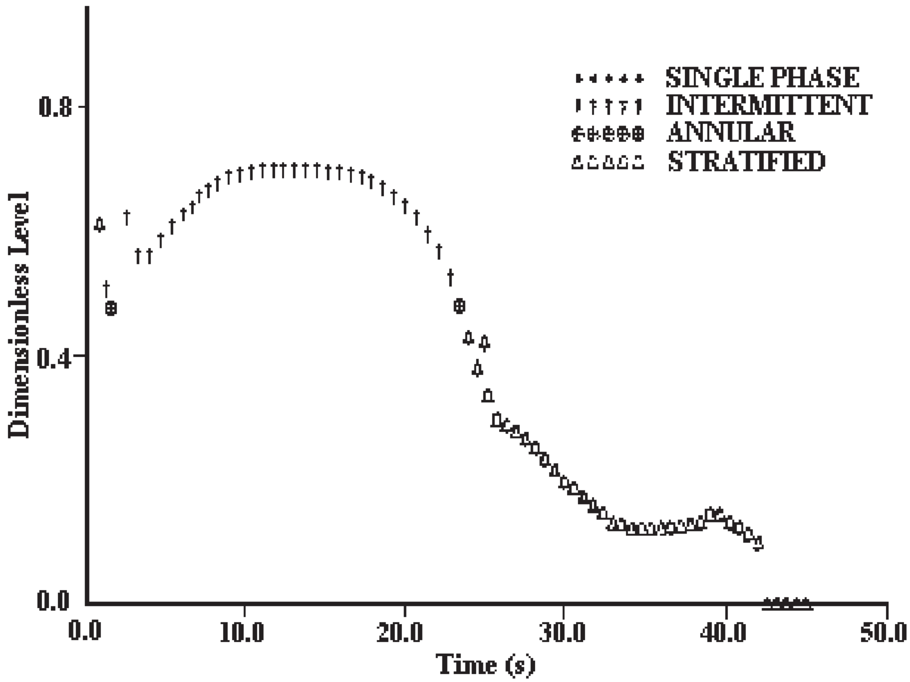


FIG. 103. Predicted flow regimes in a hot channel away from the break following a LOCA and the loss of the ECCS.

The consequences of this event are analysed by predicting the transients of coolant flow, coolant pressure, and fuel and sheath temperatures.

A flow reduction is sensed by several regulating and safety system signals such as: low flow, no power to the HTS pumps, low header-to-header pressure drop, high HTS pressure and possibly high power/high rate of power increase. These conditions automatically result in shut down of the reactor.

At the initial stage, HTS pressure rises and fuel cooling deteriorates; the formation of voids in the channels causes a power increase. The pressurizer steam bleed valves open and discharge coolant into the degasser condenser in order to maintain coolant pressure control. A further rise in pressure may cause the HTS liquid relief valves to open and to discharge coolant into the degasser condenser. These valves close when the HTS pressure drops below set points following the decrease in reactor power. After pump rundown, flow continues over the steam generators and around the HTS, induced by the buoyancy force arising from differences in coolant density (Fig. 105). This thermosyphoning may be two phase.

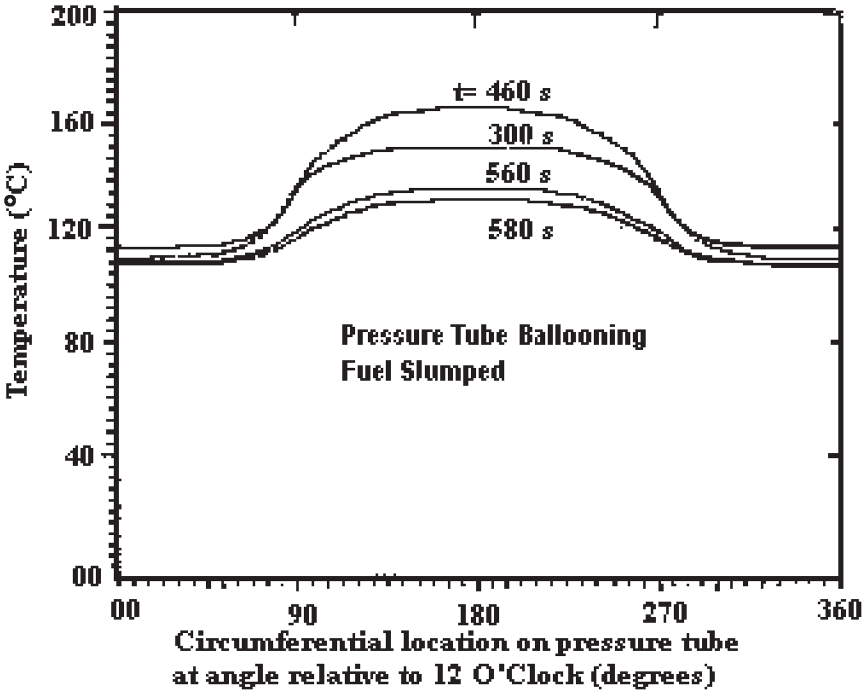


FIG. 104. Circumferential temperature distribution in a pressure tube following a LOCA and the loss of the ECCS.

(b) Thermosyphoning

Thermosyphoning is defined as the natural convective flow of HTS coolant over the steam generator. It is the predicted mode of heat transfer in many scenarios, for example, loss of electrical power, where the HTS pressure is high and the extent of voiding is small, and DBE, where the system pressure is near atmospheric and a significant degree of boiling is possible.

Thermosyphoning is, therefore, of interest over a range of HTS coolant inventories and secondary circuit pressures (or temperatures). The reactor core power is at decay power levels of ~4% or less.

The thermosyphoning flow can be single or two phase. Boiling (two phase) thermosyphoning is predicted under reduced HTS coolant inventory conditions which may be caused either by a net leak from the system or by uncompensated shrinkage due to cooldown.

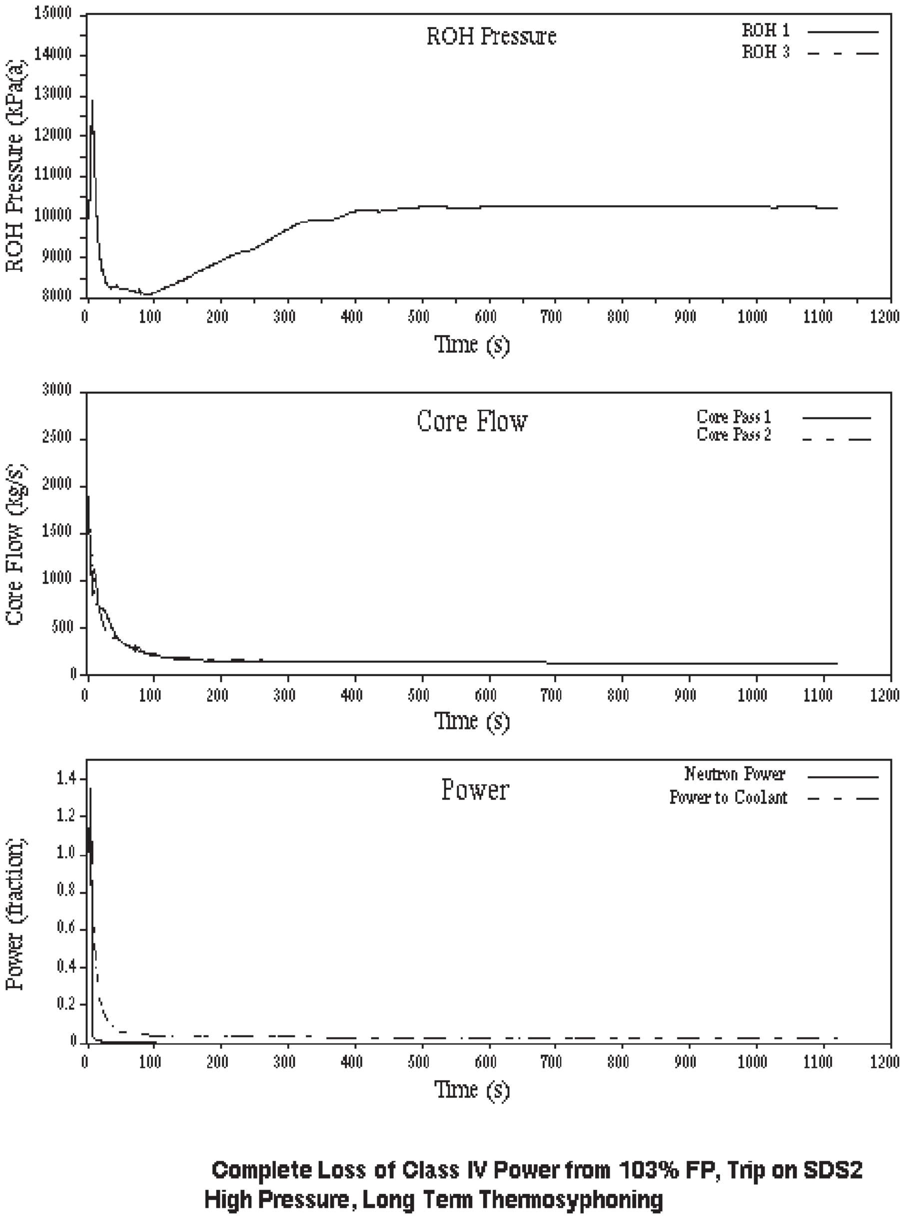


FIG. 105. Long term thermosyphoning after loss of Class IV power.

The effectiveness of single phase thermosyphoning in the design has been demonstrated in reactor commissioning tests. Code predictions show that two phase thermosyphoning can be effective at low pressure because fuel channels are liquid filled and the system void is concentrated in vertical components where the buoyancy force produces flows considerably greater than in single phase thermosyphoning. Two phase thermosyphoning flows are lower at high pressure and channel flow stratification can be expected for system voiding greater than about 20%. With regard to further reductions in HTS inventory, unidirectional thermosyphoning will break down and the flow pattern will change to one of intermittent flow from individual channels to the headers, and countercurrent steam/water flow from the headers to/from the steam generators. Thus, the safety analysis must demonstrate that sufficient coolant inventory is present to prevent fuel sheath damage.

5.3.1.3. Loss of heat removal

(a) MSLB

The safety aspects for steam line breaks are:

- The loss of a reactor heat sink as the secondary side inventory is lost through the break,
- The potential for damage to mitigating equipment,
- The containment overpressure.

In the case of MSLBs occurring outside containment (in the turbine hall), it must be demonstrated that equipment which is required and which is assumed to mitigate the event is not damaged by the forces causing the break and that no damage is incurred by the turbine hall structure. In MSLBs occurring inside containment, the pressure rises rapidly and the safety aspect of concern is the reactor building integrity, including the integrity of the reactor building's internal walls.

The depressurization of the secondary side causes a corresponding depressurization and cooling of the primary side, which results in insignificant reactivity increase or, in designs where boiling in the channels occurs, a negative reactivity and a power decrease, and is not a safety concern.

Large steam line breaks are limiting in terms of early containment peak pressure and the time available to introduce an alternative heat sink. Small breaks test the trip coverage and can lead to long term containment pressurization after the containment dousing water is exhausted.

Since the HTS pumps may trip on low pressure, the ability to remove heat through thermosyphoning, particularly if the HTS is two phase, must be confirmed.

The short term heat sink is provided by continued feedwater flow to the steam generator. In the longer term, HWRs have a high pressure backup heat sink (the shutdown cooling system), and therefore it is not necessary for the operator to depressurize the HTS before valving in this alternate heat sink in emergencies. The shutdown cooling system is backed up by a low pressure EWS which is connected to the steam generators in many HWR designs.

(b) Feedwater line break/loss of feedwater

Loss of feedwater events include a loss of secondary circuit feedwater resulting from loss of power, a feedwater line break and the DBE. They have no immediate effect on fuel cooling but a long term reactor heat sink must be provided. These events will induce an early trip, for example, on low steam generator water level or on high HTS pressure, in which case power will be quickly reduced to decay power. Unless secondary circuit inventory is lost, as in a large steam main break, the liquid remaining in the steam generators is sufficient to remove the decay heat for a period of ~30 min. In the longer term, the shutdown cooling system can be activated to provide an alternative heat sink to the steam generators. If secondary circuit inventory is also lost, as in a DBE, the EWS is activated on depressurization of the steam generators. In Indian HWRs, firewater performs this function. The EWS or firewater also acts as backup to the shutdown cooling system.

The DBE is more complex. The assumptions include a complete steam main severance outside containment (steam and feedwater lines inside containment are seismically qualified) plus a complete loss of all non-qualified electrical power systems, which, were they to occur, would cause a rapid trip. Make-up to the steam generators is achieved either through the EWS or through firewater. The HTS eventually stabilizes at low pressure and ≤10% void, with two phase thermo-syphoning operating as the mode of heat removal. In Indian HWRs, the auxiliary feedwater system and the power supply to the auxiliary steam generator feed pumps are seismically qualified and will be available.

5.3.1.4. Loss of reactivity control

Control of reactivity is required to keep reactor power at the specified level and to compensate for changes in reactivity that occur as a result of actions such as refuelling and fuel burnup. Spatial control of reactivity is required to keep the reactor flux shape close to nominal in order to avoid local power peaking, thereby maximizing the power produced while at the same time keeping within operating power limits.

The reactor regulating system is designed to perform these functions. It is composed of input sensors (ion chambers, in-core flux detectors), control logic, reactivity control devices (adjusters, light water zone controllers, mechanical control absorbers), hardware interlocks, and a number of display devices. Reactor regulating system control is accomplished by digital computer controller programs which process the inputs and drive the appropriate reactivity control and display devices.

Short term reactivity changes are initiated primarily by level changes in the liquid zone controllers. Adjusters and control absorbers augment the range of control. Long term reactivity control is achieved by removal of moderator poison and by on-power refuelling. Automatic addition of moderator poison and withdrawal of shut-off rods are also regulating system functions. Withdrawal of shut-off rods is permitted only after a reactor trip signal has been cleared following shutdown.

A digital computer controls all major aspects of short term reactivity. An identical and independent computer acts as a backup to the first.

If a fault occurs in the reactor regulating system and leads to an increase in reactivity, the set-back and/or step-back functions would likely intervene and reduce power. However, these are assumed to be unavailable for the purpose of this analysis. Malfunctions which lead to a decrease in power have no safety significance and are not considered.

The objective of the safety analysis is to demonstrate that each shutdown system can prevent fuel centre line melting and fuel sheath damage, and ensure that HTS pressure stays within the appropriate code limits. Since the rate and depth of the shutdown systems are set by large and small LOCA respectively, fast loss of reactivity control accidents are not limiting.

In the case of neutron power increases of greater than 10%/s, shutdown will take place on receipt of the high rate log neutron power trip signal. With a more gradual rise in power, the shutdown will occur on a high HTS pressure trip and/or on a high neutron power trip. In the case of a very slow loss of reactivity control, the shutdown systems can also be manually actuated.

Loss of spatial reactivity control is also analysed for large HWRs. In these HWRs, two independent networks of regional overpower protection system detectors, one on each shutdown system, are distributed through the core. To test their effectiveness, a very large number (hundreds) of distorted flux shapes, which may occur during normal operation, are simulated, and then the reactor power is assumed to rise until the shutdown system trips on the regional overpower protection detectors. The fuel conditions at the time of the trip are then determined and compared with the criterion that the fuel sheath must not be damaged (i.e. it does not dry out or experience a limited sheath temperature excursion). Indian HWRs use a variant of the regional overpower protection system, termed the core overpower protection system.

5.3.2. Severe accidents

Despite the high level of safety of currently operating nuclear power plants, their response to severe accidents has become the subject of greater international interest in the last two decades [87]. This section discusses the unique aspects of HWR design relevant to severe accidents and summarizes severe accident and PSA work done by HWR designers and owners. Finally, it describes the unique phenomenology of severe core damage accidents in HWRs.

5.3.2.1. Barriers to severe accidents

The characteristics of the HWR relevant to severe accidents are set first by the inherent properties of the design, and second by the safety and 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 for some time, 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/or the concrete calandria vault would significantly delay the challenge to containment. Furthermore, should core melting lead to containment overpressure, the containment behaviour would be such that resulting leaks through the concrete containment wall would reduce the possibility of catastrophic structural failure occurring.

The HWR licensing philosophy in most jurisdictions requires that each accident, together with the 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 HWRs with two, independent, dedicated shutdown systems, and the likelihood of anticipated transient without scram is negligible.

Severe accident sequences are normally identified through a Level 1 PSA. However, the event sequences for HWRs include a number of severe accidents (LOCA plus LOECC) within the design basis. In these cases, the moderator can remove decay heat from the reactor in the absence of any coolant in the channels. The fuel will be badly damaged but the UO_2 will not melt and channel integrity will be preserved.

A second category of severe accident encompasses containment impairments, e.g. failure of the containment ventilation system to effect isolation. Within the design basis, process system failures are combined with containment impairments. However, a process system failure produces either no fuel damage, or limited fuel damage (e.g. large LOCA). Thus, although the combination of process system failures and containment impairments does not result in severe fuel damage, it is considered severe because it has a low probability of occurring and can lead to radiological releases via the pathways outside containment.

It is, therefore, useful in the case of HWRs to distinguish three categories:

- *Severe accidents within the design basis*, in which the core geometry is preserved (fuel remains inside intact pressure tubes).

- *Severe accidents beyond the design basis*, in which the core geometry is preserved. They are normally identified by a systematic plant review or by a PSA, and are too low in frequency to merit inclusion in the design basis set.
- *Severe core damage accidents beyond the design basis*, in which the fuel channels fail and collapse to the bottom of the calandria.

(a) Pressure tubes as a pressure boundary

All the reactivity devices (control rods, shut-off rods) penetrate the moderator but not the coolant pressure boundary, and therefore, with one exception, they are not subject to hydraulic forces from a LOCA. The exception is a single pressure tube failure, which constitutes a small break LOCA. The calandria tube may or may not contain the pressure tube rupture, although no credit is taken for this in the safety analysis, 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 that a channel failure will not propagate to other pressure tubes, nor will it damage more than the neighbouring few shut-off rod guide tubes. The shut-off rod shutdown system is therefore provided with enough redundant mechanisms to remain effective without taking credit for the rods that have possibly damaged guide tubes, nor for the most effective rod among those with undamaged guide tubes.

(b) 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.

(i) Shutdown cooling system

A shutdown cooling system is provided for normal decay heat removal (as an alternative to the steam generators) and for cooling below 100°C. It can operate at full HTS temperature and pressure, and can therefore be used as an emergency heat sink to cool down from hot, shutdown, full pressure conditions, should the steam generators be unavailable.

(ii) EWS

Most HWRs also have either an EWS, seismically qualified to remove heat after a DBE, or the potential to add firewater to the steam generators directly. These systems provide a supply of water to the steam generators which is independent of the normal and auxiliary feedwater. They can also be used as emergency heat sinks should normal and auxiliary feedwater, and the shutdown cooling system, be unavailable.

(iii) Emergency coolant injection

This system has been described in Section 5.2.2.4.

(iv) Moderator

In normal operation, about 4.5% of the thermal energy produced by the fuel is deposited in 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 the event of a severe accident, the same system will remove decay heat from the fuel channels, even if they contain no coolant at all. In such a case, the fuel would be severely damaged, but the UO_2 would not melt, and the channel would remain intact and retain the debris. This capability has been verified by full diameter channel tests carried out at the Whiteshell Laboratory in Canada. The moderator thus constitutes a distributed, low pressure emergency heat sink surrounding each fuel channel.

(v) Shield tank cooling

The cylindrical calandria shell is located inside either a metal shield tank or a concrete calandria vault, which is filled with water to provide both cooling and radiation shielding. This passive water inventory can absorb decay heat in the event of a severe accident and slow down the progression of core melting.

In addition, heat can be actively removed. In normal operation, heat is generated in the calandria shell and in the end shields which support the fuel channel 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. The vault floor itself is typically 2.5 m thick. Should both the emergency coolant injection system and the moderator heat sink be lost after a LOCA, 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.[4]

Figures 106 and 107 show a stylized sequence of events and the defences accorded by the moderator and calandria vault/shield tank.

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[4] In particular the availability of cooling water and electrical power to the shield cooling system.

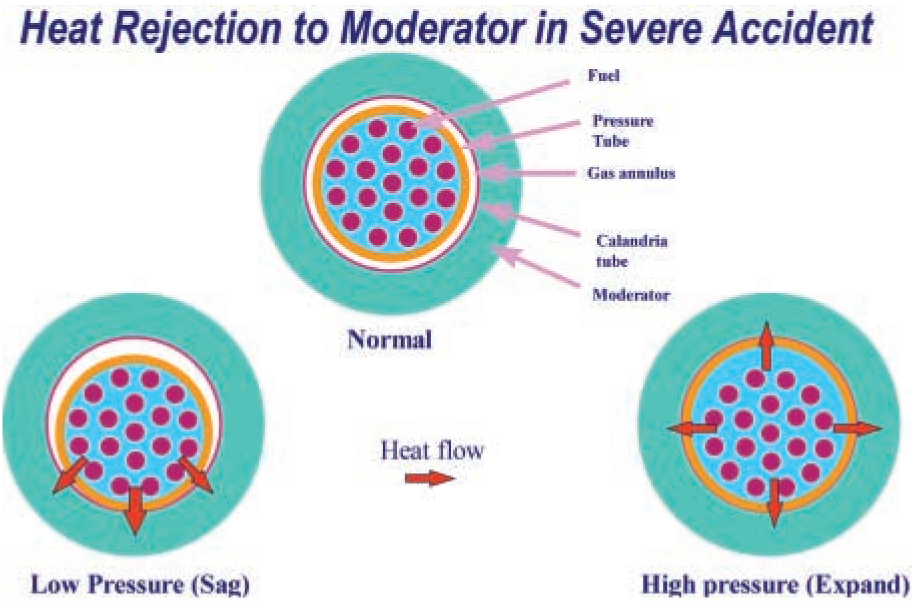


FIG. 106. Heat rejection to moderator in the event of a severe accident.

(c) Containment

The types of HWR containment system have been described in Section 5.2.2.5. 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 the containment. The control of hydrogen following such severe accidents is influenced by:

- Natural circulation of the containment atmosphere, especially in single unit HWRs.
- Forced circulation provided by containment air cooler fans.
- The effect of hydrogen igniters in recent HWRs. These are powered by the most reliable source of electricity (Class I batteries) and are engineered to reduce local flammable hydrogen concentrations before they can reach explosive levels and to ensure that the energy from combustion is released gradually.

The long term containment hydrogen concentration in a LOCA/LOECC for a single unit HWR, if fully mixed, is about or below 4%, depending on the plant. Local

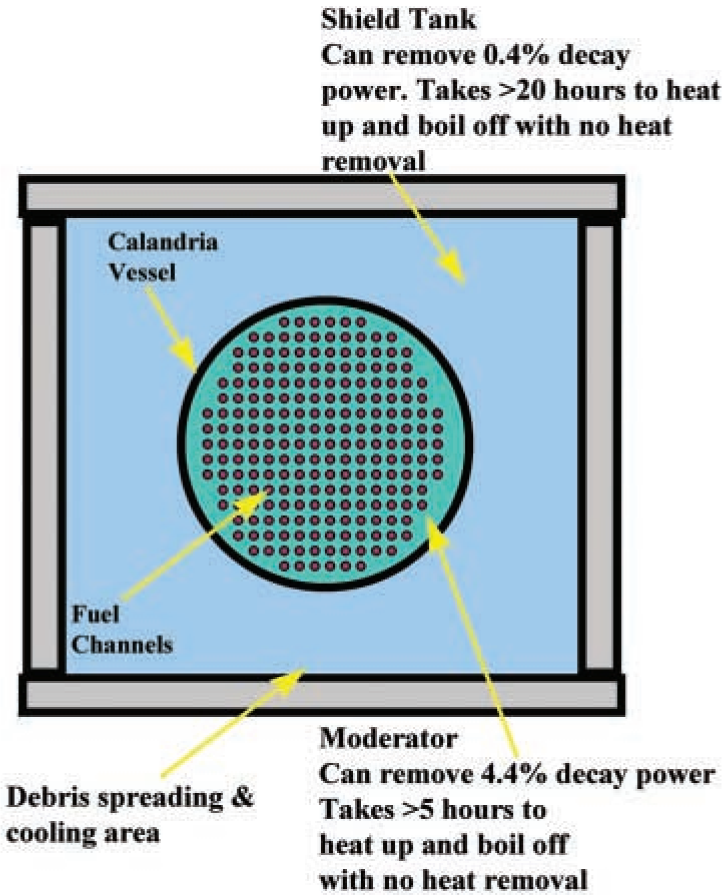


FIG. 107. Heat rejection to shield tank in the event of a severe accident.

transient concentrations could be higher. There are several reasons for the relatively low concentration:

- When the moderator acts as an emergency heat sink, it limits the pressure tube temperatures to below the level at which significant oxidation can occur. Thus, the pressure tube metal does not contribute significantly to the hydrogen source term.
- Not all the fuel sheaths are oxidized, particularly at the ends of the channel where the bundle power is lower.
- In those HWRs having two isolatable HTS loops, only half the channels experience prolonged loss of coolant inventory and high temperatures.

- Metallic structures other than the pressure tubes and fuel sheaths (e.g. guide tubes for reactivity control devices) are located within the moderator and do not experience overheating.
- HWRs have large containment buildings relative to the thermal power of the reactor.

(d) Reactivity control and shutdown systems

The reactivity control devices penetrate the low pressure moderator but not the coolant pressure boundary, as noted earlier. They are not, therefore, 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/s, and at most it reaches about 1 mk/s during shut-off rod withdrawal from a shutdown state. The total reactivity hold up in the movable control devices is ~15 mk. This low value is set, not by the need to compensate for excess reactivity, but by the operational requirement on decision and action time after a reactor trip. On-power refuelling is the longer term means of maintaining reactivity control.

Protection against reactivity insertion accidents is provided both by the control system itself, via power step-back on high rate log and high flux, and also by either shutdown system. In a severe core damage accident accompanied by loss of core geometry, the shutdown mechanisms could be damaged and would mix with the debris at the bottom of the calandria shell. However, because of the low reactivity hold up in an HWR, the reactivity effect of severe core damage is negative and therefore there is little likelihood of recriticality occurring.

(e) Two group philosophy

The two group approach has been described in Section 5.2.2.1(b). It reduces the chance of severe core damage due to common cause initiators.

5.3.2.2. Frequencies of severe accidents

The HWR nuclear industry (designers, plant owners and plant operators) has undertaken a number of probabilistic studies. From the early use of risk in defining design requirements, probabilistic techniques evolved in the 1970s into fault tree/event tree analyses, which were used to confirm availability and separation goals specified for the design. The first such study was performed in 1975 on the service water system at the Bruce A generating station. The results of this study permitted:

- Comprehensive identification of cross-links, since service water has interfaces with many systems;
- Identification of which support functions needed backup cooling water;
- Definition of the necessary operator actions required to mitigate the loss of service water.

PSA studies were performed between 1978 and 1983 on those HWR plants then being designed or built.

(a) Darlington probabilistic safety evaluation (DPSE)

The DPSE was carried out by OH as part of the design verification process for the 4 × 880 MW(e) Darlington station during the final stages of station design and construction. The study was a Level 1 PSA, and provided some analysis of the consequences and health effects for the more frequent sequences. However, no detailed off-site consequence analysis was performed for certain low probability severe accidents. Neither did the scope of the study include initiating events external to the station, nor those arising from internal fires or flooding, which were judged unlikely to contribute significantly to risk because of rigorous, deterministic design criteria.

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. Human reliability modelling was likewise comprehensive, and included detailed identification of human error opportunities both prior to and subsequent to the initiating event. Preliminary human reliability quantification was performed using prepared data tables. Final quantification of important human error probabilities was obtained by using a structured expert judging 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 covered the range of events considered to meet the severe accident definition provided in this section. FDC1 to FDC3 dealt 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 served to act as the heat sink. FDC0 comprised all events with the potential to cause a loss of core structural integrity (severe core damage). This can occur as a result of the failure of the moderator to act as a heat sink when required, failure to shut down (if such a failure were to result in fuel damage), or severe overstressing of the calandria structure.

The magnitude of the fuel damage associated with FDC3 was quite small and largely represented an economic, rather than public health, risk. FDC2 corresponded to significant fuel damage, and FDC1 was conservatively estimated to correspond to 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 were the result of the complete computer assisted integration of the event trees and fault trees, fully accounting for system cross-links. The severe core damage frequency (CDF) was bounded by the frequency for FDC0, at 4×10^{-6}/reactor-year for the scope as described (internal events).

The complex multiunit CANDU containment includes a negative pressure vacuum system and an emergency filtered air discharge system. The containment event trees included failures of overpressure suppression, envelope integrity, long term pressure control and filtration. Fuel damage and containment failure logic were fully integrated in order to search for potential cross-links. The consequences resulting from a wide range of FDC and containment subsystem failure combinations were assessed. The results included 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, off-site release. The mean frequency was estimated to be 8×10^{-7}/reactor-year.

(b) PSA of CANDU 6

Although probabilistic studies were used in the design of CANDU 6, the first Level 2 PSA was done in 1986–1987. This involved a probabilistic evaluation of internal events with a frequency of less than 10^{-7}/a, and a consequence analysis of severe core damage events and related releases. The study was undertaken jointly by AECL and KEMA, the Dutch utility organization [88]. The reference plant was an existing Canadian CANDU 6 unit, licensed for operation in 1983, with the addition of automatic cooldown of the HTS on receipt of a high end shield temperature signal. The required licensing, operating, and design information were readily available. The initial plant state was a 100% full power operation. External events such as earthquakes and fires were not assessed in this study, although covered deterministically in the design.

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 took into account the possibility of failure of the required mitigating systems.

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

The operator model used was a post-initiating event one in which operator actions were shown explicitly in the PSA event trees. A limited comparison was made, where appropriate, with other operator models.

The study analysed a total of 32 'internal' initiating events, using detailed event tree analysis to estimate the frequencies of release categories. The severe CDF for the reference CANDU 6 plant was of the order of 5×10^{-6}/a for internal events. The major contributor to severe core damage was the loss of service water, as this affects cooling water for systems such as the moderator, calandria vault, secondary HTS 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 shut down had a probability of 3×10^{-8}/a, the low value reflecting the fact that there were two independent shutdown systems.

The analysis of severe accident releases drew upon existing HWR safety analysis for predictions of the containment behaviour of many events. The study demonstrated the long duration between the initiating event and the challenge to containment integrity, as discussed in Section 5.3.2.1(b). Finally, if the sequence were to progress further, the prestressed concrete containment could crack (but was unlikely to fail) as a result of overpressure, reducing potential releases from a severe core damage accident.

(c) PSA of Wolsong 2, 3 and 4

The most recent PSA on HWRs was undertaken for Wolsong 2, 3 and 4, as required by domestic regulatory policy [89]. The utility, KEPCO, established a project which comprised two phases: the feasibility study (phase 1) and the main project (phase 2). The objective of phase 1 was to establish the PSA methodology and strategy for HWR analysis in the Republic of Korea. Conclusions drawn from phase 1 were that the scope and methodology of the analysis used to perform individual plant examination and individual plant examination of external events on PWR plants were applicable to HWRs with some additional work [90].

The primary objectives of phase 2 were to:

- Develop an overall appreciation of severe accident behaviour;
- Understand the most likely severe accident sequences that could occur at Wolsong 2, 3 and 4;
- Gain a more quantitative understanding of the expected probability of core damage and radioactive releases;
- Identify any plant specific vulnerabilities to severe accidents;
- Provide necessary information for the development of the containment improvement and accident management programme which would help prevent or mitigate severe accidents.

The Level 2 PSA was performed to cover the containment performance analysis and source term analysis. The Level 2 PSA can provide the failure probability of the reactor building for a given core damage sequence, and the release frequency for each specific release category. The scope of the study also includes initiating events external to the plant or arising from internal fires or flooding, which were judged likely to contribute significantly to risk (compared with the case of a PWR), even though considerable uncertainty would be ascribed to the result. The phase 2 project also established the framework for the accident management procedures.

(i) Advances in PSA in the Republic of Korea

Level 1 internal events PSA: The fault tree method is used in the system modelling. However, the Republic of Korea study was the first to consider CCFs for HWRs. The CCFs have been determined to be important contributors to system unavailability in the case of both safety and non-safety systems in other types of plant. Therefore, the Level 1 PSA contains the analysis of CCFs. The analysis level is very detailed and thus the fault trees are complex. The plant database could be developed by combining plant specific data and generic data utilizing the Bayesian technique. However, in HWRs the plant data are classified in great detail.

In addition to the consideration of CCFs, human reliability analysis is performed more systematically, and is based on the schematic framework of the systematic human action reliability procedure (SHARP) [91]. With regard to the in-depth analysis of human action, the Accident Sequence Evaluation Programme (ASEP) human reliability analysis procedure and detailed recovery action analysis were followed for all accident sequences [92].

Owing to the wide variety of fuel cooling mechanisms, the HWR PSA methodology leads to the creation of many, sometimes complex, event trees. Accident sequence analysis is prepared through an event sequence diagram. Also in HWRs, core damage states are categorized according to the characteristics of the release mechanism of radioactive materials. The accident sequence quantification is processed for 35 internal initiating events by the Republic of Korea's cut set generation code, KIRAP. In HWRs, the final CDF is estimated for each of the core damage states, since several core damage states are defined.

External events analysis: Previous PWR PSAs have indicated that the risk from external events could be a significant contributor to the CDF, depending on the unique features of the plant under assessment. The Republic of Korea study was also the first to include external events analysis in an HWR PSA. In this first external events analysis, it was judged that, for example, the seismic analysis could be performed using one of the methods mentioned in NUREG-1407. However, it is essential that, owing to the unique features of the HWR design that protect against the external events, the general methods applied in PWRs should be modified to reflect these features.

The seismic hazard model was used to estimate frequency of exceedance versus ground acceleration level. The specific seismic fragility of components is also developed in terms of the peak ground acceleration at the site. The internal fire and internal flood analyses were performed with due consideration being given to the HWR design and to operational features such as fire ignition or flooding sources, and to each protection measure.

Level 2 analysis: The tasks of Level 2 PSA comprise:

— Selection of accident sequences for the analysis,
— Grouping of the plant damage bins,
— Development of the containment event tree,
— Accident progression analysis,
— Source term analysis.

HWR licensing practice, as previously noted, includes a number of severe accidents within the 'design basis' that require the proponent to identify a complete list of initiating events and event combinations. As regards Wolsong 2, 3 and 4 licensing analysis, probabilistic analysis was used to identify all events with a frequency above 10^{-6}/a (systematic plant review). Any of these events which resulted in the release of radioactive materials were then analysed in order to demonstrate that regulatory release limits were met. However, with a Level 2 PSA, there is no frequency cut-off. Also, design basis accidents that could release radioactive materials into the environment are included in the Level 2 analysis.

Accident sequences are categorized according to the degree of core damage sustained and the quantity and nature of radioactive nuclides released. The headings of the containment event tree are determined on the basis of the results of the accident progression analysis and the status of the safety systems. The containment event tree is quantified utilizing a support logic such as the decomposition event tree. In order to simulate the accident, from the point where core damage is sustained up to containment failure, a computer code termed the integrated severe accident analysis code, has been developed. This is based on the Electric Power Research Institute's MAAP4 code, which has been used to evaluate severe accidents at other types of plant [93]. Accident progression analysis has been performed in detail using the integrated severe accident analysis code.

(ii) Assessment results

Level 1 internal events analysis is being performed on the basis of known results supplied by AECL and by reconstruction of the initiating events. If regrouping of initiating events is required to aid the efficiency of the work and ease review, additional analysis for defining the success criteria can be performed. Incorporating the CCFs in the system model and supplementing the human reliability analysis using a more detailed methodology are also essential. Therefore, the Level 1 PSA, which considers CCF and the detailed human reliability analysis, has been performed again. During the assessment, it was expected that the detailed human reliability analysis, and the use of a 24 h mission time could reduce the CDF compared with the results obtained from AECL analyses. As regards conservatism, credit was not given by AECL to the second human action, while the Republic of Korea study credited such action, incorporating consideration of the dependency between human actions. Even though such dependency increases the human error probabilities of each human action, in the overall aspect, such second human actions reduce the total CDF of the plant. In contrast, the CCF increases the total CDF by a significant amount. However, the CCF parameters used in the Wolsong PSA are the same as those used in PWR PSAs since there is no CCF data for HWRs. Therefore, it is difficult to be certain that the effect of CCF has been analysed accurately. However, the sensitivity results show that the CCF contributes about 39% to the total CDF. This result seems reasonable when compared with the results of typical PWR PSAs.

It is recognized that the most important initiating event is the loss of dual computers, as this affects all control functions such as reactor set-back, reactor control and steam generator level control. Since the transient events are the major contributor to total CDF, the importance of the shutdown cooling system and auxiliary feedwater system is higher than for other systems. The importance of the safety systems is relatively small compared with other systems.

The feasibility study was tentatively applied to several representative seismic sequences, adopting general assumptions in the seismic PSAs of other types of plant. The results indicated that it might lead to unacceptably high CDFs, owing to the seismic design features of HWRs. The approach is to be modified as follows:

— Equipment which is not seismically qualified is to be examined for its availability during or after an earthquake;
— Human error probabilities are to be specifically examined for each seismically induced sequence;
— Seismic capacity of relays used in non-seismically qualified electrical systems is to be estimated and, if required, incorporated in the plant model;
— Assumption of total dependency between redundant components needs to be checked in more detail.

The most significant events in terms of CDF have been found to be seismic events. Among the seismically induced initiating events, that causing failure of the

EWS structure is considered to be the most significant in terms of CDF. On the basis of sensitivity analysis and discussions with plant designers, some actions, such as improvement of the emergency diesel generator reliability, are recommended in order to reduce the risk due to external events.

In the Level 2 PSA, in view of the source term release characteristics, the limited core damage sequences are treated separately from the severe core damage sequences which cause core disassembly and moderator boil off resulting in calandria failure. No source term release is expected from the limited core damage sequences except for the negligible amount of fission products contained between the fuel pellets and the sheaths. The conditional containment failure probability after plant damage, for the internal events, is much lower than for the external events. The containment is apparently more effective in mitigating the internal and flooding events than the seismic and fire events. The late containment failure mode (i.e. a failure due to gradual pressurization around one day after the initiating event), is the dominant failure mode for all the initiating events.

(iii) Application of PSA during Wolsong plant design and operation

One of the major objectives of PSA is to provide risk informed input into the design, which is based on integrated logic models (event trees and fault trees) of the plant. This input is to be provided at conceptual design, detailed design and, eventually, at construction, commissioning and operation stages. Examples of the input to be provided include: incident studies; need for, and level of, redundancy and/or diversity; modelling dependencies between systems; demonstration that the rules of system independence have been successfully applied; and human interactions.

Development of emergency operating procedures: At the preliminary design stage, the licensee performed the study on the event sequence diagrams [94]. The objective of this study was to help define:

- Specific systems requiring availability analysis,
- Success/failure criteria for mitigating systems,
- Mitigating system mission times,
- System boundaries for fault tree analysis.

Event sequence diagrams are logical projections of the plant response following specific initiating events. They include information such as alarms and indications in the main control room, and plant response information with a time line up to the first operator action. From the plant response, allowable operator action times, and the relevant alarms and indications, event sequence diagrams are prepared for developing the input for the abnormal operating manual and the emergency operating procedures, as well as for event trees.

Development of accident management framework: The accident management framework planned for Wolsong plants involves the development and implementation of the accident management enhancements and the delineation of responsibilities within the plant organization for developing, implementing and maintaining the accident management procedures. This framework will involve the following steps: severe accident management guidance, accident management organization, strategy based accident management procedures, training, validation and implementation.

A particular requirement is to identify and assess candidate severe accident management strategies. It is also assured that new information on severe accidents is usefully defined and then practically applied in order to gain a better understanding of accident management. This information could be generated from the Wolsong PSA.

(d) PSA of Indian HWRs

(i) General

PSA offers a comprehensive and systematic means of safety assessment. A PSA study of the HWR design has been carried out and included a comprehensive procedure for the identification of dominating initiating events. Starting with the list of initiating events applicable to LWRs for licensing purposes, a number of these were supplemented to incorporate the design specific features of the HWR. On the basis of the analytical study of causes and consequences, a list of initiating events was prepared for detailed consideration. Fault trees for various process and safety systems were prepared, incorporating common cause considerations using partial beta factors.

An integrated approach to PSA applications covers three major areas: verification of the adequacy of design from the viewpoint of availability requirements, operational safety assessment, and management and regulatory applications. PSA has been extensively used in the design of nuclear power plants in order to identify the weak links in the systems, indicate suitable modifications and, in general, achieve a balanced design. During the PSA Level 1 study of Indian HWRs, a number of such modifications were implemented in various systems (e.g. ECCS, reactor building isolation system, moderator circulation system and SDS2) in order to improve system reliabilities, thus resulting in an overall reduction in risk.

In addition to the use of PSA in design evaluation for system safety and assessment of potential risk, PSA is also being employed in the operational safety management programme which includes assessment of the operational safety of the plant on the basis of the actual operating experiences (failure/maintenance data) of various components and systems, and utilization of this in decision making for safety issues. In order to assist the plant operations in such decision making, software termed ‘living PSA’ has been developed.

(ii) PSA in the design process

PSA has been extensively used as a tool for design evaluation, and its use has resulted in improvements being made to the availability of associated systems and to a reduction in the CDF. It is important to recognize that the use of PSA in identifying the weak links in the design of various systems in a nuclear power plant, is not significantly affected by the imprecision generally associated with the data. Some important modifications incorporated in the Indian HWRs, as a result of PSA studies, are:

- Comparative evaluation of different designs of the secondary shutdown system in order to obtain an optimum configuration from the viewpoint of simplified design, which results in high levels of availability, reduced cost and maintenance effort, and consistency with the requirements of system safety.
- Design modifications to a variety of ECCS components, in particular, to those components which account for the interdependence of various stages of injection, and to those which allow identification of components important to the improvement of procurement and test procedures, etc.
- Provision of isolation valves at the interface between the moderator circulation system and the liquid poison addition system in order to reduce the frequency of inadvertent loss of moderator (which forms the ultimate heat sink in an HWR).
- Availability improvements in the reactor building isolation system, which have been realized in the design of the heavy water condensate collection lines. In the reactor building isolation system, enough redundancy has been provided in the lines containing inlet and outlet dampers, which close and effectively isolate the containment in the case of initiating events involving activity release or pressure buildup in the containment. A significant availability improvement has already been achieved by the design modification in a subsidiary line as mentioned above.

In general, such analysis has resulted in a much better understanding of the availability requirements of various process and safety systems, CCFs, human error, etc., and has enabled achievement of a balanced design and the desired CDF.

(iii) Accident sequence quantification

PSA deals with identification and quantification of the dominant accident sequences of a nuclear power plant. Such accident sequences contributing to the CDF have been identified for the HWR, incorporating system dependencies and human reliability analysis using the SHARP methodology. Accident sequences initiated by a Class IV power supply failure leading to station blackout, and failure of the active process water cooling system coupled with unavailability of the firewater system, are major contributors to CDF. In the Indian 220 MW(e) HWRs, 3 × 100% emergency diesel generators housed in separate compartments with fire barriers have been provided and all diesel generator water jacket cooling lines are completely independent. In order to further reduce the CDF, one of the three diesel generators in a unit is located in the service area of the other unit, allowing independent cable routing, etc. In the case of station blackout, firewater can be injected into the steam generators over a prolonged period, with the help of redundant pumps driven by independent diesel engines. Long term reactivity requirements during extended station blackout are met with the help of gravity addition of boron into the moderator. The CDF is estimated as ~10^{-5}/a and this includes all categories of fuel damage. This is quite conservative since about one hour is generally available before the onset of core damage. The failure frequency of the active process water cooling system is dominated by pipe rupture, the failure rate of which, based on fracture mechanics, is quite low.

(iv) PSA in operational safety management

In addition to the use of PSA studies for generating information used for design modifications, many applications have been found in the operational safety management of nuclear power plants. In this connection, a Level 1 PSA study has been undertaken for an operating PHWR, based on operating experience, to assess plant safety and to:

- Assess the CDF;
- Identify any precursors;
- Confirm technical specification, validation and optimization of test frequencies;
- Evaluate the availability of various process/safety systems in order to check their adequacy.

Usually, the results of PSA studies are included in the plant safety reports which are available for general usage.The resulting inflexibility has, perhaps, restricted the benefits that could be realized by the plant personnel. In order to enhance the application of PSA on a continued basis, it is essential to make the results of any PSA study available to the plant operator for the purpose of guidance and to make the operator aware of the consequences of different possible plant configurations. Living PSA software has been developed to help plant personnel evaluate the importance of any component/system with regard to its contribution to system availability and CDF [95].

(e) PSAs undertaken in Argentina

A PSA project has been initiated for each operating nuclear power plant in Argentina, i.e. Atucha 1 and Embalse. Both projects have the following features:

- The PSA was the result of a requirement included in the operating licence, issued by the Autoridad Regulatoria Nuclear (National Regulatory Board of Nuclear Activities).
- The PSA was Level 1.
- The main objectives were to:
  - — Determine the estimated CDF,
  - — Verify the design and present conditions of all operation and safety systems,
  - — Verify all operating procedures for abnormal events,
  - — Detect weak points in systems and procedures and to propose improvements,
  - — Quantify the impact on the CDF of plant and procedure modifications,
  - — Reassess system and equipment test and maintenance tasks.
- The whole project was divided into phases.
- The scope of phase 1 took account of:
  - — Initial states (limited to high power operation only),
  - — Initiating events (limited to internal events only),
  - — Radioactive sources (core and coupled refuelling machine),
  - — Mission time (normally 24 h).

The main specific aspects of each project are summarized in the following sections.

(i) Atucha 1 PSA

This project was initiated at the end of 1992 and phase 1 was completed in mid-1998. As Atucha 1 is a prototype and the only unit of its type (pressure vessel HWR) operating in the world, significant effort had to be devoted to the identification of all postulated initiating events through complementary approaches, and to the establishment of a comprehensive set of deterministic simulations of accident sequences.

The initial version of the whole model was obtained in early 1996 and an IAEA mission carried out a comprehensive review of this initial, quantified model. On the basis of the mission's observations, parallel PSA findings, proposals for plant and procedure modifications, and subsequent implementation, a more refined model was obtained, with and without operator recovery actions.

Most of the implemented plant hardware modifications required very small financial outlays, while the total scope of operating procedures for abnormal events was enlarged. Through the implementation of all these measures, a significant reduction in the final CDF for phase 1 has been obtained.

Phase 2 has already been initiated and is on-going. It covers various major PSA applications and further plant safety improvements, inclusion of fires as initiating events and extension of the initial plant states considered (to low power and shutdown plant states). A rough, preliminary estimate indicates that the impact of fire events on the final CDF would be quite significant, in the absence of corrective action being taken on plant hardware. Nevertheless, such corrective action seems feasible in many cases.

(ii) Embalse PSA

This project was initiated in 1997 and phase 1 is currently in progress. The fact that Embalse is a CANDU type nuclear power plant facilitates project advances in each of the main tasks.

The identification and grouping of initiating events and the production of a first version of all event tree models for the postulated group sequences have been completed. Fault tree modelling of system failures is currently under way.

(iii) Special support

The central PSA team was constituted by the utility, Nucleoeléctrica Argentina S.A. The team comprised personnel drawn from each nuclear power plant and specialists from other areas of the company, with additional support provided by specialists from CNEA.

Important support from the IAEA has been received in different forms, particularly through the visits made by experts or through the review of PSA activities by international experts. The Embalse PSA included the valuable participation of specialists from Cuba and, especially, from Romania; the latter having previously carried out the Cernavoda PSA.

(f) Cernavoda PSA

(i) Level 1 PSA

The Romanian regulatory body requested that a Level 1 PSA be carried out for unit 1 of the Cernavoda nuclear power plant. The results of the PSA were to be included in licensing documentation.

The last revision of the Cernavoda PSA project was co-ordinated by the owner 'Nuclearelectrica' ((SNN), Safety and Licensing Compliance) and was jointly performed by the Centre of Technology and Engineering for Nuclear Projects (CITON) Bucharest-Magurele, and the Institute for Nuclear Research Pitesti.

Work on the Cernavoda PSA has been assisted by the IAEA and was the subject of two IAEA International Peer Review Service missions (1990, 1995) and a pre-International Peer Review Service mission in 1994.

The main objectives of the project were to:

- Provide a thorough safety review of the plant design and operation and to identify the most suitable areas for improvement;
- Identify the main contributors (initiating events, hardware and human failures) to the frequency of the different plant damage states considered.

The last revision of the project's scope includes an assessment of all internal initiating events leading to possible radioactivity releases from the reactor core, starting from normal plant operation (full power). The range of potential plant damage states incorporates nine categories, from early core damage to tritium release. The PSA project models the plant on the basis of the 'as built' design documentation.

The approach used for the selection of the initiating events consisted of constructing a master logic diagram to identify potential causes that can lead to the displacement of radioactive materials from their normal location. Using this method, 114 preliminary initiating events were identified. Using the decision tree method (considering factors relevant to safety and safety related system status, degree of operator involvement in mitigating the accident, etc.), the preliminary initiating events were grouped into 40 initiating event categories (final initiating events). Each final initiating event category contains preliminary initiating events producing a similar plant response. An event tree was developed for each final initiating event.

The selected event trees method is summarized thus, 'small event trees, large fault trees'. The event tree headings refer to front line systems or functions and human decision errors. The support systems were considered at the fault trees level. The included human decision errors were quantified using the ASEP method, following the SHARP 1 framework.

Fault trees were developed to determine the frequency of the initiating events and to model the front line systems and support systems. The basic events considered include component failures and unavailabilities (testing and maintenance), operator errors (pre-accident, post-accident) and CCFs. A total of 36 systems were modelled.

Quantification of the CCFs was achieved using the generic 'beta factor' (screening stage) and 'multiple greek letter' (detailed analyses) methods.

The human reliability analyses followed the SHARP 1 framework. The human errors (pre- and post-accident) considered in the fault trees were quantified in a standardized mode using the decision tree method. The decision tree data were considered on the basis of the ASEP method.

Owing to limited Cernavoda operating experience, the component reliability parameters considered were mainly derived from generic data sources. The reliability data are stored in an appropriate database, which has the capability of being periodically updated with plant specific data.

The accident sequences quantification (project integration) was achieved using the fault tree merging method. The accident sequence (minimal cut sets) was produced, as well as the occurrence frequencies. Importance analyses (risk achievement worth, risk reduction worth, and Fussel and Vesely) were performed in order to evaluate the contribution of different items to the overall results. Sensitivity studies were undertaken to evaluate the impact of major modelling assumptions.

The PSA study was developed and solved using the PSAPRO code package, a CITON in-house software package. The software includes several modules (i.e. reliability database, initiating events selection and grouping, event tree development and processing, fault tree development and processing, and the integrated project). The software is undergoing internal validation and an IAEA expert meeting was organized in 1998 for its evaluation.

The PSA project scope is being extended to include some internal and external hazards, and assessments of internal fire and flooding, as well as earthquakes, are now in progress. The Cernavoda 1 plant staff is planning to review the model, on the basis of its operating status, and to develop PSA applications for specific needs.

(ii) Level 2 PSA

The activity conducted up to now in this area at CITON has been intended to develop the capability to extend the existing Level 1 PSA to Level 2 PSA.

A pilot study was carried out using generic PSA models, in order to evaluate the resources, information and software needed to perform the full scope Level 2 PSA for the Cernavoda plant. In this study the following steps were performed:

- Preliminary definition of objectives and scope,
- Preliminary plant damage states logical diagram,

- Test study to extend KEMA Level 1 event trees to Level 2,
- Cernavoda nuclear power plant containment data collection,
- Preliminary containment event trees development (using CONEDIT software),
- Preliminary source term categories logical diagram,
- Test study to perform sensitivity and results interpretation.

The work is being developed further with IAEA assistance and based on the Cernavoda 1 Level 1 PSA model. The following steps are now in progress:

- Definition of objectives and scope,
- Quality assurance programme development,
- Methodological development,
- Severe accident code procurement.

5.3.2.3. Phenomenology of severe accident sequences

The following sections describe the phenomenology of severe accident sequences of HWRs. Although the discussion attempts to be generic, details will differ from plant to plant, especially in the case of station blackout.

(a) Station blackout

The simultaneous loss of all AC power sources (i.e. all off-site power, the main generator and all backup sources of emergency AC power), which normally energize the safety related buses, is termed a station blackout [96].

Modern HWRs have two, independent, spatially separated sets of Class III diesel generators, belonging to Group 1 and Group 2 respectively. These reduce significantly the frequency of station blackout. The Group 2 diesel generators form the emergency power supply. In these HWRs, station blackout is a residual risk sequence which is addressed in the PSA as being one contributor to the CDF. The rest of this section, therefore, applies only to plants with one set of Class III diesel generators. It is, therefore, relevant to early HWRs and to Indian HWRs and is included in order to illustrate how station blackout is mitigated in such designs. Design details discussed in this section are based on Indian HWRs.

A postulated station blackout would start with the failure of normal power supplies (Class IV power) derived from the grid and/or the station generator, coinciding with the unavailability of the on-site backup power sources (typically emergency diesel generators or gas turbines) to the safety related Class III buses. The battery backed Class II and I power supplies, including control power supplies, would remain available until the batteries were exhausted, typically within 1–2 h. The capability of an HWR station to handle this event has been assessed with respect to reactor shutdown and long term subcriticality, core decay heat removal and reactor component cooling/moderator cooling.

(i) Reactor shutdown and long term subcriticality

On failure of the Class IV power supply, the reactor will trip on receipt of a signal indicating, for example, the absence of an electricity supply to the HTS pumps or low flow. The backup trip will be triggered on receipt of an HTS high pressure signal. Prompt reactor shutdown will be effected, as usual, by one of the two, fast acting shutdown systems. In order to maintain long term subcriticality with an adequate margin (≥10 mk subcriticality margin), the Indian 220 MW(e) HWRs employ a liquid poison injection system which automatically injects boron poison into the moderator. In all other reactors, the worth of SDS1 and SDS2 each is adequate for long term subcriticality; in the very long term (days), SDS1 is supplemented by manual addition of poison to the moderator.

The reactor trip logic and shutdown systems are fail-safe with respect to failure of control power supplies.

(ii) Core decay heat removal

During the blackout, HTS forced circulation will be lost. Also, the HTS pressurization pumps (D_2O feed pumps) will not be available. The fuelling machine supply pumps, which are on Class II power, may continue to operate if they are already operating, but they will not start. On the secondary side, the main steam generator feed pumps, as well as the auxiliary steam generator feed pump(s), will be unavailable.

HTS circulation during blackout is provided initially (i.e. in the first 90 s or so) by coast down of the pump flywheels, followed by thermosyphoning. On the secondary side, the water inventory of steam generators provides a heat sink capability for a limited period by blowing steam through ASDVs and electromatic relief valves. Subsequently, water from an emergency supply such as the firewater system must be fed into the steam generators. Before this can be done, the steam generators need to be depressurized by operator action to a pressure (around 3–7 kg/cm^2) at which backup water injection can take place. The initiation and completion of steam generator depressurization must be done before the limited steam generator water inventory is exhausted. The current operating procedures for handling station blackout in Indian HWRs call for initiation of crash cooling (opening of all ASDVs) within six minutes of station blackout starting.

Some of the ASDVs (with a discharge capacity ~20% of the full steam flow) in Indian HWRs perform a fail-safe function as they are designed to open on failure of the control power supply or air supply, thus ensuring a path for steam discharge.

Ensuring continued core cooling under blackout conditions requires adequacy of both the firewater inventory and the water inventory on the HTS D_2O side.

The firewater sump inventory is a plant specific feature that would typically last at least 12 h with no diversion; replenishment provisions also exist, e.g. from the natural draft cooling towers basin (fed by gravity).

On the HTS side, there is shrinkage due to cooldown and, potentially, as a result of system leaks. In the Indian 220 MW(e) HWR design, the initial make-up requirement during cooldown will be met by the ECCS D_2O accumulator which will start supplying coolant when the HTS pressure falls to 55 kg/cm^2. With this provision, the HTS system will stay solid initially but may become two phase some hours into the incident, depending on system leaks. In the 500 MW(e) HWR design, the pressurizer provides the initial make-up requirement during part of the cooldown. Subsequently, two phase conditions will prevail until the pressure drops to below about 2.0 kg/cm^2, at which point the inventory from the HTS storage tank can start entering via the feed lines, bringing the system back to solid conditions. The maximum estimated void in the system is 14% by volume. Thermosyphoning continues to be effective with such voids in the system.

The HTS instrumented relief valves will remain closed until pressure in the local instrument air receivers becomes available. Subsequently (at about 30 min), the instrumented relief valves will open, in which case the bleed condenser will fill and form part of the main HTS pressure boundary, protection against overpressure being provided by spring-loaded relief valves located on the bleed condenser.

The assessment performed with respect to core cooling during the Narora 1 incident in March 1993, which resulted in an extended station blackout lasting 17 h, indicated that thermosyphoning flow in the HTS continued to cool the fuel and that fuel cladding temperatures were well below the normal operating values. Samples taken from the HTS after the incident and during subsequent operation have shown no abnormal rise in iodine concentration, further confirming the integrity of the fuel.

(iii) Reactor component cooling/moderator cooling

During blackout, the cooling and circulation in the moderator system, as well as in the end shield and vault water circuits, will stop. At the same time, heating of the moderator and reactor components will continue owing to decay gammas, as well as by heat transfer from the HTS (which continues to operate at an average temperature of about 125°C, even after depressurization of the steam generators).

Under these conditions, the various components and fluids (D_2O moderator, end shield cooling water, calandria vault water, calandria and end shield structures) will gradually heat up.

Current procedures for handling station blackout call for the injection of firewater into the end shields about one hour into the incident in order to limit the temperature rise of various components and fluids. No additional provisions for the cooling of reactor components or the moderator seem to be required for handling station blackout.

(iv) Other requirements

Deuterium buildup in the moderator cover gas: During blackout, deuterium generation in the moderator will continue owing to decay gammas, whereas for the cover gas, helium, circulation and recombination would have stopped, possibly leading to a buildup of D_2 concentration in the moderator cover gas. The operating procedure invoked in the event of station blackout calls for periodic purging of the system.

Monitoring of safety parameters: After the control power supplies are exhausted, indications in the control room will be lost. In order to monitor important parameters such as reactor power; HTS pressure and temperature; steam generator level; containment pressure, temperature and radiation level; terminations of cables from sensors to control equipment room have been identified (where appropriate read-out arrangements can be hooked up at the junction boxes).

Requirements on turbine generator side: As regards the turbine generator, the essential requirements to be considered during blackout are the prevention of hydrogen leakage from the generator and the lubrication of bearings in the turbine generator assembly while the machine is running. During the initial period (typically 1–2 h) when Class II power supply is available, both of these functions will be met through the DC seal oil pump and the DC lubrication oil pump respectively. Before this period ends, operator action will be required to depressurize hydrogen from the generator. Also, as an emergency action, turbine rolling can be stopped after breaking the condenser vacuum.

Thus, the design provisions of the HWR permit it to cope with an event such as a station blackout, as far as safe shutdown of the reactor, cooling of the core to maintain fuel integrity, and heat-up of moderator and core components are concerned.

(b) LOCA plus coincident LOECC

One of the classes of postulated events considered in HWR licensing analysis is a LOCA coincident with the failure of the ECCS to operate on demand.

Analyses of LOCA/LOECC sequences focus on demonstrating that the regulatory dose limit has been met and on verifying that the safety design target of maintaining the integrity of 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 gross geometry of the reactor core is well defined and can be analysed on a channel by channel basis in order to provide estimates of the timing and extent of fission product release and hydrogen generation.

(i) Large LOCA plus coincident LOECC

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 reactor power is reduced to decay power levels within seconds. The HTS depressurizes at a rate determined by the break size. Severely degraded fuel cooling and significant activity release result if it is assumed that the ECCS fails to operate on demand. Figure 108 illustrates the analysis methodology.

Under such conditions, fuel heat-up leads to fuel deformation and may cause pressure tube yielding. The coolant pressure at the time of overheating determines the mode of pressure tube deformation. High pressures (greater than approximately 1 MPa) lead to pressure tube ballooning and, at 16% pressure tube strain, to circumferential contact between the pressure tube and the calandria tube in the overheated region. At lower pressures (near atmospheric), pressure tube sag is more prevalent and leads to a 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 circumferential contact between the pressure tube and the calandria tube. In all cases — sag, strain, or no contact (for regions of low 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. The detailed assessment shows that over the entire range of large break LOCAs with emergency core cooling unavailable, gross fuel temperatures do not reach the melting point of UO_2.

An assessment is performed to ensure that channel integrity is maintained. 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 from fuel element contact, followed by rapid local strain to failure. HWR safety reports indicate that for the range of contact conditions expected to occur between the fuel and a pressure tube, local pressure tube

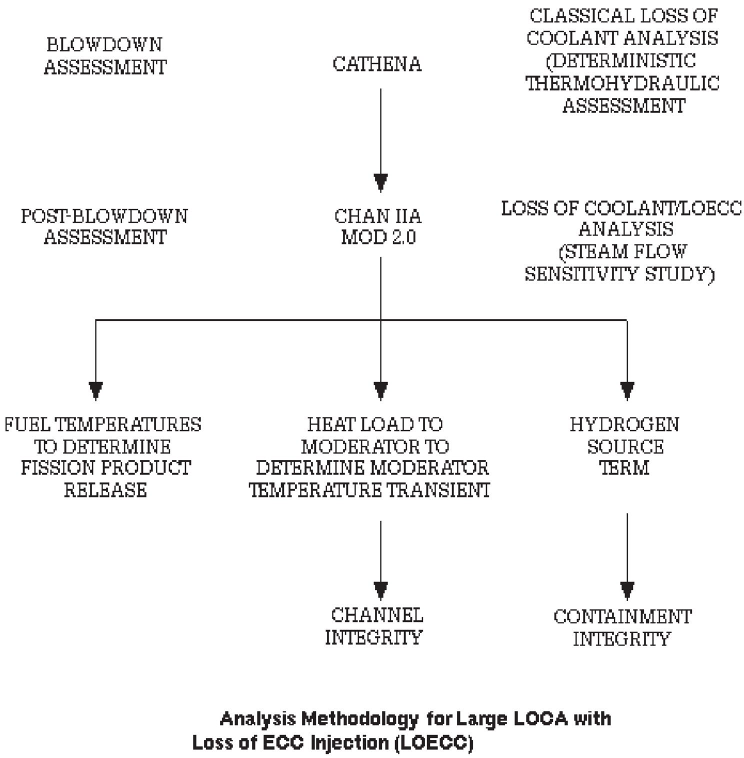


FIG. 108. LOCA plus LOECC methodology.

overheating will not be severe enough to cause localized overstrain and failure prior to pressure tube/calandria tube contact.

The prevention of sustained calandria tube dry out following pressure tube contact is a condition sufficient to maintain pressure tube integrity, since it ensures that the fuel channel will not suffer further strain. The factors which affect the potential for calandria tube dry out 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 is 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). An assessment is performed of the transient, local moderator subcooling required to prevent calandria tube dry out anywhere in the core. This required level of moderator subcooling is then compared with predictions of the moderator subcooling available during the accident. Assessment of the transient spatial variation of the moderator subcooling throughout the accident includes consideration of 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, including the feedback effect of pressure tube and fuel deformation on the distribution of steam flow in a channel, and consequently on the thermal behaviour of the fuel and pressure tube. Pressure tube ballooning and/or fuel bundle slumping promotes the bypass of steam flow around the interior of the fuel bundles in a channel. 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 heat-up; the extent of the exothermic Zircaloy–steam reaction may be further reduced owing to relocation of the molten Zircaloy 4 sheath material.

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. If the oxygen content of the molten Zircaloy is high, then the melt wets the UO_2 easily and tends to collect in pellet cracks and dishes. If the oxygen content of the molten Zircaloy is low, the melt does not easily wet the UO_2 and therefore it tends to collect along the outer surface of the element. The results of experiments on fuel bundle behaviour at temperatures in excess of 1900°C demonstrate this type of melt relocation behaviour. It has been demonstrated that contact of this eutectic with the pressure tube does not threaten pressure tube integrity.

Flow bypass due to pressure tube and fuel bundle deformation, and molten Zircaloy relocation, are both mechanisms which effectively reduce the overall rate of Zircaloy–steam reaction in a channel. This exothermic reaction is an important source of heat, which increases fuel temperatures under severely degraded cooling conditions and also determines the timing and extent of hydrogen evolution from a channel.

The fuel temperature transients generated are used in the assessment of fission product release. The distribution of active fission products within the fuel under normal operating conditions, the timing and extent of sheath failure, and the transient release of fission products from the fuel are assessed. 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 as a result of the reaction with molten Zircaloy, and long term leaching release from the failed fuel in water.

As noted, hydrogen is produced by oxidation of the sheaths and part of the pressure tubes. The 'worst' case is not a total LOECC, since that would starve the Zircaloy–water reaction and reduce both fuel temperatures and the amount of hydrogen produced. Thus, in safety analysis, the hydrogen produced is maximized by arbitrarily setting the channel flow to a value which maximizes the chemical reaction and minimizes convective cooling, typically around 10 g of steam per second per fuel channel (see Fig. 109). Even so, the hydrogen concentration in a typical single unit CANDU 6 after a large LOCA plus LOECC rises only briefly above the lower flammability limit in two rooms and never approaches the deflagration to detonation transition (Fig. 110).

(ii) Small LOCA plus coincident LOECC

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

In the case of small LOCAs coinciding with the failure of the ECCS, the fuel channels would receive adequate single phase liquid or two phase coolant until well after reactor trip. Eventually, owing to the unabated loss of coolant inventory from the HTS, 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. There is also a substantial liquid level in the channel which contributes to effective cooling. Eventually, as the liquid in the channel boils off, the fuel is cooled by a decreasing flow of steam and fuel temperature excursions commence.

Slow boil off in a fuel channel may result in temperature variations occurring 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 overstrain 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. The results of analyses indicate that localized overstrain failure resulting from thermohydraulically-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/LOECC scenarios. As in those cases, there is no gross fuel melting in any small LOCA/LOECC.

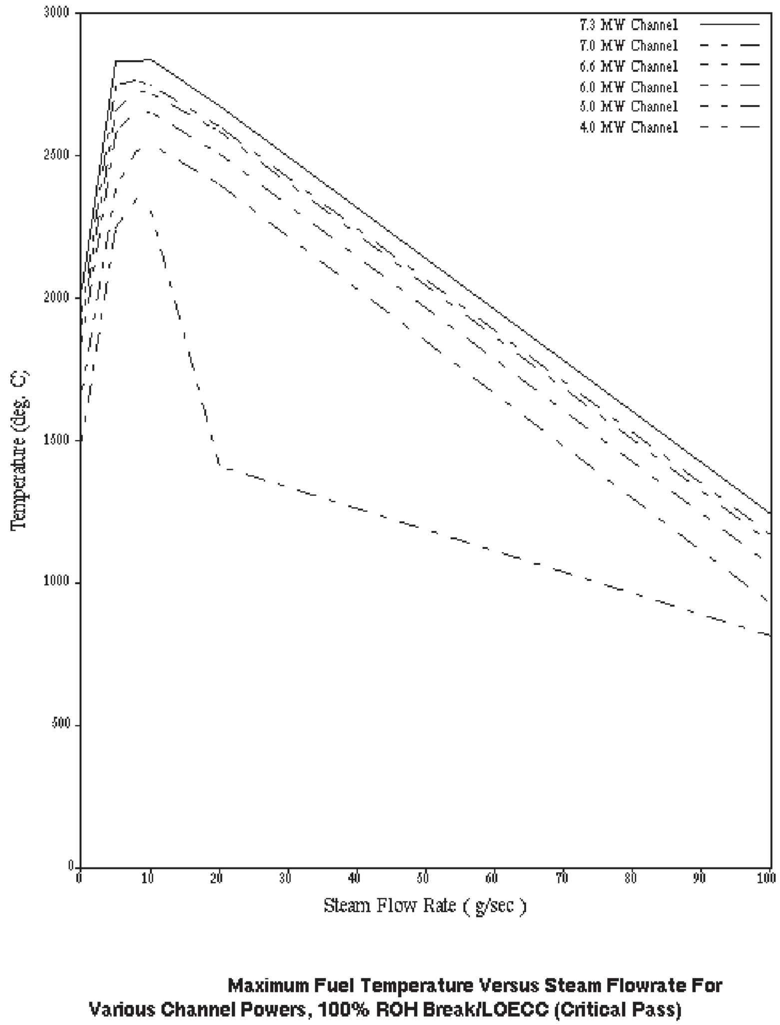


FIG. 109. Fuel temperature sensitivity to assumed channel steam flow rate.

(c) Containment impairments

The containment system comprises several subsystems. The requirement to design for dual failures means that combinations of process system and containment failures must be analysed. However, the independence which is designed into

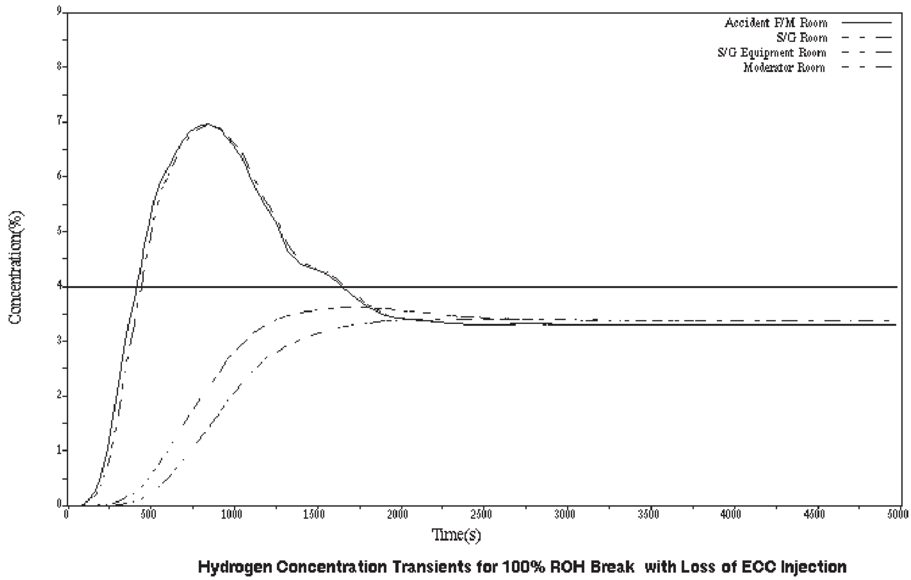


FIG. 110. Hydrogen concentration in containment for LOCA/LOECC (worst steam flow rate).

containment subsystems permits impairments of them to be considered rather than total containment system failure.

The subsystems which could be impaired are containment isolation, containment atmospheric cooling, dousing and airlock door seals. Vacuum containment stations have additional impairments, such as failure of a bank of self-actuated relief valves to open or close, and failure of an instrumented pressure relief valve to operate. Isolation impairments include failure of the ventilation inlet or outlet dampers to close, and failure of isolation logic which implies that both inlet and outlet dampers fail to close. Failure of containment atmospheric air cooling could arise from a loss of the support services to the coolers (process water and electrical power). Owing to the separation of dousing into two subsystems in some HWRs, a failure of both dousing subsystems is highly unlikely, but this eventuality has been examined in certain cases. The personnel and airlock door seals are normally inflated when the doors are closed. However, to take account of the possibility that the airlocks have been recently opened and closed and that the seals have not been re-inflated, deflated door seals are considered.

Therefore, for single unit HWRs, the containment subsystem impairments considered for the dual failure analyses are:

- Failure of an isolation damper,
- Failure of containment isolation logic,
- Failure of air coolers,
- Failure of dousing,
- Deflated airlock door seals.

5.3.2.4. Severe core damage sequences

In the case of residual risk sequences in which the moderator is also assumed to be unavailable (e.g. LOCA plus LOECC plus loss of moderator cooling), the fuel channels would gradually fail as the moderator boiled off and collapse to the bottom of the calandria. The degradation of an HWR core with no cooling and gradual boiling off of the moderator has been characterized [97]. The uncovered channels heat up and slump onto the underlying channels. Eventually, the supporting channels (still submerged) collapse and the whole core, still almost completely solid, slumps to the bottom of the calandria (Fig. 111). An empirically based mechanistic model has been

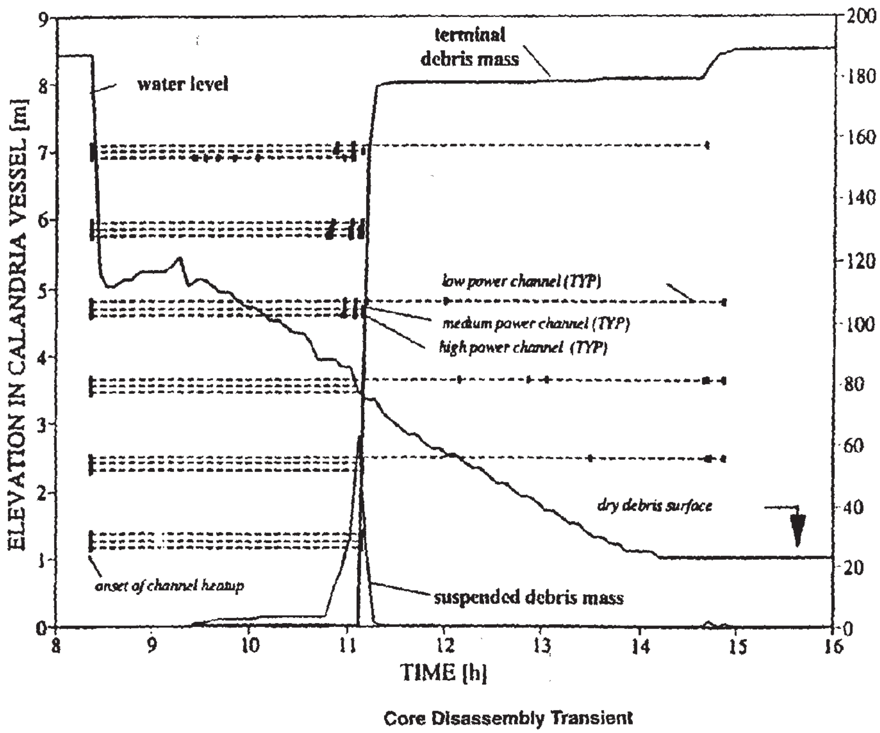


FIG. 111. Core disassembly transient.

developed which shows that the end state of core disassembly consists of a bed of dry, solid, coarse debris, irrespective of the initiating event or the core disassembly process [98]. Heat-up is relatively slow because of the low power density of the mixed debris and the spatial dispersion provided by the calandria shell, with melting beginning in the interior of the bed about two hours after the start of bed heat-up. The temperatures of the upper and lower surfaces of the debris remain well below the melting point (Fig. 112) and heat fluxes from the calandria to the shield tank (or calandria vault) water are well below the critical heat flux at the existing conditions (Fig. 113). The calandria wall can, therefore, prevent the debris from escaping. Should the shield tank water not be cooled, it will boil off and the calandria will eventually fail by melt through, but this will not occur in less than about a day, giving ample time for operator action to be taken, such as flooding the shield tank or calandria vault with water drawn from emergency supplies. If the calandria does fail, the debris will fall onto the vault floor, which is composed of concrete in single unit HWRs. Interaction of the debris with the concrete, leading to possible containment failure, would again be very slow.

Factors that contribute to the predicted effectiveness of cooling a degraded core in an HWR, in addition to the inherent heat sinks provided by the separate moderator

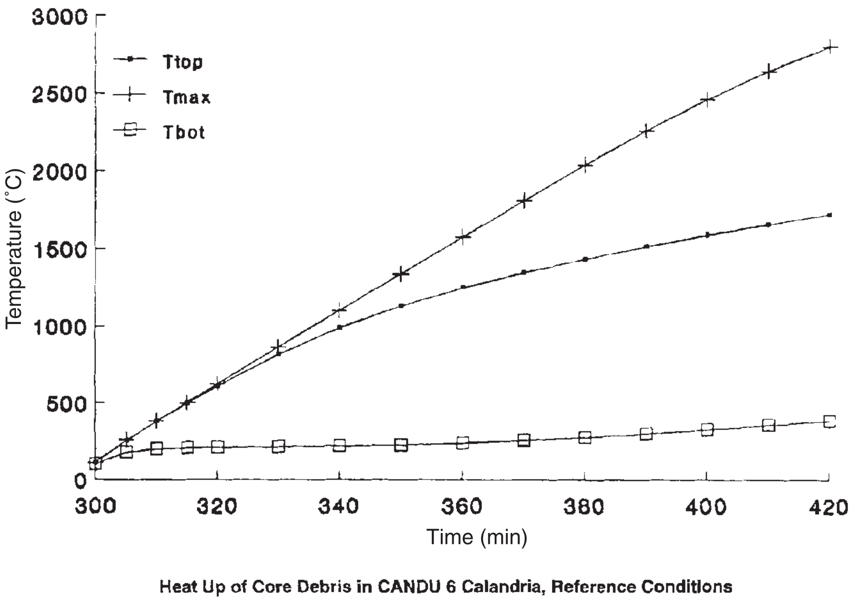


FIG. 112. Heat-up of core debris in a severe core damage accident.

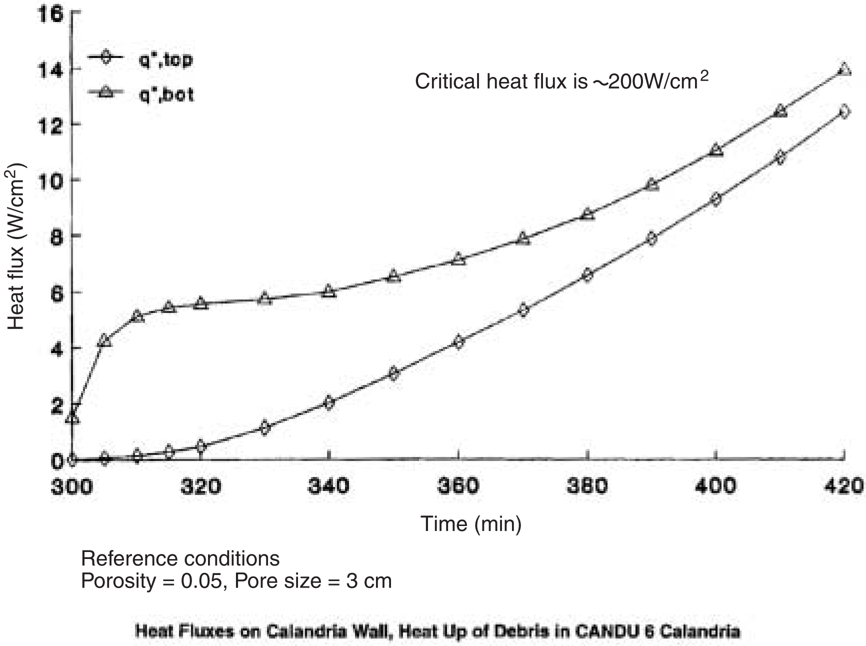


FIG. 113. Predicted heat fluxes on calandria wall.

and the shield tank water, are the low power density (about 16 W/g of fuel in a CANDU 6, based on the design power) and the extensive dispersion of the debris in the calandria, resulting in a shallow molten pool depth of about 1 m maximum, and about 0.65 m average, for the reference case.

Sensitivity studies were performed on these results and took into account the effect of assuming that the debris was contained in a pressure vessel geometry [99]. It was concluded that:

- The melt thermal conductivity is the most sensitive variable for the parameters of interest such as the steel temperatures and the heat flux to the surrounding water, although the relative effects are small.
- Calandria and vessel geometries give similar results for the same *assumed* core power density, with the exception of the steel container temperatures, which are much higher in the vessel geometry owing to the thicker vessel wall.
- The Raleigh number, Ra, which governs natural convective heat transfer from the melt to the vessels walls, is relatively low in HWRs ($<2.9 \times 10^{13}$).

Therefore, existing correlations up to a Ra value of 10^{13} for corium melts are expected to apply with confidence to HWR geometries.
- On the basis of the geometries analysed, a large fraction of the total heat (about 80% for calandria geometry versus 72% for the vessel geometry) of the molten pool is radiated upwards from the top surface. The difference in radiated heat is due to the lower pool height and larger radiative surface area of the calandria geometry.

The severe core damage sequence in HWRs suggests several fairly simple design mitigation options. Clearly, an independent source of water make-up to the moderator could, if initiated early enough, prevent failure of the channels and development into severe core damage. Alternatively, water make-up to the shield tank could cool the debris and prevent calandria shell failure. However, make-up itself is not enough; water cannot be allowed to build up inside the reactor building indefinitely, and the decay heat is removed by steaming into the building. Therefore, in the long term the containment pressure must be controlled. Design options in the longer term, therefore, include removal of decay heat from the containment atmosphere by air coolers and return of the condensed water on the floor to the shield tank or manual provision of a redundant pumped water supply, possibly from a portable external source, to the shield tank and end shield cooling system. Such options are discussed in Section 5.4.2.3.

5.4. SAFETY ENHANCEMENTS UNDER WAY FOR CURRENT GENERATION HWRs

Current HWRs, both those operating and current designs, are at a mature stage of development. As indicated previously, safety characteristics are generally well known. Safety is achieved by a combination of active and passive components and systems. Shutdown mechanisms are passive, relying on gravity or stored energy. The initial stage of emergency core cooling relies on stored energy in some designs. Decay heat removal, however, relies on conventional pumps, valves and electrical power, as well as an assured source of cooling water as the ultimate heat sink. Containment heat removal is likewise a mixture of passive systems (dousing and cold structures for short term heat removal) and active coolers and fans for long term heat removal, as well as powered (but fail-safe) redundant ventilation dampers and valves for containment isolation, and powered igniters for hydrogen control where required.

Safety enhancements for current generation HWRs fall into three categories, those that:

- Address operations related safety issues in operating stations,
- Address regulatory action items both for operating stations and for current designs,

- Simplify and enhance the effectiveness of safety related systems in current designs.

It is not the purpose of this section to provide a complete description of any particular category, but highlights are provided.

5.4.1. Operating stations

A number of examples were selected in order to indicate the type of safety related upgrades being made to existing HWRs. Routine upgrades are not discussed; instead, selected areas of interest involving major upgrades are dealt with. The list is illustrative and not complete.

5.4.1.1. Fire protection

The two group philosophy has been a robust approach to mitigate the effects of fires. In addition, national and international fire protection standards have been continually updated, and now call for a systematic analysis of fire hazards early in the design of new plants. However, some HWRs were built before these approaches were fully implemented and some jurisdictions impose further regulatory requirements.

(a) India

The fire at Narora 1 in 1993, resulted in several modifications and improvements being made to both prevention and mitigation aspects of fire protection, including:

- Re-routing of cables for power supplies to achieve improved segregation of redundant trains of supplies;
- Strengthening of existing fire barriers and provision of additional fire breaks and fire barriers;
- Improvement of periodic surveillance of fire barriers;
- Erection of a fire wall between the fire prone areas of the turbine generator and the adjoining passage through which several cable trays run;
- Making provision for improving control room habitability in the event of a fire in the turbine hall, including making modifications to the control room ventilation system.

In addition, the fire served to emphasize the importance of providing for connections in the design, in order to allow make-up water to be added manually (through the firewater system) to the steam generators and end shields.

(b) Romania

During the period 1990–1992, a fire hazard assessment for Cernavoda 1 was performed by CITON. The methodology used was set up in accordance with the guidelines/recommendations provided by specialists from the Institut de protection et de sûreté nucléaire (IPSN) (Paris), and from Electricité de France.

The Romanian fire protection standards requirements with regard to the fire resistance criteria, fire occurrence and passive means replacement criteria were also taken into account. Fire severity parameters (i.e. rate of temperature increase, maximum temperature and fire duration), were established according to the ISO 834 standard and the IPSN 284 report.

In the first stage of this assessment (the fire areas screening), fire load densities, safety related systems routes and design fire protection measures were settled or identified. Two screening criteria were employed: 200 MJ/m^2 for the fire load density, and the presence in the fire area of safety related systems.

The second stage (the analysis proper) included the determination of the expected fire duration and the maximum temperature reached, the description of the rooms in terms of process systems, construction and electrical systems, and the fire consequences.

On the basis of the findings of this analysis and on further studies, which focused on the determination of the best fire protection methods and materials for structures, openings, equipment and electrical cables, and on the provision of adequate sealing for openings as well as the determination of the performance of fire detection and extinguishing systems, SNN and AECL have implemented several actions for Cernavoda 1 in order to improve the fire protection. These actions comprise:

- Protection of steel structures and cables by coating with intumescent paint, asbestos cement and glass fibre reinforced gypsum;
- Maintenance of penetration openings by coating with gypsum milk;
- Enhancement of fire resistance of doors by coating with intumescent paint;
- Addition of a connection, directly outside the firewater ring, for the sprinkler system located in the cable spreading room;
- Installation of additional fire detection and extinguishing systems in those rooms having a substantial fire load density (i.e. cable spreading room, new fuel storage room, dirty/clean oil tank room).

5.4.1.2. Environmental qualification

Current HWRs have an environmental qualification requirement and programme that ensures that equipment credited with mitigating an accident is qualified to operate in the accident environment, and is maintained so as to keep this qualification. In the case of older plants, this was not done systematically or completely, and for a number of operating HWRs a backfit programme is under way in order to establish, or improve, the environmental qualification of their equipment.

5.4.1.3. Ageing

Most operating HWRs now have an ageing management programme in place in order to monitor and correct the effects of ageing.

(a) Indian programme

The major elements of the Indian programme involve:

- Identification of systems and components important to safety and for which ageing needs to be considered;
- Understanding the dominant ageing mechanisms in the selected components;
- Monitoring of ageing, i.e. detecting component degradation before failure;
- Timely mitigation/replacement.

Some features of the HWR with respect to ageing management are as follows:

- With the exclusion of equipment inside the calandria vault, all other key equipment is amenable to monitoring, refurbishment and replacement. The equipment inside the calandria vault (calandria, end shields and moderator system piping inside the vault) consists of low pressure, low temperature systems made of stainless steel which are resistant to radiation induced or other forms of degradation. Failure, if any, in these systems would not be of a catastrophic nature and would not pose a radiological risk to the public. However, there could be an economic penalty since the equipment is hard to access.
- By employing appropriate design specification and material selection, a minimum design plant life of 30–40 years has been ensured for all major equipment other than Zircaloy 2 pressure tubes. In the latter case it is recognized that replacement will be required after 12–15 years. Zircaloy 2 pressure tubes were used in Indian HWRs up to Kakrapar 1. Subsequent units use zirconium–niobium, which has a longer working life.

The Appendix gives details of the programme, on a component basis, for Indian HWRs.

TABLE XI(a). LIFE MANAGEMENT STRATEGIES: INDIAN PROGRAMME ON AGEING

Component/ageing concern	Monitoring/in situ inspection	Prevention/mitigation
Pressure tubes:		
Service induced flaws	Surface and volumetric inspection	Replacement, if fitness for service criteria are not met
Irradiation growth, axial elongation		Creep adjustment
Blister due to calandria tube/PT contact	Monitoring of creep sag, spacer displacement	Spacer repositioning, replacement (if required)
Degradation of material properties: Hydrogen ingress Fracture toughness	Scrape sampling and post-irradiation examination	Replacement (individual channels or en masse)
Leakage from pressure tube or rolled joint	Annulus gas monitoring	Leak handling, replacement
Steam generators:		
Tube leaks: Activity release D_2O loss	Tritium monitoring In situ inspection of tubes	Prevention: Chemistry control Sludge lancing Blowdown Mitigation: Plugging of defective tubes Replacement of steam generator when plugging exceeds design margin
End shields (RAPS, MAPS 1):		
Leaks in irradiation, embrittled end shields (3.5% nickel steel)	Leak monitoring	Prevention: Avoidance of thermal shock by following specified heat-up/cooldown cycles
(Not applicable to MAPS 2 onwards, where stainless steel end shields are used)	Condition monitoring by acoustic emission	Mitigation: Plugging of crack (RAPS 1 method)

TABLE XI(a). (cont.)

Component/ageing concern	Monitoring/in situ inspection	Prevention/mitigation
HTS system piping:		
Thermal cycle fatigue Corrosion Erosion	Volumetric and surface examination as per ASME Section XI requirements	Repair/replacement
Moderator inlet manifold (RAPS, MAPS):		
Mechanical failure of manifold leading to damage of calandria tubes, and undesirable changes in moderator flow pattern in calandria. The plate type manifold in the calandria at the moderator inlet end was partially damaged in MAPS 1 and 2, probably due to impingement of moderator at higher velocity during fast pump-up.	VT of manifolds using CCTV through calandria lattice tube (with calandria tube removed)	Prevention: Fast pump-up feature deleted Mitigation: Modification in moderator flow arrangement as in MAPS
Moderator heat exchangers:		
Tube failure: D_2O loss and tritium release	Tritium monitoring on process water side of HXs N-16 monitoring on process water side In situ inspection of tubes	Tube plugging HX replacement when plugging exceeds design margin
Safety and relief valves:		
Failure of valve components	Testing/recalibration periodically as per technical specification requirements	Repair/replacement as required
Emergency diesel generators:		
Reduction in capacity or reliability	Functional tests as per technical specification for station operation Reliability assessment based on tests/demands Periodic full load test	Repair/replacement of components based on test results

TABLE XI(a). (cont.)

Component/ageing concern	Monitoring/in situ inspection	Prevention/mitigation
Instrumentation and control cables:		
Decrease in % elongation Decrease in IR value Brittleness	Accelerated ageing tests on representative samples (no degradation reported on RAPS cable samples) Insulation quality test (by bending) and resistance test	Procurement of appropriate type of cable for a specified environment environment Replacement of affected portion of cable
Elastomers (O-rings, gaskets, diaphragms, hoses):		
Stiffness/hardening Crack on bending	Periodic inspection	Replacement by appropriate materials based on rigorous testing
Brittleness	Inspection for leakage	
Process instruments:		
Drift in parameters with time/stability Safe operating region	Periodic inspection	Replacement after failure analysis
Concrete containment structure and penetrations:		
Degradation of containment leaktightness, or structural integrity including relaxation in prestress)	Periodic integrated leakage rate test and local leak tests on penetrations	Repairs based on test results
	Periodic visual inspection Long term monitoring of strain changes in concrete structure	
Component supports, hangers, snubbers:		
Corrosion, fatigue failure of welds	VT/PT of welded joints VT of hangers/supports VT and operability check of snubber	Repairs as required

Note: RAPS — Rajasthan Atomic Power Station, MAPS — Madras Atomic Power Station.

TABLE XI (b). LIFE MANAGEMENT STRATEGIES: PRESSURE TUBES OF INDIAN HWRs

Ageing mechanisms	Potential consequences	Reactor units	Monitoring and corrective actions
Irradiation enhanced deformation (wall thickness, internal diameter, axial creep)	Deformation exceeding design limits	RAPS MAPS NAPS	Monitoring of deformation in sample channels during in-service inspection Periodic axial repositioning of creep stops to accommodate elongation En masse replacement of channels before elongation exceeds allowance provided in channel bearing length
		KAPS onwards	Monitoring of deformation in sample channels during in-service inspection Periodic axial repositioning of creep stops to accommodate elongation
Delayed hydride cracking initiating from hydride blister at pressure tube–calandria tube contact	Pressure tube rupture or failure to meet leak before break criterion	RAPS, MAPS, NAPS, KAPS 1	Monitoring of pressure tube–calandria tube gap or sag profile, and spacer positions in selected channels during in-service inspection Identification of channels which could experience pressure tube–calandria tube contact by use of a combination of vibration diagnostic techniques, calculations based on in-service inspection data on spacer location and direct inspection of selected channels Computation of hydrogen pick-up and blister depth at contact location using conservative data and assumptions Quarantining/replacement of channels having unacceptable computed blister depth Repositioning of spacers in channels where inspection has shown displacement to have occurred and in which the blister has not yet exceeded the acceptable depth En masse channel replacement undertaken on the basis of an assessment of hydrogen content and contact time
		KAPS 2 onwards	Periodic inspection (normal) No special action required. Four tight fit spacers ensure no pressure tube–calandria tube contact

TABLE XI(b). (cont.)

Ageing mechanisms	Potential consequences	Reactor units	Monitoring and corrective actions
Delayed hydride cracking initiating from a stress concentration (e.g. from service induced flaw)	Failure of leak before break	All units	Volumetric/surface examination of sample channels during BLI and in-service inspection Evaluation of cause and effect on pressure tube integrity of any detected flaw exceeding calibration standard Repair/quarantine/replacement of channel if necessary
Changes in tube properties during operation	Failure of leak before break		Material surveillance: Post-irradiation examination of selected channels from lead units; scrape samples
Reduction in fracture toughness (critical crack length) owing to hydrogen pick-up and irradiation	Failure of leak before break		Operating procedure for avoidance of cold pressurization Replacement whenever hydrogen levels become unacceptable

Note: RAPS — Rajasthan Atomic Power Station, MAPS — Madras Atomic Power Station, NAPS — Narora Atomic Power Station, KAPS — Kakrapar Atomic Power Station.

TABLE XII. INDIAN HWR FUEL CHANNELS

Reactor	In-service date	Pressure tube material	Garter spacers (number/fit)
RAPS 1	1971	Zircaloy 2	2/loose
RAPS 2:			
Original	1981	Zircaloy 2	2/loose
Refurbished	1998	Zirconium–niobium	4/tight
MAPS 1 and 2	1984	Zircaloy 2	2/loose
	1986	Zircaloy 2	2/loose
NAPS 1 and 2	1991	Zircaloy 2	4/loose
	1992	Zircaloy 2	4/loose
KAPS 1	1993	Zircaloy 2	4/loose
KAPS 2 and later	1995 onwards	Zirconium–niobium	4/tight

Note: RAPS — Rajasthan Atomic Power Station, MAPS — Madras Atomic Power Station, NAPS — Narora Atomic Power Station, KAPS — Kakrapar Atomic Power Station.

(i) Fuel channels

As indicated in Table XI, current Indian HWRs (Kakrapar 2 onwards) have pressure tubes of Zr–2.5%Nb equipped with four tight fitting spacers. The material for these tubes is quadruple melted to control trace elements and hence improve fracture toughness. The initial hydrogen concentration specified for this material is <5 ppm.

In the case of the older Indian HWRs, in which Zircaloy 2 pressure tubes are used along with two or four loose spacers, a well-structured assessment and life management programme is in place which includes in-service inspections, post-irradiation examination and maintenance/replacement of channels as required. Table XII summarizes the life management strategies for the pressure tubes of Indian HWRs.

A high priority element of this programme involves preventing the formation of unacceptable hydride blisters. Monitoring and repositioning of displaced spacers is therefore an important part of the programme.

En masse replacement of coolant channels with those of the current design is either being undertaken or is planned for these older units. This activity is taken up at an appropriate stage for each unit (after 8.5–12 full power years of operation) and is based on the:

- Predicted duration of pressure tube–calandria tube contact, and consequent blister growth;
- Axial elongation exceeding allowances available in the channel bearings;
- Deterioration in fracture toughness due to hydrogen/deuterium pick-up, which reduces the margins on meeting the leak before break criterion.

The operating procedure for startup and shutdown of the plants avoids cold pressurization of the pressure tubes. This reduces the probability of delayed hydride cracking, and also increases the probability of leak before break occurring.

In order to detect leaks in pressure tubes, in the Narora Atomic Power Station onwards, an annulus gas system is used. In reactors developed prior to Narora (i.e. Rajasthan Atomic Power Station and Madras Atomic Power Station), where the pressure tube–calandria tube annulus is open to the calandria vault atmosphere, the leak detection system is based on the detection of moisture in the calandria vault atmosphere. The sensitivity of this leak detection method is assessed as being adequate to satisfy the leak before break criterion.

The pressure tube life management programme ensures that the margins available for the safe operation of the pressure tubes continue to be adequate at all times.

(ii) Improvements in Rajasthan 2 during retubing

As part of the pressure tube replacement operation in Rajasthan 2, significant upgrades were performed in order to bring the safety systems up to current standards. These are listed below.

Rajasthan Atomic Power Station containment dousing system: The original containment dousing system at the Rajasthan Atomic Power Station incorporates a flow modulation feature whereby the dousing flow is varied in proportion to the velocity of the steam–air mixture flowing in the vicinity of the dousing curtain. This velocity, in turn, will depend on the discharge flow rate from a postulated LOCA. This scheme has been modified to produce a simpler, optimized one in which the modulation feature has been dispensed with, and a fixed dousing flow rate set which can cater for all LOCA break sizes [100].

ECCS: In the original design, emergency core cooling was provided by circulation of moderator heavy water using the normal moderator circulation pumps at relatively low pressures (6 kg/cm^2 or less). In the modified design, the ECCS has been enhanced by adding a high pressure accumulator injection system capable of supplying at a pressure of 55 kg/cm^2 [101].

Supplementary control room: A supplementary control room has been added at the Rajasthan Atomic Power Station and located away from the main control room [102]. The instrumentation in this room has cables and power supplies independent of the main control room. The features provided ensure that the supplementary control room together with local panels/controls have the capability to perform essential safety functions (i.e. safe shutdown and decay heat removal, as well as monitoring of plant safety status) independently of the main control room.

Minimization of in-leakage of instrument air into containment: During postulated LOCA conditions, which require boxing up of containment, the continued in-leakage of instrument air into the containment results in its gradual repressurization. To minimize instrument air in-leakage, a design modification has been made whereby valves requiring continued air supply under LOCA conditions have been provided with a separate air supply, to allow the main instrument air supply to the reactor building to be cut off during containment box-up conditions. This air cut-off is envisaged as being manually actuated about one to two hours into the accident.

Augmentation/improvements in station electric power supplies: The backup power supply to the Class III buses, which was originally based on 2 × 100% capacity diesel generators, has been augmented by addition of a third diesel generator set. This diesel generator, having a different cooling system (air cooling) to the existing ones, is located above the maximum anticipated flood level arising from the postulated failure of the upstream dam. Re-routing of cables for power supplies connected to safety related loads has been carried out in order to achieve improved segregation of redundant trains of supplies.

(b) Canadian programme [103]

The objectives of the Canadian plant life management programme are to:

- Maintain the long term reliability and safety of HWRs during the design life (life assurance);
- Maintain the long term availability and capacity factors of these plants with controlled and reasonable generating costs during the nominal design life (life assurance);
- Avoid ‘surprises’ by identification of potential ageing issues, ahead of their occurrence, and provide the means for their monitoring and mitigation in order to ensure reliable component performance;
- Preserve the option of extending the lives of current plants while maintaining good safety and availability at reasonable cost beyond the nominal design life of 30 years, up to 50 years or more (life extension).

Plant systems, structures and components have been classified into one of the following categories:

- Critical non-replaceable (e.g. reactor and civil containment structures);
- Critical replaceable (e.g. pressure tubes and steam generators);
- Non-critical (systems, structures or components that do not directly impact safety or plant reliability and hence can be replaced or maintained during the plant's regular outages and maintenance call-outs).

Critical components are defined as those which rank highly in terms of the following:

- Consequence of failure on plant safety,
- Consequence of failure on plant operation,
- Cost and/or duration to refurbish or replace,
- Impact on plant availability (while on refurbishment or replacement),
- Radiation dose to repair or replace,
- Regulatory importance.

On the basis of this approach, the plant life management strategy adopted is to:

- Identify critical components;
- Undertake ageing assessment studies of such critical components;
- Implement life management programmes to maximize component life, ensure good performance and monitor plant condition;
- Plan, 'scope out' and implement the required programmes to attain the original design life;
- Prepare the economic cases for rehabilitation and life extension;
- Implement rehabilitation and operate beyond the nominal design life.

Figure 114 illustrates the elements of the plan. Various programmes are currently under way, comprising:

- Detailed ageing assessment studies of critical systems, structures and components. To date, two assessment studies have been completed: the reactor structures assembly and the civil containment structures.
- Fuel channel life assurance programme. This prescribes inspection, monitoring and mitigation techniques required for the effective management of pressure tube life over the lifetime of the plant. Calandria tube life is expected to exceed 50 years.
- CANDU industry inspection and maintenance manual.

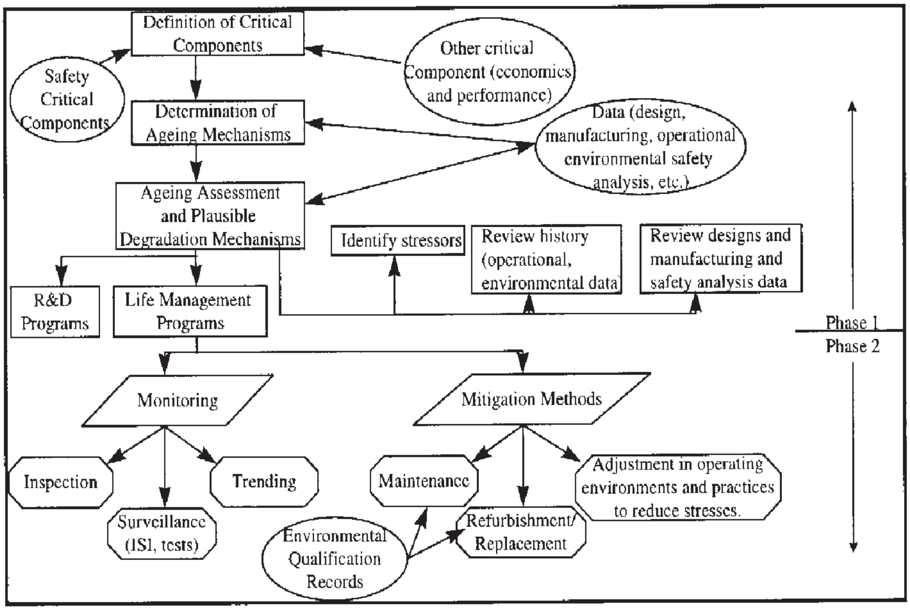


FIG. 114. Plant life management elements.

Table XIII shows the phases of the plant life management. A 'technology watch' programme has been initiated in order to ensure early identification of potential ageing issues, and is based on review of:

- Canadian and world nuclear plant operating experience;
- Canadian and international experience with ageing related phenomena;
- R&D on HWRs supplemented by data from academic and research institutes, such as the Electric Power Research Institute;
- HWR systems, structures and components that are exhibiting an increase in failure rate with increasing age;
- The AECL feedback monitoring system which documents the lessons learned from operating HWR plants;
- Effective programmes adopted by HWR utilities and which are known to have yielded good results.

TABLE XIII. THE CANDU PLANT LIFE MANAGEMENT MULTIPHASE APPROACH

Phase	Period	Scope
1	1994–1998	‘Scoping’ phase: • Identification of critical systems, structures and components • Implementation of ageing assessment studies of critical components • ‘Scoping’ of regulatory and safety related design • Identification of modifications • Support of R&D • Development of advanced technology • Phase 2 planning
2	1998–2008	Detailed planning, evaluation and engineering assessment: • Formulation and planning of the detailed plant life management programme • Detailed inspection and specific residual life of CANDU 6 • Assessment of key components • Implementation of some life management programmes, e.g. plant monitoring, surveillance • Documentation for submission to the regulatory authorities • Continuation of advanced technology development
3	2008–2012	Refurbishment, replacement and rehabilitation

The issue of cable degradation in older plants is being addressed. The following actions have been taken:

- Plant specific review of different cable types and formulations used for power, and control and instrumentation cables.
- Review of available test data on cable types and formulations in order to confirm (or otherwise) the acceptability of degradation in actual HWR environments over a 30-year plant life, including post-accident conditions.
- Review of the validity and reliability of some cable condition monitoring techniques for a given degradation mechanism.

- A test programme is under way. Remedial plans for partial cable replacements may be required if further test results indicate the need.

From the point of view of preventing the erosion of margins to safety system trip set points, or preventing conditions which affect safety analysis, the following aspects of ageing are important:

- Compensation for pressure tube creep to prevent loss of margin in the regional overpower trip set points. Both fuel and non-fuel options have been explored.
- Assurance that multiple steam generator tube failure (resulting from common degradation) is not possible; the alternative is to have to analyse it in the safety analysis. Steam generator degradation is part of the plant life management programme.
- Assurance that containment meets its leaktightness requirements *or* demonstration that a higher leak rate is acceptable.

(c) Ageing management in Romania

At the request of SNN, CITON undertook a comprehensive study in order to understand ageing problems and to develop an ageing management programme (see Fig. 115). This work was also realized with the help of the IAEA (information, documentation, fellowships, etc.). From the general programme, CITON has:

- Conducted general analyses of component ageing information and developed a programme for ageing management,
- Conducted an analysis of CANDU 6 component ageing characteristics and established condition indicators (preliminary),
- Selected the systems and components to be monitored and the end parameters to be recorded,
- Established the reference values for condition indicators (partial).

One of the characteristics of the programme is that it tries to obtain, at any one moment, a general view of the plant ageing condition (based on the ageing status of representative structure lines and components). In this way it will be much easier to establish requirements for maintenance (what to repair, when to repair it), so as to ensure plant safety and at the same time justify to the regulatory authority that the safety operation envelope has been maintained.

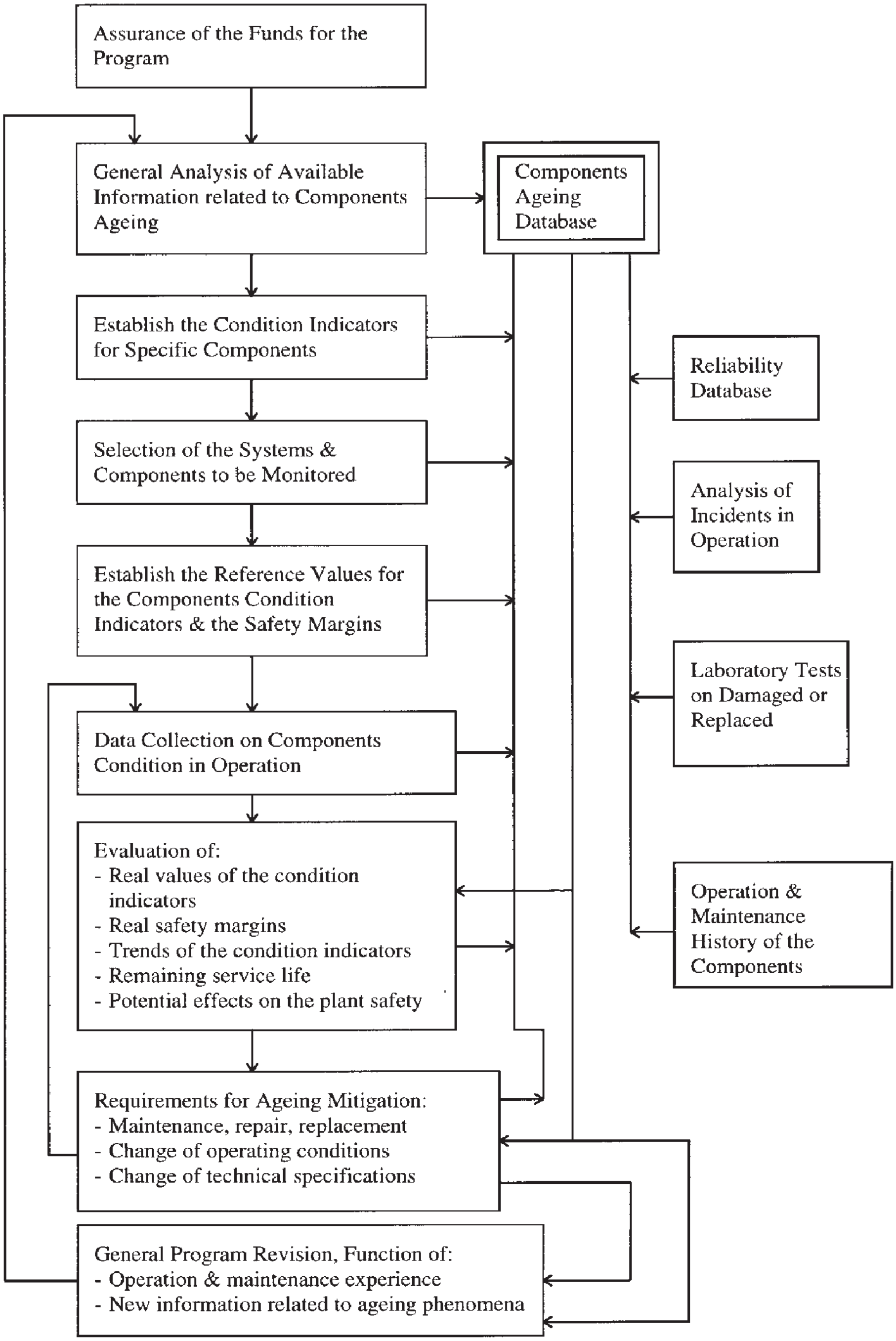


FIG. 115. General programme for component ageing management.

(d) Activities in Argentina related to plant life evaluation, management and extension

(i) Introduction

As indicated previously, since September 1994 the organization of Argentina's nuclear power activities has been conducted by the following entities:

- Nucleoeléctrica Argentina (which operates two nuclear power plants: Atucha 1 and Embalse).
- Autoridad Regulatoria Nuclear.
- CNEA.

Within this scheme, one of the main activities undertaken by CNEA is to provide technological assistance to Nucleoeléctrica Argentina for nuclear power plant operation. Work on the life extension of nuclear power plants is included in these activities.

In the past, nuclear power plant life management in Argentina was based mainly on the corrective maintenance concept which is, in turn, based on the replacement of damaged parts detected during the periodic outages. In recent times, as a consequence of increasing concern about ageing and life extension, a programme was initiated to address these subjects.

The programme, now in its initial phase, comprises several tasks. The first task involves the identification of components, ageing mechanisms and materials likely to age. In the area of mechanical components, a starting point is provided by consideration of a matrix involving major components, e.g. reactor pressure vessel, reactor internals, steam generators, pressure tubes and piping. Another major area of activity corresponds to instrumentation and control equipment.

To fulfil these requirements, CNEA has initiated a comprehensive programme of action. With regard to major mechanical components, CNEA has set up a technical committee for nuclear power plants support (CAPCEN). CAPCEN is intended to function as a knowledge 'reservoir' of those issues that are concerned with the performance, safety and life extension of nuclear power plants. Other units of CNEA also contribute to efforts in the area of plant life management and extension, especially in relation to instrumentation and control equipment testing, recalibration, modification, upgrading or replacement by updated or technologically more advanced designs. Similarly, specific studies are carried out in certain areas to identify those areas where improvement can be made, for instance, in radiological conditions relevant to maintenance tasks, risk level in safety systems, operational efficiency of systems and components, and in operational optimization, in particular with regard to fuel management.

With regard to CAPCEN, one of its most important activities is to promote research work; the main technical areas covered are pressure vessel and piping, steam generators, heat exchangers, and fuel channels and core internals.

Pressure vessel and piping: To reduce uncertainty with respect to the results obtained to date in the Atucha 1 reactor pressure vessel surveillance programme. Such uncertainty is due to the different irradiation conditions suffered by the vessel walls and those applied to the specimens used in the surveillance programme. The aim of this activity is to carry out studies and provide the necessary guidance concerning the pressure vessel integrity. If necessary, a life extension programme based on the above objective will be carried out.

Steam generators and heat exchangers: To provide support to operating nuclear power plants, as well as provide support for inspection and life prediction. The main activities comprise monitoring of cooling system chemistry and undertaking studies on the feasibility of equipment cleaning during programmed outages. This project also includes the study and resolution of problems concerning steam generators and moderator circuit heat exchangers.

Fuel channels and core internals: To perform studies in order to provide advice and guidance directed to achieving the optimum behaviour of the reactors' internal components. At this stage, the main objective is to increase knowledge of the in-service behaviour of fuel channels and other core internals of the Atucha 1 and Embalse plants.

(ii) Pressure vessel and piping

In order to monitor radiation embrittlement (a key parameter in nuclear power plant life management) of the Atucha 1 reactor pressure vessel, a surveillance programme is being carried out. Owing to the special reactor pressure vessel design, surveillance capsules, containing radiation and temperature monitors, Charpy impact, fracture mechanics and tensile specimens, cannot be placed close to the reactor pressure vessel's inner wall, as is the usual practice in LWRs. Instead, the Atucha 1 capsules were placed at the bottom of the coolant channels with resultant negative consequences as regards leading factor, irradiation flux and spectrum. Consequently, two actions were taken:

- Ex-vessel dosimetry was implemented: placement and removal of neutron dosimeters at the outer wall of the pressure vessel was undertaken during the programmed outages. All capsule holders were placed at the same mid-core level in three different azimuthal positions. Each capsule contained six neutron dosimeters (i.e. cobalt/aluminium, titanium, iron, niobium, copper, silver), wrapped into a batch with an aluminium foil holder. In total, 36 monitors were furnished by the reactor pressure vessel surveillance programme. A new dosimetry survey is being implemented to confirm the results obtained.
- Tests on samples of the reference materials irradiated at the same place as the surveillance ones are being performed. In 1980, SET-3, comprising 10 capsules

containing specimens of IAEA reference steel 20MnMoNi55 (IAEA Code JF) including Charpy, were installed at Atucha 1.

(iii) Steam generators and heat exchangers

Steam generators are among the other key components of nuclear power plants that can lead to a derating of the station owing to a reduction in the heat transfer capability. There are also some heat exchangers in the Argentinian stations which, because of the particular plant design, are relevant to operation, safety and economy [104, 105]. Consequently, they have to be considered key components and included in the scope of the present programme. The strategy adopted by CAPCEN for the surveillance of steam generators and heat exchangers is based on the following four aspects:

- The heat exchange current performance is determined on the basis of an accurate heat exchange calculation. The variables involved (i.e. temperature, flow rates and pressure drops) are measured accurately. The results are then verified by cross-checking balances. The surveillance programme covers the behaviour of each critical component, addressing the effects of fouling and failures in the instrumentation and chemistry control.
- Some modifications in an ageing station can be introduced into the chemistry for several reasons:
  - — If copper alloys were removed, then the secondary side chemistry could be shifted to high AVT with the benefits in the reduction of the iron transport and the balance of plant.
  - — More stringent specifications for the elimination of potentially damaging species have been introduced over the years and new oxygen scavenger concentrations and alkaline agents are currently in use.
  - — More sensitive devices, together with data acquisition and predictive software, are available. Their implementation in the plants is expected.
- Non-destructive tests, as well as those performed on tubes withdrawn from steam generators and heat exchangers, are, despite the cost and effort they represent, a source of invaluable data in the prevention of corrosion problems. They also provide a direct idea of the magnitude and composition of the deposits and the presence of unexpected species in them. In addition, an extensive programme of corrosion surveillance on alloy samples located in on-line autoclaves provides information on corrosion, corrosion product transport and activity transport.
- The advances in modelling and the power offered by current three dimensional computer codes which simulate equipment behaviour are considered useful. In this connection, validation of the results is performed by new devices. A good

example is the use of ultrasonic flowmeters for the detection of deviations in the downcomer flow rate.

(iv) Fuel channels and core internals

This topic covers the surveillance of fuel channels and behaviour of core internals, optimization of their performance, and the planning of repair and replacement of these components. CAPCEN also supports R&D work related to these activities, mainly in the fields of deformation under irradiation, corrosion and hydriding, fracture mechanics and non-destructive testing. The activities of both operating nuclear power plants, Embalse and Atucha 1, are described in the following sections.

Embalse: At Embalse, during the annual programmed outages (which are undertaken at other CANDU plants), the operator (Nucleoeléctrica Argentina) undertakes the repositioning of the garter springs which prevent contact between pressure tubes and calandria tubes. Contact between the pressure tubes and calandria tubes could eventually lead to failure of the pressure tube through the delayed hydride cracking mechanism. The pressure tubes are made of cold worked Zr–2.5%Nb, while the calandria tubes are made of Zircaloy 2. CNEA participates regularly in repositioning activities in the following areas:

- Pre- and post-repositioning inspection of the fuel channels through non-destructive ultrasonic testing,
- Analysis of fuel channel deformation through the specially developed MACACO code,
- Fracture mechanics assessment of pressure tubes during the repositioning.

Moreover, CNEA has participated in the planning of strategies of pressure tube life evaluation. The ‘fitness for service’ condition of operating pressure tubes depends on the fulfilment of the leak before break criterion, which guarantees that any leaking crack in a tube will be detected before it reaches the critical size. Fitness for service evaluation is conducted because the properties of the pressure tube suffer degradation during operation through the combined effects of corrosion and hydrogen (deuterium) uptake, irradiation embrittlement, and irradiation creep and growth.

At Embalse, two pressure tubes, in the A-14 and L-12 lattice positions, were replaced in 1995. CNEA has initiated a programme for evaluating the condition of these pressure tubes. This programme includes the following activities:

- Visual inspection;
- Metallographic evaluation (oxide thickness, hydride distribution);

- Measurement of hydrogen isotope concentration;
- Measurement of in-service deformation;
- Testing of tensile and fracture properties;
- Determination of delayed hydride cracking velocity.

Hydrogen concentration measurements have already been carried out and the results are being used by Nucleoeléctrica Argentina to support the repositioning strategies. Other tasks are to be performed in the near future in hot cell facilities at CNEA's Ezeiza Atomic Centre. The main objective of this programme is to fulfil the requirements of the new Canadian standard relevant to replaced fuel channels [73].

Atucha 1: In the Atucha 1 pressure vessel reactor, the coolant channels and the other core internals suffer several types of degradation during operation comprising oxidation and hydrogen (deuterium) absorption, irradiation (and hydrogen) embrittlement, and irradiation creep and growth. The oxidation rate of the coolant channels has shown signs of acceleration (breakaway) after 10 full power years, and consequently the deuterium uptake rate has also reached very high values. This phenomenon has led to embrittlement and the loss of integrity of the thin insulating coolant channel foils. The irradiation growth of the coolant channels has also exhibited breakaway behaviour.

As a consequence, the decision to replace the coolant channels was taken some years ago. The replacement programme was initiated in 1990 and continued in the following years during the programmed plant shutdowns. It will be completed in 2002.

The coolant channel replacement strategy and other core internals life management activities were planned when CNEA owned the plant, prior to August 1994. Currently, CNEA participates in the inspections of core internals and frequently issues recommendations about remedial actions.

R&D activities: In addition to the described activities, which are directly related to the technical support of both power plant operation and life management, CNEA actively conducts the following R&D:

- Changes in mechanical properties under irradiation;
- Enhanced irradiation growth;
- Post-irradiation inspection of components;
- Phase transformations and microstructure of zirconium alloys;
- Diffusion at the grain boundaries in zirconium–niobium alloys;
- Mossbauer spectroscopy;
- Zirconium oxides in single crystals and alloys, and as precipitates of the type $Zr(CrFe)_2$ in the oxide;

- Corrosion and hydriding, hydrogen effects on performance, hydrogen embrittlement;
- Zirconium hydride blister measurements;
- Oxide cover, i.e. depth, properties and measurements;
- Hydrogen and deuterium content measurement by neutron diffraction;
- Fracture and delayed hydride cracking of zirconium alloys;
- Texture and residual stresses.

5.4.1.4. Limiting conditions for operation

In mid-1992, the Romanian organization CITON initiated and developed a programme for implementing limiting conditions for operation of Cernavoda 1. This action, complying with Romanian nuclear safety norms (Nuclear Reactor and Nuclear Power Plants, article 16), implied development of a long term work plan taking into account the difficulty of matching the CANDU HWR's design and requirements to the US Nuclear Regulatory Commission's Regulations (NUREG 1431 (1991)).

Consequently, the documentation was developed in three stages, although this is not yet mandatory for the licensing process:

- Phase A, which was finished in December 1997, generally complies with NUREG 1431 requirements in terms of content, technical information and technical bases. The background for this documentation was set by the Cernavoda 1 design, reference design documentation (operating policies and principles, impairment manuals, operating manuals, etc.), IAEA Safety Guides, Romanian nuclear safety norms, as well as by the results of commissioning phases A, B and C. The limiting conditions for operation also reflect the safety features of CANDU HWR design, which are outlined according to the AECL philosophy for operation and the new worldwide safety concepts of defence in depth, diversity, two group separation, reliability, operator oriented emergency operating procedures, etc.).
- Phase B is intended to improve the information related to operator action (times, steps, etc.), complying in a more accurate manner with the operational procedures and operating experience. At the same time, the surveillance requirements will be revised.
- Phase C shall provide a licensing documentation of a high standard, accounting for the dynamics of international standards, rules and practices. The limiting conditions for operation will be fully revised according to analytical studies and safety analyses (thermohydraulics, PSA, severe accident, etc.).

5.4.2. Current designs — evolutionary improvements

Utilities, as the customers, have the final decision on the acceptability of a design. In recent years, utilities have become quite conservative, demanding:

- Proven plants and/or components;
- Enhanced safety, operability and owner protection;
- Economic competitiveness compared with alternatives.

An optimal balance is required among these requirements. The resolution to date, and for the foreseeable future, is to pursue an evolutionary approach in HWR design, making incremental improvements in the areas desired by the utilities but without radically departing from the reactor concept. This evolutionary approach is evident in the improvement of the Indian 220 MW(e) series HWR, the CANDU 9 and the CANDU 6.

This section, therefore, summarizes the changes being made to current designs, focusing on the CANDU 6 series and the Indian 220 MW(e) series as examples of stations being built, and the CANDU 9 and Indian 500 MW(e) as examples of current designs.

5.4.2.1. CANDU 6

The Wolsong 2, 3 and 4 plants initially had 84 contract changes made from the reference operating plants in Canada, many of which pertained to safety. These were also carried over to the Qinshan plants in China, with further changes being made. The major safety related changes were as follows:

(a) Main control room panels

While the human factors aspect of the CANDU 6 control room has been demonstrated through operating experience, in some areas the need for improvement was indicated. The emergency core cooling panel in particular was very complex, and operation of the ECCS required some manual actions to be taken in the medium term. Consequently, the panel was totally reworked in order to incorporate human factors considerations and the switch over of emergency core cooling to the recovery mode was automated.

(b) Safety system availability

In the case of HWRs, it must be demonstrated that each safety system (SDS1, SDS2. emergency core cooling and containment) meets an unavailability target of 10^{-3} years/year. This was not always met for emergency core cooling, and the testing was somewhat onerous for the owner. The emergency core cooling design was improved in order to reduce the dominant failure modes contributing to the unavailability, e.g. through addition of a redundant ECCS heat exchanger. Shutdown availability was adequate but further trip parameters were added to improve two parameter trip coverage and, where practicable, to replace regions where manual trip was formerly used.

(c) Tritium releases

Doses to the public and to station staff incurred during the normal operation of HWRs have been adequately low compared with world performance. However, release of tritium was further reduced by the incorporation of an inlet air dryer system in the reactor building and by the doubling in capacity of the D_2O vapour recovery dryer for the heavy water management area [106].

(d) Releases from containment

A reduction in the number of known leakage paths from containment and the application of R&D to study the chemistry of fission product behaviour led to a reduction in the predicted release from containment for certain accidents. In addition, the ingress of instrument air into the containment following an accident was minimized by adding a post-LOCA instrument air compression system. This system extracts air from the containment and supplies it to instruments needed for accident management, thereby reducing the long term containment pressurization rate. A gross containment leakage monitoring system was added to reduce the operational burden of full pressure testing.

(e) Hydrogen igniters

The first generation CANDU 6 plants relied on natural circulation and the relatively low concentration of hydrogen in a LOCA plus LOECC to preclude detonation: combustion was unlikely but could be tolerated. In order to achieve greater protection against potentially damaging localized hydrogen concentrations, a network of hydrogen igniters was installed in the containment buildings at Wolsong 2, 3 and 4, and Qinshan 1 and 2.

(f) Environmental qualification

A structured environmental qualification programme was undertaken to ensure that equipment credited in an accident was formally qualified for it. Very clear provisions are included in the design manual and in the technical specifications of the equipment and components requiring environmental qualification, and these are monitored during equipment manufacture.

(g) Safety analysis

On the licensing side, the identification of design basis accidents followed on from the requirements stipulated in the Atomic Energy Control Board (AECB) consultative document C-6 Rev. 0.

This document, which was first applied during the licensing of the Darlington nuclear power plant in Canada, calls for a more thorough investigation of potential initiating events and event combinations (five event classes versus two formerly), and in many cases reduces the allowable doses. This was supplemented by a Level 1 PSA.

(h) Safety critical software

An approach based on the use of mathematical specifications for the software requirements and software design has been implemented for the trip computers in the shutdown systems. This allows the use of mathematical verification to demonstrate that the design meets the requirements, and results not only in increased reliability of the software, but also in the ability to demonstrate the correctness of the software.

(i) Qinshan safety related improvements

In addition to having made the same improvements as those made at Wolsong 2, 3 and 4, the Qinshan plant has made the following major safety related enhancements:

- In the case of those HWRs employing the two group design provisions, there was no requirement to have a fire protection system provided with seismically qualified electrical power. In the event of an earthquake, seismically qualified Group 2 systems would perform the safety functions of shutting down the plant, removing decay heat and monitoring plant status. Any fires in the Group 2 area induced by the earthquake would be very local and could be extinguished manually. However, in the case of Qinshan, the customer required a seismically qualified power supply for the firewater pumps. This leads to a robust fire protection design.

- The Qinshan site is at some risk from tornadoes. Hence, the plant accommodates a design basis tornado, which requires the reinforcement of certain structures and the protection of selected equipment.
- To assist in emergency planning, a technical support centre and a critical safety parameter monitoring system are part of the Qinshan design.
- The spent fuel bay uses a stainless steel liner to reduce the likelihood of leakage.
- The spent fuel bay water is cooled by recirculated cooling water instead of raw service water; this reduces the risk of flooding.
- Seismically qualified heating, ventilation and air conditioning are provided for the secondary control area to ensure heat removal after an earthquake.
- Several smaller changes were made in order to comply with Chinese regulatory requirements.

5.4.2.2. Indian 220 MW(e) HWRs

Indian HWR designs have undergone four major stages of evolution, represented by following groups of plants:

- Rajasthan/Madras Atomic Power Stations,
- Narora/Kakrapar Atomic Power Stations,
- Kaiga Atomic Power Station/Rajasthan 3 and 4 (current 220 MW(e) HWRs),
- Tarapur 3 and 4 (current 500 MW(e) HWR designs).

This section discusses the first three; the 500 MW(e) design is covered in Section 5.4.2.4. Some of the improvements made to the later designs are as follows.

(a) Layout

In the current designs (both 220 MW(e) and 500 MW(e)), the orientation of the turbine generators with respect to safety related structures, including the reactor building, the control building and the diesel generator building, is so arranged as to be outside the zone of influence of low trajectory missiles from postulated turbine failures.

The three 100% diesel generators for the Class III power supply of each unit are distributed and housed in separate, safety related diesel generator buildings.

(b) Containment

At the Narora/Kakrapar stations, the primary containment is capped by a flat slab with the steam generators protruding into the secondary containment space. In subsequent reactors, the primary containment was extended in height so as to house the steam generators entirely, and was capped by a dome. This design eliminates the need for secondary containment blow-out panels which were required in Narora/Kakrapar stations to cater for steam line breaks.

Within the primary containment, a structural wall has been introduced for supporting the various floors, thus avoiding any loading from internal structures/ equipment on the primary containment envelope.

(c) Instrument air leaks into containment

This topic has already been discussed in Section 5.4.1.3.

(d) Reactor shutdown system

In order to achieve long term subcriticality at the Narora/Kakrapar stations, a slow acting reactivity addition system (automatic liquid poison addition system (ALPAS)) supplements the two fast acting shutdown systems. Operation of ALPAS depends on the availability of moderator system flow. In the case of postulated emergency conditions involving the non-availability of moderator flow, a system for the direct gravity addition of boron to the calandria (GRABS) is provided. In subsequent reactors these two systems (ALPAS and GRABS) have been replaced by a process independent system (liquid poison injection system) which injects liquid poison directly into the moderator under gas pressure.

(e) Closed loop process water system

Current Indian HWRs have a closed loop process water system that rejects heat to a process water cooling system which employs induced draft cooling towers as the ultimate heat sink. In this design, tube failures in heat exchangers having D_2O on the tube side do not lead to an uncontrolled release of tritium into the environment.

(f) Ultimate heat sink

Special provision has been made for storing make-up water for induced draft cooling towers. This enables removal of decay heat for up to 10 days under emergency conditions. This storage facility is qualified for SSE conditions.

(g) Shutdown cooling system

In current designs, the shutdown cooling system design has been upgraded to withstand the full pressure and temperature conditions of the main HTS, thus allowing it to be connected to the HTS in hot shutdown conditions, if emergency conditions so require.

5.4.2.3. CANDU 9

CANDU 9 is a more recent evolutionary design which is based on the Bruce/Darlington reactor core adapted to a single unit containment (Fig. 116) [107, 108]. The more significant safety enhancements are as follows [109]:

(a) Containment

The CANDU 9 design uses a cylindrical, prestressed containment building with no dousing sprays and a steel liner to achieve superior capability combined with increased simplicity relative to CANDU 6. The design leak rate of 0.2%/d at design pressure is less than half that of the CANDU 6. As a result of the lower leak rate, the

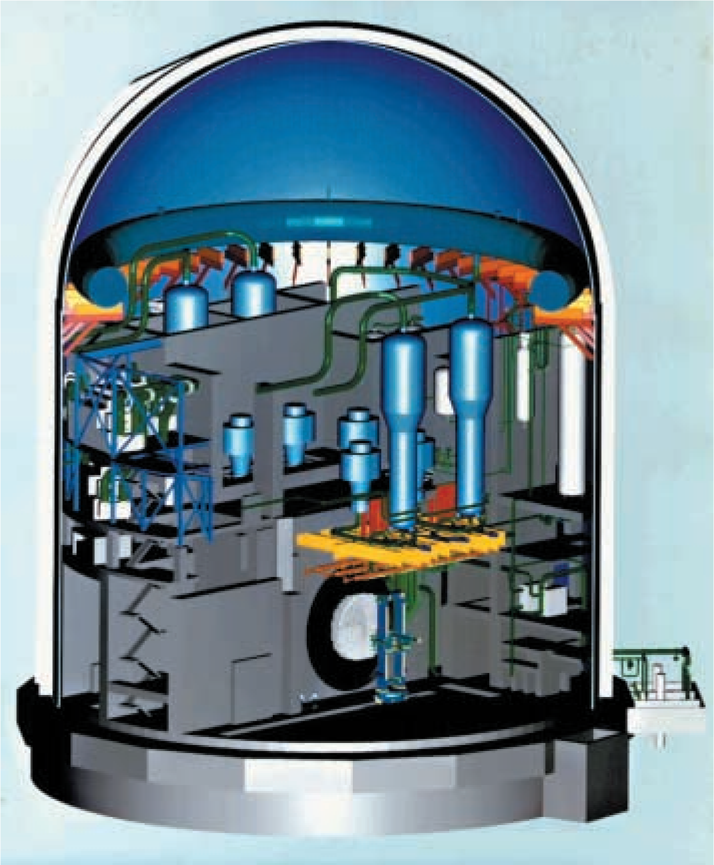


FIG. 116. CANDU 9 cutaway.

exclusion area boundary radius has been reduced to less than 500 m (calculated for an intact containment and the source term arising from a LOCA plus LOECC). However, Canadian licensing practice also requires consideration of accidents such as LOCA plus an assumed failure of the containment ventilation system to effect isolation. It also requires that the ventilation isolation system, as a subsystem of containment, has a demand availability of greater than 0.999, and that this be demonstrated by periodic testing. To ensure that the plant meets licensing requirements in Canada, as well as licensing practice in other countries, two separate and independent ventilation isolation systems, each with its own set of redundant isolation valves, have been provided for CANDU 9. This increases the availability of containment isolation to the point where failure of containment ventilation isolation after an accident can be excluded from the design basis dual failure accidents.

The ventilation system within the containment continues to operate after isolation and provides enhanced atmospheric mixing within the reactor building following a LOCA. High ventilation flows promote mixing, and this is backed up by having igniters and passive recombiners distributed throughout the building (Fig. 117) in order that hydrogen produced in severe accidents will either recombine or be locally burned before it can form explosive concentrations.

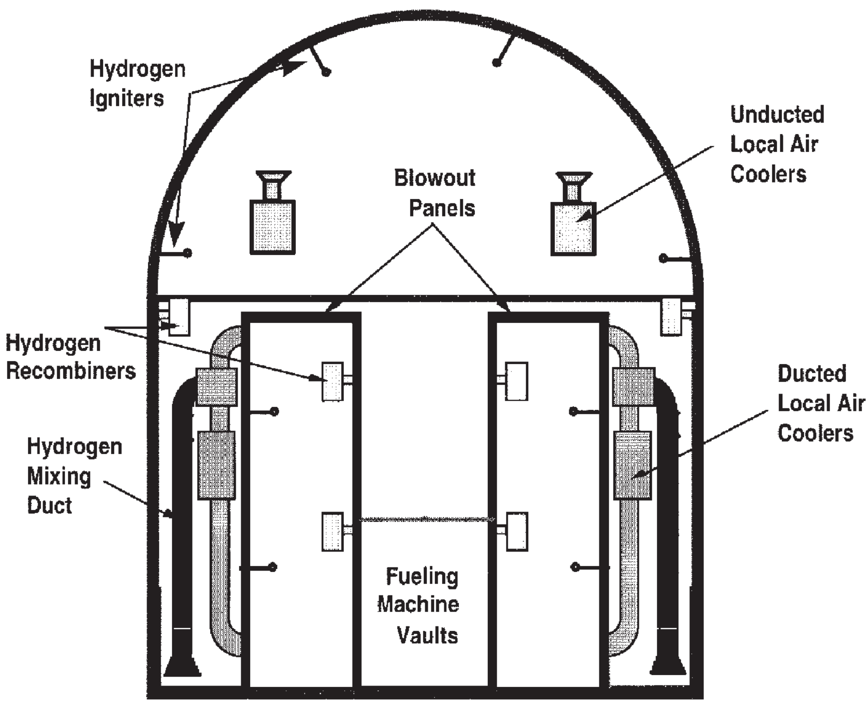


FIG. 117. Hydrogen control in CANDU 9.

The ultimate containment failure pressure is several times greater than the design pressure, allowing time for operator action to be taken in the event of severe core damage accidents.

(b) Grouping and separation

The grouping and separation of safety related systems has been strengthened. In HWRs generally, safety and process systems are divided into two physically separate and functionally independent groups. Each group, by itself, can shut down the reactor, remove decay heat and monitor the state of the plant. In the CANDU 9 design, there is now full separation of Group 1 and Group 2 electrical systems (elimination of cross-connections), seismic qualification of the Group 2 electrical systems and also portions of the Group 1 system, and better separation of 'odd' and 'even' trains of the Group 1 electrical system and the ventilation systems.

(c) Main control room capability

All accidents, including earthquakes, can now be managed from the main control room without the need to use the secondary control area. The secondary control area is retained in the eventuality that the main control room itself becomes uninhabitable as a result of a major fire or a hostile takeover.

(d) HTS

CANDU 9 uses the single loop Bruce B HTS arrangement with two inlet headers and one outlet header at each end of the reactor. Improvements made include increased pressurizer volume to accommodate all of the inventory change from zero power 100°C to full power hot conditions. This increases the reliability of thermosyphoning after transients or accidents which result in loss of forced flow, including a steam line break combined with loss of Class IV electrical power, by ensuring that full reactor coolant inventory in the HTS is maintained at any time during cooldown.

(e) Reserve water tank

A large reserve water tank is located at a high elevation in the reactor building (Fig. 116). This tank, conceptually similar to the CANDU 6 dousing tank, supplies water to the ECCS sumps in the event of a LOCA; the ECCS pumps can then recirculate the water back to the HTS. The tank provides a severe accident prevention function (providing backup emergency water supplies to the secondary side of the steam generators and to the HTS). It also provides a severe accident mitigation function (supplying make-up water to the moderator and/or the shield tank). Steam relief pipes, sized to remove decay heat, have been added to the shield tank to effect these functions.

(f) ECCS

The ECCS has been simplified. All system components, except for the pressurized gas tanks, are situated within the reactor building, thereby reducing the probability of an ECCS failure occurring outside the containment and eliminating the need for a number of isolation valves. The arrangement also results in relatively short ECCS water injection lines, thereby enhancing performance. One way rupture discs and floating ball isolation valves (Fig. 118) are used to separate the HTS from the ECCS, simplifying the ECCS and reducing both capital and maintenance costs.

(g) Dose reduction

Further reductions in occupational and public radiation dose are expected. In order to reduce operator dose, the following changes, among others, were made:

- The reactor building dryer capacity was more than doubled,
- The amount of chromium in the outlet feeders was increased to reduce the corrosion rate,

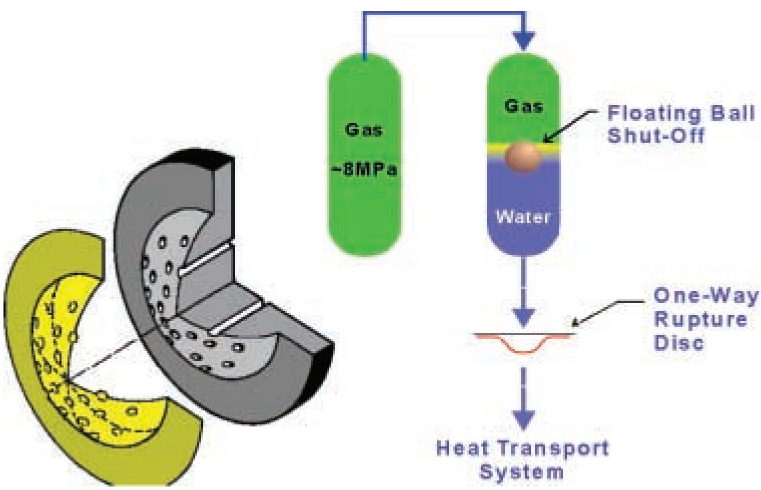


FIG. 118. ECCS simplification in CANDU 9.

- The new fuel loading area was relocated to a site outside the containment building.

In the case of public dose reduction the following actions were carried out:

- Drying of all the air vented from the containment building during normal operation,
- Relocation of all auxiliary heavy water systems from outside into the containment building.

On the basis of these changes, a target collective occupational dose of 1 Sv/a for CANDU 9 has been adopted.

(h) Operator action time

Operator action time has been increased to eight hours for almost all initiating events. This is accomplished by, for example, automatically initiated Group 2 emergency electrical power and water supplies which provide safety grade backup to Group 1 emergency power and water supplies. A Group 2 feedwater system has been engineered to supply emergency water to the steam generators automatically in the event of a total loss of the Group 1 feedwater systems (main plus auxiliary). The system is capable of high pressure operation so as to cope with all the possible operating conditions in the steam generators, and can supply water to the steam generators for decay heat removal for approximately 10 hours.

(i) Fuel channel interlacing

The reactor fuel channels have been fully interlaced with respect to coolant connections, in order to reduce the rate of rise of the power transient following a large LOCA, that is, adjacent fuel channels connect to alternate inlet headers, in contrast to the arrangement at Darlington where the fuel channels on one side of the core connect to one inlet header. This arrangement distributes the flux increase due to coolant void across the core instead of concentrating it on one side. It has almost no effect during normal operation other than simplifying the system and reducing the number of components.

(j) Human factors

The top level human factors requirements for CANDU 9 are consistent with the Republic of Korea's standard PWR requirements and the Nuclear Regulatory Commission's 10 element review plan for evaluating reactor design. Project specific design guides are produced to serve as the primary method of ensuring human factors engineering design criteria are incorporated into plant system documentation throughout all stages of the design process. An example is the requirement imposed by several project document procedures to ensure that system designers incorporate specific sections that address the human factors design requirements. In addition, the project human factors specialist is required to review and comment on all system design documentation in support of the human factors design principles. The CANDU 9 control centre design and layout, which represent an evolution of the current CANDU control centre designs, are supported heavily by human factors engineering input and review. The control centre design includes the layout of work and emergency response areas, control centre panels, operator consoles and displays, and panel device layout. Consideration of emergency procedures, critical safety parameters and post-accident management standards are also included in the control centre design. A full-scale mock-up of the main control room panels and consoles has been built to allow verification and validation of the interactions between the operator and the control features of the plant.

The control room has the following key human factors related features:

- Colour touch screen monitors for both control and safety displays, which give an overview of plant/system status;
- Advanced annunciation system with an alarm–interrogation workstation;
- Automated testing of safety systems and monitoring of the results;
- Centralized large screen systems.

(k) Trip coverage

CANDU practice is to provide, as far as practicable, two trip parameters in each of the two shutdown systems for each design basis event. Trip parameters have been added in CANDU 9 to enhance trip coverage. High and low level moderator trips have been added to both SDS1 (gravity insertion rods) and SDS2 (liquid poison injection) to increase protection against in-core LOCA and moderator failure. A high reactor building temperature trip has been provided in both shutdown systems to enhance protection of reactor building integrity for small and intermediate steam line breaks inside the building.

(l) PSA and accident management

The CANDU 9 project team has set an overall target of 10^{-5} events/a for the cumulative frequency of severe core damage from both internal and external events. This target is consistent with that set for the evolutionary advanced LWRs.

A comprehensive PSA programme has been set up to ensure that this target will be met. The PSA includes both Level 1 analysis (accident sequences) and Level 2 analysis (accident progression and consequence analysis, including containment behaviour).

A preliminary PSA has been completed for internal events. The primary objective of this was to assess the feasibility of satisfying supplementary targets with regard to the frequency of accident sequences for internal events. These targets include:

- Individual accident sequences leading to severe core damage (e.g. LOCA plus LOECC plus loss of moderator heat sink) should have a frequency of less than 1×10^{-6} events/a.
- Individual accident sequences where the moderator is available as a heat sink (e.g. LOCA plus LOECC) should have a frequency of less than 1×10^{-5} events/a.

To date, both supplementary targets have been met, which is a preliminary and necessary condition for achieving the overall target for the cumulative frequency of severe core damage. The overall target will be conclusively confirmed by the project specific final PSA, which will be based on the detailed engineering and site conditions at each plant. However, the demonstration that the design is sufficiently robust and reliable to meet the overall target will be given earlier, through the implementation of the CANDU generic PSA programme. This programme was instituted to develop the generic PSA methodology (requirements, procedures, methods and tools) for CANDU reactors, and to define the design improvements needed in both CANDU 6 and CANDU 9 products to acquire a high confidence in the achievability of the overall target. The generic PSA programme includes the analysis of both internal and external events, as well as the analysis of severe core damage and containment behaviour. Figure 119 shows the key steps of this programme.

By applying these two products to the site specific and plant specific information, the final PSA of any CANDU 9 plant will be generated.

5.4.2.4. Indian 500 MW(e) HWRs

Many of the improvements made to the Indian 220 MW(e) HWRs listed in Section 5.4.2.2 have been carried over to the 500 MW(e) design, in particular those pertinent to the layout, instrument air, closed loop process water system, ultimate heat sink and shutdown cooling system. The following additional improvements pertain to the 500 MW(e) design.

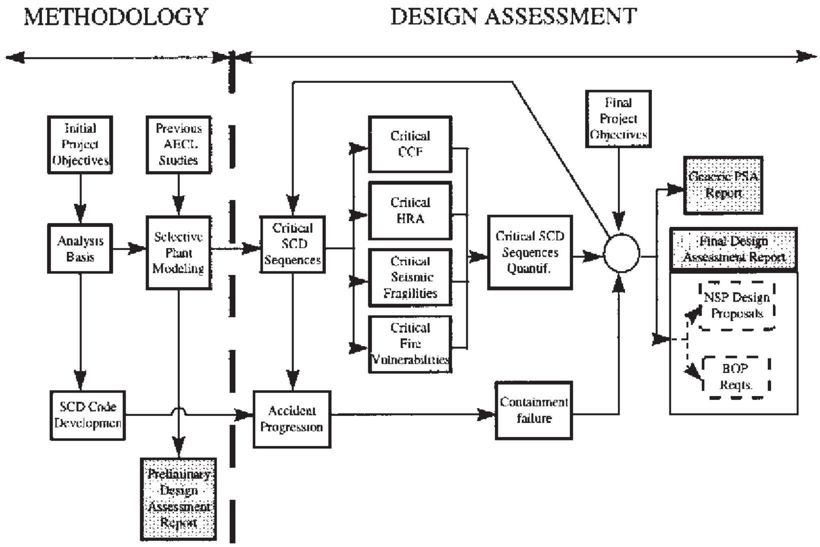


FIG. 119. Generic PSA methodology.

(a) Two group concept for safety systems

The two group concept is incorporated for protection of the plant against common mode failures, such as fire, missiles, etc. The safety systems have been divided into two groups to the extent possible and are physically separate from each other. Each group is capable of performing the basic safety functions: shutdown of the reactor, removal of decay heat, minimization of the escape of radioactivity and monitoring of the safety status.

(b) Pressurizer

A pressurizer has been added for pressure control, to smooth out pressure transients by providing the necessary vapour cushion for the HTS. This reduces the number of spurious trips caused by small pressure variations due to transients involving swelling and shrinkage in the HTS inventory. While the pressurizer controls the pressure variations, the feed and bleed circuit performs the volume control function, resulting in greater flexibility of plant operation.

(c) HTS

There are two figure-of-eight HTS loops (see Section 5.2.1.3), with an interconnection which can be isolated during a LOCA. The pressurizer rides on this line connecting the two loops and controls the pressures of both. The two loop design, with provision for isolation of the loops, limits the LOCA to a half core and reduces the consequences of the accident. In this design, the scheme of all-header injection of ECCS water has been adopted. While the water inventory requirement is increased, it simplifies instrument logic and improves the reliability of the system.

(d) Shutdown system

SDS2 has been designed to inject liquid poison directly into the moderator and has a large shutdown worth which ensures long term subcriticality.

(e) Containment

In the suppression pool system design, direct 'orificing' on the distribution header has been incorporated, thus eliminating the need for downcomers.

(f) Instrumentation

In-core, self-powered neutron detectors have been used for reactor control, flux mapping and overpower protection, with separate detectors used for the control and protection functions.

(g) Power supply

The on-site emergency power supply has been divided into two, physically separate, electrically isolated divisions (I and II) which are located in separate station auxiliary buildings. Each division has two 50% capacity diesel generator sets, making a total of 4 × 50% diesel generators for the system.

5.5. HWRs OVER THE NEXT TEN YEARS

Some of the reactors which will be deployed over the next ten years are currently in design, e.g. CANDU 9 and the Indian 500 MW(e) plant. The safety design requirements for these evolutionary plants reflect the improved safety characteristics requested by customers, and developed by international consensus through organizations such as the IAEA. Use of proven technology will determine, and to some extent limit, the safety technologies that can be employed. Meeting a competitive cost envelope will be a prerequisite for even being considered by a utility. Thus, the HWR design developed over this time-scale will represent an evolutionary improvement on the current concepts, preserving the fundamental characteristics of HWRs and improving safety systems without risking operating availability.

The previous section considered specific designs. This section discusses the general safety requirements that evolutionary HWRs are expected to meet over the next ten years. There is obviously some overlap.

5.5.1. Severe accidents

In common with other reactor designs, HWRs have focused increasing attention on severe accident prevention, mitigation and management. As noted previously, some severe accidents have been within the HWR design basis and are well understood. Thus, future work will concentrate on the subset of severe accidents which lead to severe core damage (loss of channel geometry). The long accident time-scales and the capability to arrest severe core damage progression before it can threaten containment are important HWR advantages which should be followed through. The development is seen as having the following components:

- For severe core damage mitigation, the designs should incorporate some or all of the following aspects by:
  - Providing the capability to add water to the moderator and/or the calandria and/or the end shields. The former will prevent a severe accident from progressing to severe core damage; the latter two will contain severe core damage within the calandria shell. These capabilities are aids to the operator in severe accident mitigation and management; therefore, they are not designed to the same standards as engineered safety systems, but rather use conventional engineering standards. In many severe accident sequences, loss of electrical power and/or service water are contributory failures, and therefore it may not be effective to provide these mitigation systems with conventional electrical power and water. Instead, use of firewater, or stored water driven by gravity or by portable diesel pumps, gives more independence from the cause of the severe accident.
  - Providing the means to remove decay heat from the containment and to recirculate the water added to the end shields or moderator. Following a severe core damage accident, the containment will be inaccessible for a long period. The addition of water from an external source cannot be

continued indefinitely, as eventually the building would fill up. Also, continued steaming resulting from core heat being removed by the shield tank or calandria vault water will cause a containment pressure increase beyond failure pressure. Thus, in the long term, heat must be removed from the containment and the water recirculated from the floor back to the shield tank; or the time-scale over which recirculation would be needed should be shown to be enough to effect accident recovery. Existing containment air coolers could be used for the former function, although again the severe accident event sequence could have disabled their cooling water and/or power supplies (the capability to connect external power and water to the air coolers would overcome this). Recirculation would appear to require pumps and electrical power. However, depending on the elevation of the reactor relative to the building floor, it may be possible to maintain a flood level sufficient to ensure heat removal from the calandria/shield tank assembly.

- — Taking into account the composition of the concrete below the calandria. In LWRs, one of the means of mitigation of a core melt is to provide a type of concrete below the pressure vessel that does not release large quantities of carbon dioxide when attacked by molten fuel. This reduces the rate of containment pressure rise. It is also possible to achieve this in HWRs, but this would not be needed if sufficient measures were taken to contain the debris within the calandria vessel.
- — Controlling hydrogen within the containment. Some HWRs already have hydrogen igniters to effect long term hydrogen control following a design basis accident. However, the requirements for hydrogen control following a severe core damage accident should be defined. Passive control using hydrogen recombiners is attractive, once the capability is proven.
- — Providing the capability to add emergency feedwater to the steam generators at high pressure avoids the need for manual or automatic action to be taken to depressurize the steam generators. Conversely, if the steam generators are depressurized, supplying emergency water passively, from an elevated tank (gravity), could be an important advantage in the mitigation of severe accidents where all AC power has been lost.
- — Developing severe core damage accident management programmes, procedures and guidelines for HWRs to ensure that the operators use the HWR severe accident mitigation characteristics to their advantage.

• Further development of severe core damage models is desirable in order to gain a better understanding of severe core damage accident progression. The models are used to:

- — Guide the designers on the requirements for mitigation provisions, as described above;

— Provide customers and regulatory agencies with more quantified assessments of the results of severe accidents;
— Assist in the development of severe accident management guides.

HWR specific severe core damage models do exist (e.g. ISAAC (Republic of Korea) and MAAP4–CANDU (Canada)) but are in the relatively early stages of development and have not been benchmarked against HWR specific tests.

- R&D to confirm behaviour unique to HWRs would provide further confidence in the effectiveness of severe accident management. For example, confirmation of the mode of collapse of the pressure tubes and confirmation of the heat transfer between a calandria shell and the surrounding shield water.
- Owing to the long time-scales involved in reaching a severe core damage state, there is no need for early evacuation in HWRs (before about a day). With the provision of make-up to the shield tank or calandria vault, containment heat removal and hydrogen control, there is a good possibility of preserving containment integrity in the longer term for a severe core damage accident, and therefore there may be no need for evacuation at all. It would greatly facilitate siting of future HWRs if this could be established.

5.5.2. Exclusion area boundary

In many countries, land is at a premium and it is difficult to develop new nuclear sites. There is an incentive to get the most use from existing sites, and therefore to reduce the amount of land surrounding the site that must be controlled by the owner as an exclusion area boundary.

Determination of an exclusion area boundary varies from country to country and can be set:

- On a ‘tradition’ basis,
- On a calculation basis (the dose at the boundary calculated for accidents must be less than some limit),
- On a prescriptive basis (the dose at the boundary assuming a prescriptive source term in containment and a prescriptive containment pressure transient),
- At some fixed value.

With the incentive to make the best use of existing sites, it is expected that the calculation basis will become more important since it recognizes, credits and encourages improvements in the nuclear power plant design. It is important that future HWRs have the capability to justify a smaller exclusion area boundary where a calculation basis or a prescriptive basis is used. In both cases, the results are sensitive to the containment leak rate. Thus, siting in countries which require a small exclusion area boundary implies lower leakage rates for the containment.

The provision of double containment offers a similar advantage since the *effective* leak rate is small, there is a higher tolerance to minor impairments in the primary envelope, and it also permits time to clean up the atmosphere of the primary containment envelope before primary containment controlled discharge needs to be initiated.

In some circumstances, justification for a smaller exclusion area boundary may require stronger or more detailed emergency planning.

It should be noted that a smaller calculated exclusion area boundary may not be enough. The footprint of the plant can be equally important in maximizing land utilization. For example, a group of CANDU 9 reactors generates more than twice the power on the same land area as a group of CANDU 6 reactors, owing to the CANDU 9's compact footprint.

5.5.3. Containment isolation

Although the containment isolation provisions of HWRs are similar to those of other reactor designs, and are testable to demonstrate a demand unavailability of 10^{-3}, HWRs have included failure of containment isolation as part of the design basis. While this leads to a robust design, it also places HWRs at a disadvantage with respect to siting, as their designs must take into consideration failures that other designs need not. Thus, their apparent releases from containment for design basis accidents are larger. There is, therefore, a need to reduce the release from containment, while remaining within the framework of HWR licensing practice. Two approaches have been taken to date:

- In the case of CANDU 9, the entire containment ventilation isolation system has been duplicated (see Section 5.4.2.3) using much the same philosophy as that employed for the two shutdown systems, i.e. diverse components, separate logic and instrumentation, and geometric separation (one system is located inside containment, the other outside). This results in four dampers on each ventilation line, two for each system. The Canadian regulatory authority (AECB) has reviewed this design and indicated that designers can assume that only *one* system need be considered to fail as part of the design basis dual failure analysis. In effect, the releases from containment with both systems acting, or with one system unavailable, are about the same as for current designs where the single ventilation isolation system is credited. This approach does reduce the design basis releases from containment considerably. There are other containment envelope impairments that have been considered as part of HWR licensing practice. Most do not lead to as large a release as failure of ventilation isolation. However, as regards airlocks with inflatable door seals, the dual failure analysis assumes that both sets of seals (one on each door in the airlock) have failed to inflate, and this could be a significant release pathway. Therefore, the inflatable seals on CANDU 9 have been replaced by face seals, a more passive seal which actually becomes more effective as containment pressure increases.

- The Indian AHWR (Section 5.7.2) uses a passive containment isolation system in which the pressure inside containment acts on the elevated water tank and floods the ventilation ducts, sealing them [110]. As a passive system with no valves or moving parts, it should be possible to demonstrate very high levels of availability so that failure of isolation is not credible.

5.5.4. Containment pressure suppression options

HWRs initially used high volume, high flow rate dousing sprays for pressure suppression after breaks inside containment. Dousing in the multiunit vacuum building designs was triggered passively, by the high pressure caused by the pipe break, and occurred in the vacuum building. In single unit designs, dousing was initiated actively (by opening valves) and the water sprayed over the reactor building volume. However, this has the disadvantage that a spurious signal could cause damage to equipment in the building and require a lengthy cleanup, as has happened on two occasions.

After the first two HWR stations, Indian designs have used a suppression pool in the basement of the reactor building. This is a passive system, and there is no risk of spurious actuation.

Similarly, the single unit CANDU 9 has eliminated the dousing system entirely. The containment design pressure is higher, and as a consequence a steel liner has been incorporated to reduce the leakage rate at the higher pressure.

Either of these options makes the pressure suppression function more passive and avoids the economic risk of a spurious douse occurring.

5.5.5. Human factors engineering

Evolutionary HWR designs will incorporate human factors engineering at an earlier stage. The human factors design of ‘repeat design’ HWRs was not applied systematically and it is difficult to prove that it is comprehensive, even though it has been largely validated by operating experience. An evolutionary HWR design should be able to demonstrate systematic use of human factors in the development of the design. This requires that human factors specialists be included on the design teams and a formal process of setting human factors requirements, incorporating them into the design, and having the design reviewed by the specialists for compliance. Feedback of operating experience is an important element of this programme.

Many current HWRs use the main control room for the management of most design basis accidents. However, in some cases the main control room must be evacuated and a secondary control area provided for performing the three main safety functions: ensuring safe shutdown, removing decay heat and monitoring essential safety parameters. It is, however, more desirable if the operator can remain in the main control room rather than be required to leave during the high stress period following an accident and assume control from the secondary control area. This is particularly true of external events such as earthquakes. In the case of future HWRs, the main control room should be qualified, not just to protect its occupants, but to allow accident management to be continued. The ideal would be to preserve safety control from the main control room for all but hostile takeover, direct aircraft strike or fire in the main control room itself. Indeed, some Indian HWRs already have fully seismically qualified main control rooms.

HWRs have always used computer control of the reactor, and in more recent years, of the shutdown systems. While there is no strong incentive to extend the overall scope of computer control of the safety systems, particularly given the burden of safety critical software validation and verification, it is expected that the operator interface will continue to improve with the use of modern digital aids. It is not a big extrapolation to expect the HWR control room over the next ten years to incorporate:

- Human factors requirements in all aspects of control room layout and the human–machine interface.
- Use of large overhead mimics to give an overall picture of system status.
- On-line normal and abnormal operating procedures.
- Computer assisted event diagnosis and suggestions for accident management.
- Further automation of certain safety functions to ensure longer operator action times. For example, the medium term and long term emergency core cooling is now automated on all recent HWRs.

Human factors extend beyond the main control room into areas such as maintenance and in-service inspection. An evolutionary design should incorporate the human factors aspects of in-service inspection at an early stage in the layout.

5.5.6. Operator action times

HWR designs have traditionally allowed a minimum of 15 minutes for any required operator action that must be taken from the main control room, and a minimum of 30 minutes for any action that must be taken outside the main control room, for all design basis events, including dual failures. In the case of the higher frequency events (i.e. an accident where all the mitigating systems are assumed to be available), utility requirements for new designs may require prolonging these times [111, 112]. With the already extensive automation of reactor control, and fully automated shutdown systems with dual parameter coverage for all design basis accidents, the achievement of this goal should, in most cases, prove relatively straightforward. For example, CANDU 9 has an eight hour operator action time for virtually all single process failures. In most cases, it is expected that achievement of this goal would not require addition of new systems, but automation of existing ones, for which in some cases a suitable signal already exists.

5.5.7. Layout improvements

With the constraint imposed by small sites in some countries, and the more rigorous attention being paid to two group separation, layout improvements needed to meet these requirements are expected. No general prescription can be given here, since layout is a function of the reactor design, the specific customer requirements and the specific site. However, layout improvements should include most of the following:

- A more compact and simplified layout to reduce land utilization;
- Positioning the turbine generator radially relative to the reactor building to reduce the consequences of turbine break-up (avoids turbine missiles striking the reactor building);
- Stronger separation between Group 1 and Group 2 safety systems, allowing the latter to be more effectively protected from internal and external events;
- Protection against external events common to almost all sites (e.g. earthquakes), plus space to add protection for external events specific to the site (e.g. tornadoes, tsunami).

5.5.8. Process system improvements related to safety

Modest changes to process systems can be used to reduce or eliminate uncertainties in accident analysis, and/or improve safety margins. Again, the choice will be design dependent, but the following are typical:

- HTS improvements made to reduce the likelihood of inventory loss, or to make up for it, for sequences that lead to thermosyphoning forming the heat removal mechanism (for plants in which channel boiling occurs). An example would be a station blackout, or a LOCA plus loss of Class IV power. Such improvements could include environmental qualification of the D_2O make-up system for a LOCA, or use of a larger pressurizer.

- Moderator system improvements undertaken to increase operating margin. Currently, for HWRs which do not use moderator dump, the maximum allowable moderator temperature at the hot spot is set by the dual failure LOCA plus LOECC. Since the fuel decay heat is to be removed by the moderator, prolonged dry out of the calandria tube must be prevented. The actual maximum temperature can be reduced by relocation of the moderator inlet and outlet nozzles.
- Prevention of end fitting ejection. Currently, one of the larger source terms into containment comes from end fitting ejection, assumed to occur spontaneously or as a result of a pressure tube guillotine failure. In addition, in the latter case there may be a loss of moderator from the hole in the calandria. Strengthening of existing end fitting restraints, such as the positioning assemblies and the bellows, would address both these aspects. Similarly, failure of the channel closure induced by a failure in the fuelling machine is currently assumed in the safety analysis, but appears to have no clear mechanical basis. Prevention of such failure could be further ensured by undertaking engineering changes or analysis to demonstrate that the end fitting shield plug can serve as a backup to the seal plug in preventing ejection of the fuel bundle string, perhaps by strengthening or qualifying the rolled joint between the end fitting and the liner tube.
- Further separation and protection of Group 1 electrical systems to allow continued operation of the main control room after accidents which create a harsh environment in the turbine hall. Provision of Group 2 electrical services to the main control room (for plants which do not have seismically qualified Group 1 electric supplies) to enable the operator to remain there to perform the safety functions after an earthquake.

5.5.9. ECCS improvements

The ECCS is a relatively complex system, since the requirement to separate heavy and light water components leads to a large number of valves being employed. Simplification of the system is both possible and desirable; for example, CANDU 9 has replaced one set of valves by one-way rupture disks and has used ball seals instead of valves at the bottom of the water tanks. The injection is initiated by the passive rupturing of a metal disc, the flow being stopped by a floating ball valve which seats on the tank outlet when it is nearly empty. Full-scale testing has demonstrated that the rupture discs can be designed to incur negligible additional pressure loss, and that the ball valves will seat reliably without leakage.

5.5.10. Public and operator dose reduction

While existing HWRs meet legal requirements for releases and doses, and are comparable to other reactor types, the ‘as low as reasonably achievable’ (ALARA) concept requires continued dose optimization. Public dose and operator dose should really be optimized as a package, since reducing operator dose (e.g. by more frequent containment air changes) can, if not done in an integrated fashion, increase public dose. About one third to one half of the dose is derived from tritium. A number of measures have already been taken on both operating plants and current designs, e.g. better material specification in HTS components to reduce activation, provision of inlet air dryers to the containment ventilation system, increased dryer capacity within containment, better segregation of high tritium and low tritium areas, and use of welded feeders on CANDU 9. Naturally, the more leaktight the heavy water systems are, the greater the tritium buildup over time.

One strategy for reducing the occupational hazard posed by spills of highly tritiated water is therefore to remove tritium from the moderator as the station matures. Such capability already exists at, for example, the Darlington station in Canada, but the capital cost of that facility is quite high. Economic technologies for extracting most of the tritium are in the process of being demonstrated. It should be noted that the ‘end product’ of the Darlington tritium extraction plant is pure tritium. As regards hazard reduction, it is only necessary to extract a mixture of tritium and deuterium in which the concentration of the former is sufficiently high as to avoid an undue economic penalty. Tritium removal should, therefore, be seen as one option in an ALARA strategy. As is the case with other changes, the cost of dose reduction via this route should be carefully assessed against the benefits.

5.5.11. External events

The Wolsong Level 2 PSA described in Section 5.3.2.2 was the first appraisal of HWR external events from a probabilistic standpoint. It will provide a number of ‘lessons learned’ in terms of the dominant failure modes for beyond design basis external events, and will be a useful tool for designers, enabling them to enhance and optimize protection against severe accidents initiated by fires, flooding and earthquakes.

5.5.12. Safety analysis tools

Requirements for the safety analysis of HWRs are, of course, set by each national regulator. They have in common, unique among reactor types, the requirement to consider multiple failures as part of the design basis. Generally, the codes used are ‘best estimate’ physical models but the input data and assumptions are chosen so as to give conservative results. However, even this can be misleading when judging the ‘actual’ risk due to an accident in a PSA, or when trying to write abnormal event procedures, both of which require realistic analysis. It is, therefore, expected that the current conservative analysis will, over the next ten years, be supplemented and eventually replaced by a more realistic analysis. Regulators will not, of course, accept a purely realistic analysis; therefore, it will be coupled with an uncertainty analysis which identifies key parameters, illustrates the sensitivity to them, and gives bounds on the possible consequences. Such analyses will have to include uncertainties in:

- Fundamental physical models (as incorporated into the computer codes),
- Plant models,
- Station data and parameters.

Safety analysis tools for HWRs are, with the exception of severe core damage models, relatively mature. Formal validation and verification are internationally accepted requirements and should be completed for HWR tools in the near term. IAEA co-operative programmes such as the International HWR Standard Problems should be helpful in this respect. Some additional tools will need to be developed in order to address specific phenomena (e.g. three dimensional models of hydrogen transport in containment and combustion for dual failure and severe accidents), and to validate the effectiveness of igniters and recombiners.

Since HWR practice in most countries places the onus on the owner to make the safety case, rather than prescribing very detailed safety analysis requirements, it is relatively easy to incorporate the results of validated R&D into safety analysis codes. For this reason, HWR codes are likely to change with time, as the owner incorporates the results of R&D.

5.6. SAFETY ENHANCEMENT OPTIONS FOR NEXT GENERATION HWRs (TEN TO TWENTY YEARS)

5.6.1. General approach

As indicated in Section 5.4.2, the development path of HWRs for the foreseeable future will be an evolutionary one. Initially, this may seem likely to make it difficult to progress to the next generation HWRs, incorporating advanced and passive features, but a logical path is possible, based on cost benefit and risk reduction considerations.

An overall ‘vision’ of the desirable characteristics of advanced HWRs is first developed. An outline (but still conceptual) design of the plant, particularly its proposed safety related systems, is then established with the principal features. It is expected that the advanced design will incorporate a number of passive safety features, owing to their simplicity and reliability, but they will not be completely passive, since active systems will be retained where safety and reliability are adequate or demonstrably sufficient. Moreover, passive features offer one way of addressing what is becoming more and more apparent as requirements for future reactor designs: that there be long time frames before significant active human intervention is needed, that accidents evolve over longer time frames by virtue of the design features, and that there is no need for evacuation of the public in the case of an accident because releases are low or negligible at the site boundary. Such an objective can be achieved by either active or passive systems, i.e. it is possible to design either an active or a passive system, or produce a hybrid active/passive design, to meet a numerical safety target. However, passive designs can achieve availability targets with a higher confidence level. Passive designs offer more relative safety in the sense that their reliability is higher or can be more readily demonstrated, particularly in a hybrid design. A hybrid design, thus, invokes the concept of diversity and redundancy in safety system principles of operation, a feature of current HWRs which have two independent and diverse shutdown systems. Therefore, the concept of evolution and the combining of the best features of active and passive systems are highly attractive.

However, the passive approach is not an end in itself. It must pass the test of practicality, i.e. it must be demonstrated (not just assumed) that a passive system is indeed:

- More reliable (e.g. as a result of the absence of powered valves, pumps, motors and control logic). The reliability of passive systems must be demonstrated, even with subtle failure modes involving human error in design, maintenance and operation rather than mechanical or control failure. In addition, some passive systems can be difficult to test (although it can also be argued that little testing is required).
- Easier to maintain through reductions in the number of components, trains, inspection needs and replacement parts.
- More economical. This includes savings made during operation as a result of less maintenance and testing, as well as in capital, construction and installation costs. The savings due to simplicity are offset by the fact that passive systems tend to be larger than active ones because of their lower energy transport rates per unit volume.
- ‘Safer’ in operation, design and consequence, which can also be more of a perception than an independent attribute. If a system is functionally adequate and meets its availability target, it is safe regardless of whether it is active or passive. Passive systems can be safer because they are less sensitive to unknown cross-links which may invalidate the calculated availability of active components.

Having chosen a conceptual design for the advanced HWR, aspects of it (on a system basis) can then be incorporated into current evolutionary HWR designs. Thus, the next generation HWR is not built afresh but incorporates elements on an evolutionary basis. In this way, there is no need for a prototype plant, although a prototype plant is not precluded, if it has the required support.

Of the three safety functions (shutdown, cool, contain), passive safety generally concentrates more on heat removal systems from the core and from containment.

The shutdown systems in HWRs are already passive in execution; once initiated by a signal, they are actuated by gravity (shut-off rods, GRAB) or stored energy (poison injection). There is probably little benefit in trying to make the detection aspect passive; it is preferable to concentrate on removing some of the initiating events that require shutdown.

Containment is currently partially passive, the exceptions being isolation, hydrogen recombination and cooling. There is no reason why these functions cannot be made more passive and there are incentives to do so. A passive containment ventilation isolation system has been noted in Section 5.5.3 and CANDU 9 has passive hydrogen recombiners. Some countries have opted for a double containment. While there are certain advantages to this concept, use of a double containment is more likely to be a national requirement than vendor initiated. A double containment is not in itself necessarily passive if it requires active coolers, heat rejection, spray systems, valves, overpressure control and active hydrogen recombiners. A truly passive design would require *no* active initiation or manipulation for a designated (and long) time-frame. As noted in Section 5.2.2.5, the choice of containment type is not linked to the HWR concept.

Aspects of the next generation HWR for which passive heat removal systems make sense are those which have a low energy density. Hence, the most likely application of passive systems will be in decay heat removal, either from the reactor, the steam generators, the moderator, the shield tank and end shields, or the containment. Since decay heat has to be removed only once, the same ultimate heat sink can be used for all these locations. Interestingly enough, the HWR design lends itself easily to passive heat removal because of the presence of a large body of cool, low pressure water located about one centimetre away from the fuel bundles — the moderator; and another large body of cool, low pressure water surrounding the calandria — the shield tank/calandria vault. Several passive concepts make use of one or more of these water inventories. The relatively large containment volume per unit power and the low reactor energy density (large distributed core volume per unit power) of HWRs also provide room for stored water and assist passive heat removal.

One ‘vision’ is therefore a water storage tank which can accept decay heat from any or all of the above sources for a period of time long enough for accident recovery to be credible. In the case of passive heat removal, it would have to be elevated with respect to the heat sources, and large enough to absorb decay heat for several days. The following sections give details of such a concept.

Current HWRs have a positive void coefficient and require prompt shutdown after a large LOCA. The provision of two, independent, diverse, reliable, testable and fully capable shutdown systems means that those accident sequences involving a LOCA (or other initiating event) and failure of *both* shutdown systems are in the residual risk category. Some HWR fuel cycles described elsewhere in this section do use enriched fuel for other reasons, in order that some 'tuning' of the void coefficient is possible: a smaller, near zero void coefficient would relax the requirements on the actuation times of the shutdown systems but would not materially change the design concept.

5.6.2. Passive emergency water system (PEWS)

An important passive safety feature under consideration for future CANDU reactors is the PEWS, which is a vented water pool in the containment dome [113]. It serves as a general purpose emergency heat sink, having a volume of some 2000 m^3, which would provide, after boil off, a three day heat sink for a large reactor. It will serve as a heat sink for various functions and systems as described below and is illustrated in Fig. 120.

To provide redundancy of supply, the tank can be subdivided, either by azimuth, subtanks, or multiple tanks. To provide natural circulation heat rejection and adequate flow, the supply sources should be elevated relative to the reactor (or heat source), to enable sufficient flow for heat removal, but not so high as to increase containment height (and cost) or increase the seismic risk (thus substituting one risk for another). However, the dependency of flow on the elevation varies, typically, according to the square root of the height, and, therefore, is relatively insensitive to the elevation, whereas the containment free volume is directly proportional to the height. Heat rejection to and from the tank itself can be effected via passive heat exchangers, natural circulation coolers and air heat exchangers to the external environment. In that sense, the water tank constitutes an intermediate heat sink, the ultimate heat sink being the external environment or the atmosphere itself.

In addition, the PEWS containment design allows for natural circulation of the air, steam and any hydrogen within, so as to effect cooling via heat exchangers. This avoids long term overpressurization of the containment. To that end, the pathways, flow loops and convection patterns must be inherent to the design, and not added as an afterthought.

Sizing of the PEWS tank is then governed by the decay heat removal magnitude and time scales, tank failure leakage scenarios, natural circulation pathways (both inside and outside the structure), and the chosen elevation.

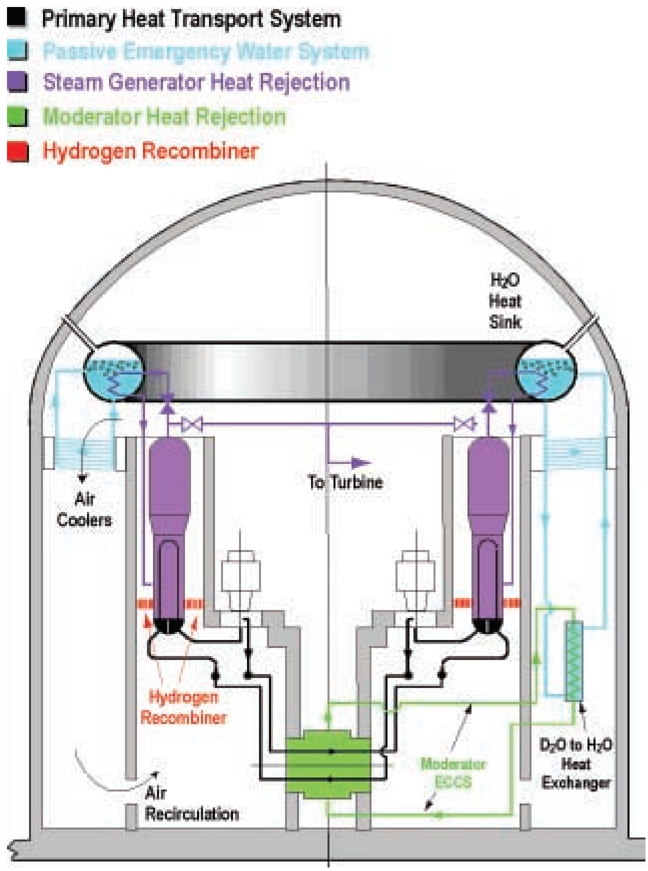


FIG. 120. A schematic diagram of the PEWS.

5.6.3. Moderator system

In current designs, the role of the moderator in severe accident mitigation could be strengthened by ensuring make-up at a rate sufficient to remove decay heat by natural circulation and boiling (steaming). The present moderator heat removal systems are sized for ~4.5% full power, equivalent to the decay heat shortly after shutdown. Thus, there is already a built-in heat rejection system, which can be made passive by designing natural circulation loops for decay heat removal from the moderator (calandria). In addition, a moderator make-up system provides backup longer term cooling.

The next step is passive moderator heat removal in severe accidents. This is effective except in accidents where the moderator can drain (i.e. pressure tube/calandria tube combined failure plus an induced failure of the annulus bellows would have to be prevented or moderator make-up provided).

In the PEWS concept, as a specific example, moderator heat rejection is effected through a boiling by flashing natural circulation D_2O loop, which in turn rejects heat to a natural circulation H_2O loop connected directly to the PEWS tank. The moderator is allowed to run near saturation (in order that its heat can be used to improve station thermal efficiency through feedwater preheating). The stability of the heat removal process in the natural circulation loop under flashing conditions has been demonstrated in tests at AECL, where it has been shown that adequate heat removal can be achieved. The number of loops then simply depends on the magnitude of the required heat rejection, but 4 × 50% capacity loops would give a very high passive reliability.

The safety advantage gained is that the role of the moderator in severe accidents is more passive, depending only on the presence of water in the PEWS tank. Use of the controlled heat transfer fuel channel (Section 5.7.1.1) would, in addition, permit rejection of decay heat to the moderator without the need for depressurization via the steam generators. Thus, a built-in heat rejection pathway from the fuel to the PEWS tank is ensured.

Hydrogen in the cover gas is currently actively recombined. An adjunct to the passive moderator heat removal would be recombination of hydrogen in the cover gas using passive recombiners, which have now been tested and qualified to accident conditions.

5.6.4. ECCS

Water is a cheap emergency core coolant. However, it produces large volumes of steam in the feeder pipes which slow down refill of the core. Analysis shows that cooling is ensured but is relatively insensitive to the flow rate above that needed to cool the metalwork. Water also reacts chemically with the Zircaloy sheaths at high temperatures, producing hydrogen. In addition, an emergency core coolant injection requires an isotopic upgrade of the D_2O/H_2O mix afterwards, in order to recover the heavy water. Alternative emergency core coolants which overcome these disadvantages would be attractive to HWR owners.

The use of passive actuation devices for high pressure emergency core coolant injection from the accumulator tanks has been shown to be effective in CANDU 9: it also reduces the number of active valves. It can be envisaged that the ball valves will also include transponders or magnetic locators, which indicate passively their location, motion and sealing. In addition, such valves can be used for other systems.

5.6.5. Containment system

Passive containment and steam generator heat removal concepts developed for other reactor types could equally apply to HWRs. Most concepts use a large water tank within containment as a decay power heat sink sufficient for several days' operation. Again, as a specific example, in the case of passive cooling of the containment under accident conditions, a water filled tube bank is located at an upper elevation within containment. Water from the PEWS is supplied to, and returns from, the tube side of the tube banks by natural circulation via vertical headers.

As in existing HWR designs, the containment itself is divided into an inner zone and an outer annular zone (accessible areas). By providing a special connection between these two zones near the lower elevation (see Fig. 120), natural circulation of gases up through the steam generator enclosure and down through the accessible area can be facilitated. A reduced containment pressure follows from a high rate of heat transfer from the containment gases to the banks of water cooled tubes and passive recombiners placed strategically in the flow path. Thus, a long term passive heat and hydrogen removal system can be designed.

5.6.6. Steam generator heat removal

The PEWS water tank could also be used to absorb steam rejection from the steam generators when they are removing decay heat from the reactor.

In the PEWS concept, the steam generators would be isolated from the main steam line after an accident by active valve arrangements and be connected to steam condensers located in the PEWS. The condensers would depressurize the steam generators and the condensate would return by gravity to the steam generators. This return flow eliminates the need to supply emergency water to the steam generators. As the need for blowing steam into the atmosphere is eliminated, activity release into the atmosphere will not take place, which could otherwise happen in the event that certain accident scenarios involving steam generator tube leaks occur.

In addition, the moderator heat can be rejected to feedwater preheaters for the steam generators via the so-called feedwater heat rejection system. This not only provides economy of operation (less waste heat) but also a redundant heat rejection pathway for moderator heat removal. In essence, long term moderator heat removal is guaranteed by both the PEWS and the feedwater heat rejection system, both of which are capable of removing decay heat after shutdown, and in the long term. The elevated condensate storage tank also provides an additional natural circulation source of cooling water for the steam generator secondary side.

5.6.7. Ultimate heat sinks

There are two ultimate heat sinks: the atmosphere and any local water bodies. Thus, the containment can be designed to reject heat to the atmosphere via conduction through the walls, and reject heat from the PEWS to the outside by steaming or by the use of air-cooled heat exchangers. The former results in a reduction, over time, in the PEWS inventory, and therefore closed loop heat rejection is preferred for longer time-frames. As with LWRs, double-shelled containments could be adopted to provide a natural circulation pathway in the interspace between the shells, thereby avoiding additional penetrations. Alternatively, and preferably, air-cooled heat exchangers could be placed in the upper containment structure. Should these fail, the rejection of heat through the boiling of the passive energy water still exists, and hence a dual (redundant) ultimate heat rejection pathway exists.

5.6.8. Passive flow monitoring

While passive heat removal is a useful goal, it does not by itself address all safety issues since there are other mechanisms that can cause fuel failures without impairment of the capability to remove decay heat. Single channel events and fuel handling accidents can produce a radioactive source term in containment which is independent of passive systems. Passive methods of detecting single channel blockage are therefore being examined, including detection of pipe breaks (which is a partial solution), and the combined passive monitoring and 'imaging' of the feeder flow rates using a combination of channel power distribution and D_2O residence time in the channels. It may be technically possible to detect single channel events such as flow blockage and feeder stagnation breaks, and to shut down the reactor before fuel in the channel is significantly damaged. In addition, the high temperature channel concept includes a double feeder entry to provide inherently for a reduced probability of damaging flow blockage.

5.7. OPTIONS BEYOND TWENTY YEARS

Nuclear power plants currently compete against fossil fuels in power generation. Although in the long run fossil fuels will become scarce and expensive, the opposite has been happening in the last few years in many countries. In the longer term, required reductions in the use of carbon based fuels and restrictions on gaseous hydrocarbon emissions may lead to the widespread need for high efficiency and short construction time HWRs. It would not be prudent to judge the viability of the HWR on the basis of expensive fossil fuels, even in the twenty year period. The major cost of an HWR comes from the capital cost component (about 80%), with fuel and O&M accounting for about 10% each. New fuel cycles, discussed elsewhere, can therefore provide savings if they allow the plant to operate at a higher power level. However, in order to make a major impact, HWR thermal efficiency (currently about 30%) must be increased. This implies the use of high temperature coolants, either water at a higher temperature or even at supercritical conditions, or entirely different substances. LWR designers, faced with similar challenges, are also investigating supercritical water. The use of an alternative coolant also reduces expenditure on the heavy water.

This section discusses two approaches: the Canadian considerations for HWRs using alternative or supercritical coolants, and the Indian AHWR approach.

The designs will not be driven by safety but will incorporate different safety characteristics. There is no reason in principle why the passive safety systems for decay heat removal described in Section 5.6.1 could not also be applied to advanced HWR designs, but detailed work has yet to be done in this area.

5.7.1 Safety aspects of CANDU advanced HWRs

The HWR design could evolve in terms of coolant temperature and enthalpy, from conventional pressures and temperatures to supercritical pressures and temperatures (see Fig. 121 which is based on the properties of light water, and Section 8.2.3). Two stages of development of a supercritically cooled HWR with coolant core-mean temperatures approaching 400°C and 500°C have been studied by AECL [114]. Designated mark 1 and mark 2, respectively, these stages are based on the use of heavy or light water coolant operating at a nominal pressure of 25 MPa. Mark 1 transfers heat from a heavy water primary system to a light water secondary system operating at 19 MPa and is expected to operate with conventional or near conventional zirconium alloy clad fuel. Mark 2 requires advanced fuel and operates with heavy water to light water, or light water to light water, in an indirect cycle, or with light water in a direct cycle.

The advanced HWR has four passive safety features:

- Passive high temperature channel (no failure in accidents),
- Elimination of consequence of channel flow blockage,
- Natural circulation heat removal wherever possible,
- Passive containment heat removal.

The second, third and fourth items have been discussed when dealing with the PEWS.

5.7.1.1. Controlled heat transfer fuel channel

Both concepts require a new fuel channel design. With a conventional HWR fuel channel, an increase in coolant pressure would require an increase in pressure

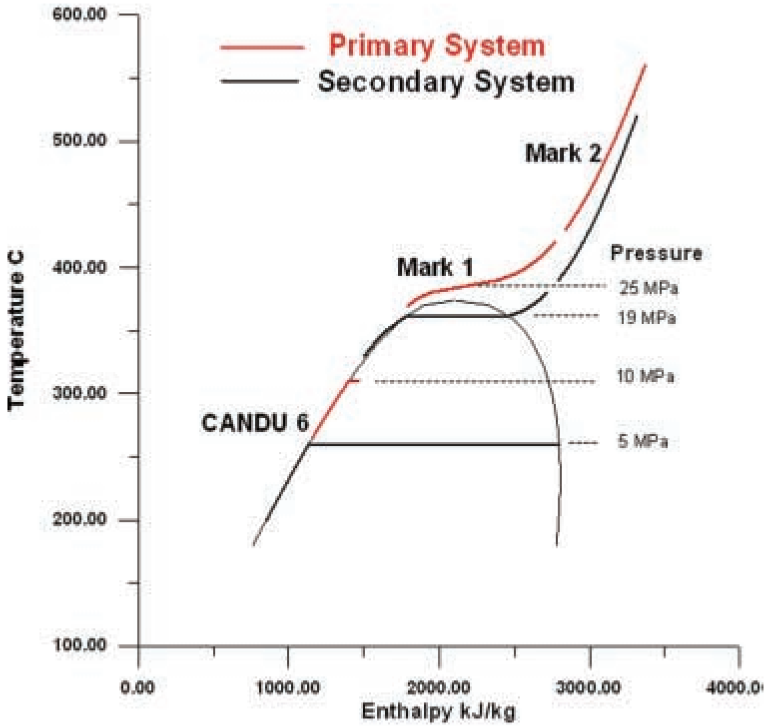


FIG. 121. HWR thermodynamic conditions.

tube thickness and a loss of neutron economy. To preserve neutron economy, especially at high coolant temperatures, an insulated fuel channel (CANTHERM) is being developed. This provides the option of having no calandria tube, with the cold pressure tube in contact with the heavy water moderator, and insulated from high temperature coolant (Fig. 122, and Section 8.6).

The concept comprises choosing the insulating material in order to achieve two objectives:

- To transfer sufficient heat at decay power, from the fuel through the insulating material and the pressure tube to the moderator, and to ensure that the fuel is not damaged in an accident, even if the accident results in loss of coolant in the channel. This provides, in effect, a fully passive ECCS.
- To achieve heat losses from the channels to the moderator in normal operation which are acceptable from the point of view of station thermal efficiency.

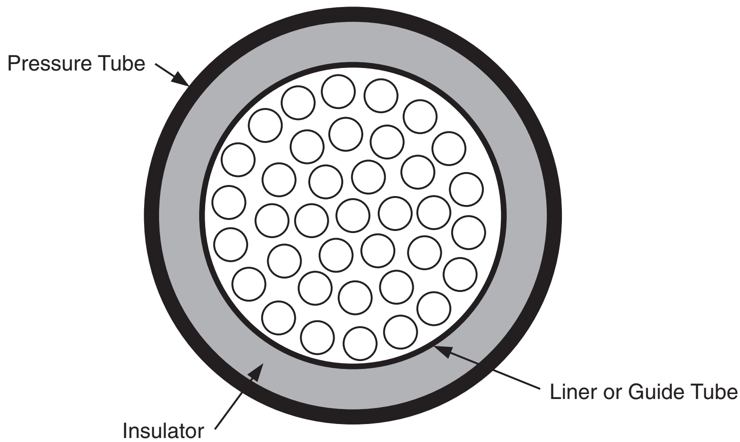


FIG. 122. Insulated pressure tube concept.

Normal heat losses may even be larger in this concept than they are in current HWRs; some of the heat could be recovered if the moderator were operated at saturation conditions in a natural circulation loop as described in Section 5.6.3.

Operating conditions that keep the pressure tube cold increase both the strength and the creep resistance of the pressure tube so that no neutron economy penalty is incurred as a result of the higher coolant pressure. Such a fuel channel could be employed to increase coolant temperatures and pressures in a conventional HWR HTS, or it could be employed, as envisaged here, in a redesign for supercritical coolant conditions.

From a safety point of view, the CANTHERM channel opens up the possibility of passive heat rejection from the fuel to the moderator, either with or without heat removal from the coolant, but without incurring as much fuel damage as in a LOCA plus LOECC.

5.7.2. Passive safety features of the Indian AHWR

5.7.2.1. Introduction

The Indian AHWR is being designed as a vertical, pressure tube type, boiling light water cooled and heavy water moderated reactor. The reactor is designed to produce most of its power from thorium, aided by a small input of plutonium based fuel. The reactor will have a large number of advanced safety features, such as passive safety systems which do not require either external power or operator action for activation. Safety aspects include:

- Reactor physics design ‘tuned’ for using thorium based fuel, and with negative void coefficient of reactivity;
- Advanced coolant channel design features, with easily replaceable pressure tubes;
- Passive systems for core heat removal, containment cooling and containment isolation.

5.7.2.2. Negative void coefficient of reactivity

The reactor physics design of the AHWR has been based on several requirements which include the achievement of a slightly negative void coefficient of reactivity.

The fulfilment of these requirements has been made possible through the use of PuO_2–ThO_2, and ThO_2–$^{233}UO_2$ in different pins of the same fuel cluster, and the use of a heterogeneous moderator consisting of amorphous carbon and heavy water in a 4:1 volume ratio. The negative void coefficient of reactivity considerably simplifies the burden on the reactor regulating system.

The ^{233}U enriched ThO_2 based fuel has a positive void coefficient and is subcritical, whereas the MOX fuel has a negative void coefficient of reactivity. With the proper combination of MOX and ^{233}U–ThO_2 pins in a cluster, it is possible to achieve an overall negative void coefficient of reactivity under all operating conditions. As a result of this inherent feature, the reactor will be shut down automatically if there is any increase in voiding due to any transient or accident condition. Achieving a slightly negative void coefficient for the AHWR core configuration requires a tighter lattice pitch than that required for a conventional HWR. This is illustrated in Fig. 123.

5.7.2.3. Engineered passive safety features

(a) General description of passive features in the AHWR

Apart from establishing a slightly negative void coefficient of reactivity, the AHWR incorporates several other passive features. These include the following:

- Heat removal through natural circulation (driven by thermosyphoning) under both normal operation and hot shutdown conditions;
- Direct injection of ECCS water into fuel;
- Passive systems for containment cooling and isolation;

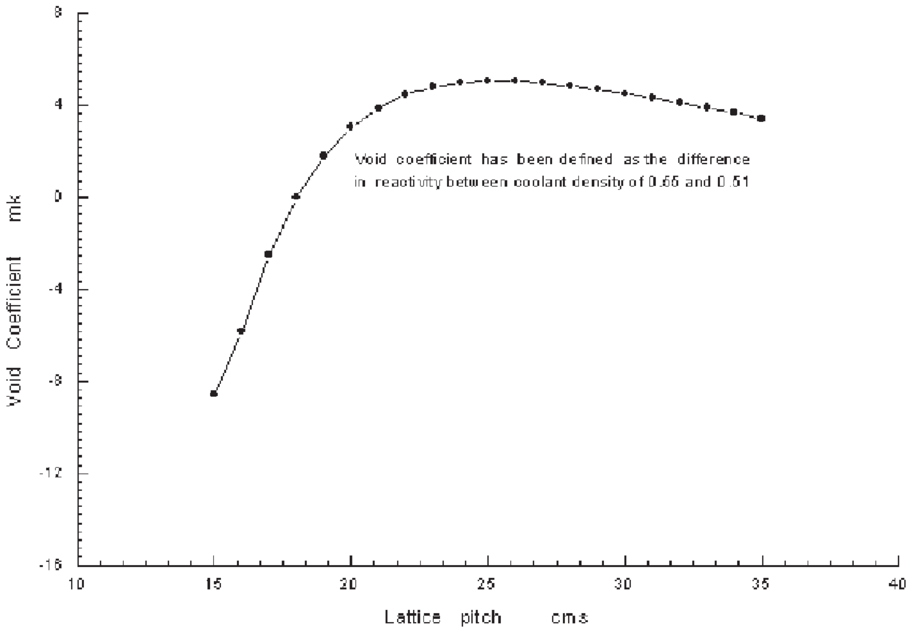


FIG. 123. Void coefficient in the AHWR as a function of lattice pitch with heavy water moderator alone.

- Availability of a large inventory of borated water in an overhead GDWP to facilitate sustenance of core decay heat removal, ECCS injection and containment cooling for three days without invoking any active systems or operator action.

(b) Natural circulation of primary coolant

During normal reactor operation, full reactor power is removed by natural circulation. The necessary flow rate is achieved by locating the steam drums at a suitable height above the centre of the core, taking advantage of the reactor building height. Figure 124 shows the variation in HTS flow rate as a function of power for the design configuration of the reactor.

By eliminating the need for nuclear grade HTS circulation pumps, associated valves, instrumentation, power supply and control system, the plant is made simpler, less expensive, and easier to maintain compared with options involving

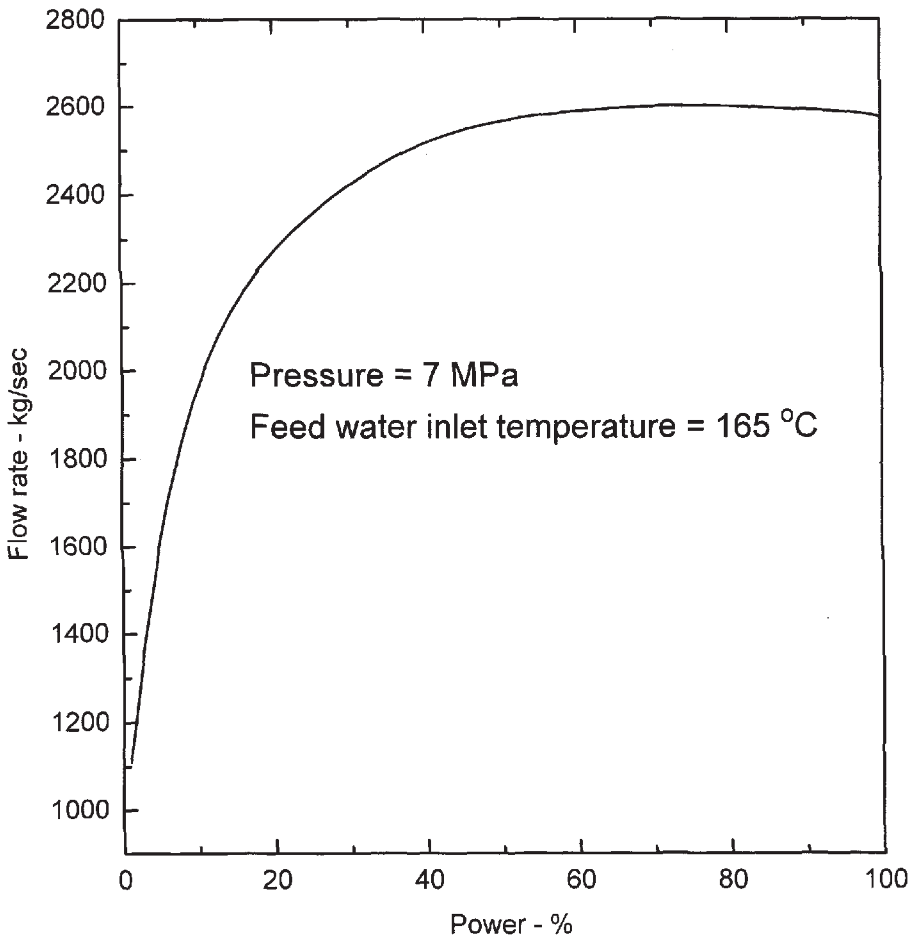


FIG. 124. Effect of power on primary flow rate.

forced circulation in the HTS. This, in turn, leads to considerable enhancement in system safety and reliability since pump related transients have been removed. A major experimental programme has been launched to confirm the analysis leading to the determination of loop height, and to study the thermohydraulic stability of the HTS loop.

(c) Core decay heat removal system

During normal reactor shutdown, core decay heat is removed by passive means by utilizing isolation condensers submerged in a GDWP located above the steam drum. Core decay heat, in the form of enthalpy of steam, enters the isolation condenser pipe bundles through natural circulation. The steam condenses inside the pipes and heats up the surrounding pool water. The condensate returns by gravity to the core. The water inventory in the GDWP is adequate to cool the core for more than three days without any operator intervention and without boiling of GDWP water. A separate GDWP cooling system is provided to cool the GDWP inventory in case the temperature of the inventory rises above a set value. An active shutdown cooling system is also provided to remove the core decay heat in case the isolation condensers are not available.

(d) ECCS

During a LOCA, emergency coolant injection is provided by passive means to keep the core flooded and thereby prevent overheating of the fuel. The ECCS is designed to fulfil the following two objectives:

- To provide a large amount of cold, borated water directly into the core in the early stage of a LOCA and then a relatively small amount for a longer time to quench the core. This objective is achieved through use of a passive fluid flow control device.
- To provide water through the GDWP to cool the core for more than three days.

Long term core cooling is achieved by active means by pumping water from the reactor cavity to the core through heat exchangers.

The ECCS accumulators and GDWP are connected to the HTS by rupture discs, check valves and isolation valves which are in series. During reactor startup, accumulators and the GDWP are isolated by closing the isolation valves. When the HTS pressure reaches the operating pressure level, these isolation valves are opened. The nitrogen pressure in the accumulators is always maintained at 5 MPa to keep the system in a state of readiness. Following a postulated LOCA, when the HTS system pressure falls below 5 MPa, the rupture discs open out, allowing cold borated water from the accumulators to flow into the core. When the accumulators become exhausted, a low water level signal closes the isolation valves and water from the accumulator stops flowing into the core. At this stage, water from the GDWP starts flowing into the core under gravity. Through an optimum positioning of the discharge nozzles, the GDWP based ECCS flow rate is closely matched to the requirement for core decay heat removal, enabling an extended period of availability of ECCS flow (more than three days), as illustrated by the results in Fig. 125.

After three days, water from the reactor cavity (which is filled with hot water from the ruptured pipe, and water from the accumulators and the GDWP after cooling the core) is pumped back into the core through heat exchangers in a long term

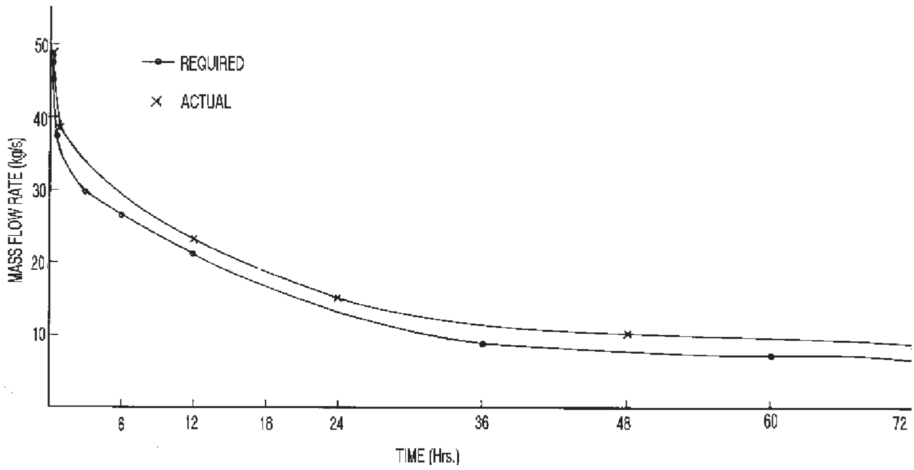


FIG. 125. GDWP flow rate for emergency core cooling.

recirculation mode. This heat is transferred, in the heat exchangers, to the process water which in turn dissipates its heat to the ultimate heat sinks, i.e. sea water or the atmosphere (via cooling towers).

(e) Core submergence

Following a postulated LOCA, water from the HTS, ECCS accumulators and the GDWP, after cooling of the core, will be guided and collected in the space around the core, termed the reactor cavity. Thus, the core will be submerged under water. In the unlikely event of failure of the GDWP to retain the water inventory, under any postulated scenario, the whole GDWP inventory will be collected in the reactor cavity and provide a heat sink for heat removal from the core.

(f) Failure of the ECCS during a LOCA

The AHWR contains, in and around its core, a large inventory of heavy water moderator and surrounding vault water. Although the possibility of failure of the ECCS is very remote, if the ECCS is not available during a LOCA, the fuel temperature will start to rise and ballooning of the pressure tubes will occur. Ballooning will cause the pressure tubes to come into contact with the calandria tubes and heat will be transferred to the moderator, and from the moderator to the vault water, thereby providing a large heat sink for the removal of core heat.

(g) Passive containment isolation

In the case of containment isolation, in addition to the normal inlet and outlet ventilation dampers, a passive system has been provided in the AHWR. The reactor building air supply and exhaust ducts are shaped in the form of U-bends of sufficient height. In the event of a LOCA, the containment becomes pressurized. This pressure acts on the GDWP inventory and transfers water, by swift establishment of a syphon, into the ventilation duct U-bends. Water in the U-bends acts as a seal between the containment and the external environment, providing the necessary isolation between the two. Drain connections provided to the U-bends permit the re-establishment of containment ventilation manually when desired. A schematic of this system is shown in Fig. 126.

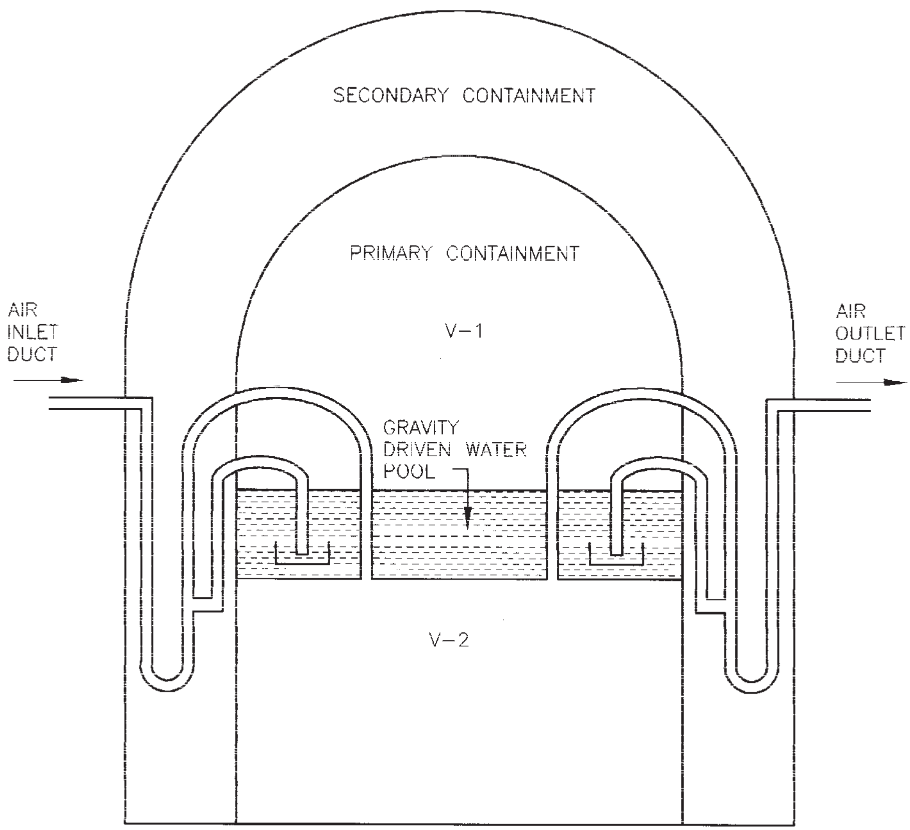


FIG. 126. Schematic of passive containment isolation system.

(h) Passive containment cooling

Passive containment coolers are utilized to achieve post-accident primary containment cooling by passive means and to limit the post-accident primary containment pressure. A set of passive containment coolers is located below the GDWP and connected to the GDWP inventory. During a LOCA, the mixture of hot air and steam is directed to flow over the coolers. Steam condenses and hot air cools down at the cooler tube surface and hence provides long term containment cooling after the accident.

5.8. CONCLUSIONS

This section has described the combination of inherent and engineered safety characteristics of the HWR. The design approach adopted for HWR development has been evolutionary. Thus, the safety characteristics of future HWRs will build on those existing already. In current designs, the moderator and shield tank prevent or slow down the progression of severe accidents. These functions can be strengthened by water make-up of these sources, or by passive heat removal, thereby providing the opportunity to create an HWR design which does not require public evacuation after an accident.

More generally, HWR designs in which decay heat is removed passively from the moderator, the steam generators and the containment, coupled with a controlled heat removal channel, offer not only improved safety but better investment protection.

In the longer term, the ability to compete with other forms of electricity generation will drive the HWR design towards the use of advanced coolant cycles such as natural circulation boiling light water or supercritical water. The safety characteristics of such designs are not as well known and need to be delineated, but advanced HWRs will retain many of the inherent characteristics of current HWRs.

The safety characteristics of the HWR — large inventories of cool water situated near the fuel, low core power densities, and neutron economy — lead to a 'forgiving' and robust behaviour during normal operation and in the event of accidents, including severe accidents. The evolution of HWRs, from passive designs to advanced high thermal efficiency designs, will continue to take advantage of these characteristics in ensuring HWR safety.

6. HWR FUEL CYCLES

One of the hallmarks of the HWR is its high neutron economy. While high neutron economy is a necessity for utilizing natural uranium fuel, it results in a reactor that has a very high degree of fuel cycle flexibility. This enables a country, or utility, to optimize its nuclear fuel cycle according to its own unique circumstances.

This section does not attempt to define a unique fuel cycle strategy for all time and for all countries. Rather, it describes the diverse range of advanced fuel and fuel cycle options that are available, and describes how these fuels and fuel cycles can address specific national or local objectives. It represents the current state of the art in HWR fuel cycle technology. A broad time-frame for implementation is suggested, and the document generally follows this, starting with the current natural uranium fuel cycle, then moving to the fuel and fuel cycle options that are likely to be implemented in the near term, followed by a description of the longer term options.

Section 6.1 describes key features of the natural uranium fuel cycle, including fuel design, fabrication and performance; uranium utilization and fuel cycle costs; reactor core characteristics; load following capability; spent fuel disposal considerations; and non-proliferation and safeguards features. The current natural uranium fuel cycle in its simplicity offers so many advantages that any advanced fuel or fuel cycle for HWRs must offer compelling benefits.

Section 6.2 describes in more detail those features of the HWR that result in high fuel cycle flexibility, and some of the drivers that will influence the choice and timing of implementation of these fuels and fuel cycles. The principal driver in the short to medium term will be the employment of advanced fuels and fuel cycles to reduce plant costs, both operating and capital costs.

In the short term, within the next five years, it is likely that advanced fuel bundle designs still employing natural uranium fuel will be introduced in HWRs to increase the thermohydraulic margins, as a plant life management tool, or to provide other specific benefits. Several advanced fuel bundle designs have been recently developed for HWRs: the CANFLEX bundle, jointly developed by AECL and KAERI, and featuring higher thermohydraulic margins and 20% lower linear element ratings compared with the 37 element bundle; the ‘SEU 43’ bundle of a different design developed in Romania specifically for use with SEU; a bundle (CARA) that can be used in both the conventional HWR and pressure vessel HWR in Argentina; and several advanced bundles developed in India which offer higher performance. These are described in Section 6.3, and they illustrate how the fuel design can respond to different drivers in different countries.

While natural uranium fuel provides outstanding advantages, the use of SEU in CANDU offers even greater benefits, and the use of SEU/recycled uranium in HWRs is the topic of Section 6.4. Recycled uranium from reprocessing spent PWR fuel is a subset of SEU that has significant economic promise. The use of SEU/recycled uranium is the first logical step after the use of natural uranium in CANDU, and it is anticipated that SEU or recycled uranium will be introduced on a large scale, as the reference fuel in both operating reactors and new reactors within the next ten years. In operating reactors, SEU/recycled uranium can provide one of few opportunities for significantly reducing operating costs, and in addition can provide reactor power uprating, where sufficient heat removal capability exists or can be provided during an outage. New reactor designs can be optimized for using SEU/recycled uranium (while still allowing the use of natural uranium), and along with an advanced fuel carrier, can offer significant reductions in capital cost over the next decade (through reactor power uprating capability that is not viable with natural uranium fuel). The benefits of enrichment extend to the back-end as well, with a reduction in spent fuel disposal cost as high as 30% compared with natural uranium fuel. The first enrichment level that will likely be introduced is around 0.9% SEU, which also corresponds to the nominal enrichment of recycled uranium. Higher enrichments, around 1.2%, would follow, once experience is gained at the lower level. Fuel management schemes have been devised to handle a wide range of enrichments.

In the case of countries having both LWRs and HWRs, the capability to use low fissile material in HWRs offers a wide range of unique synergistic fuel cycle opportunities, which are described in Section 6.5. These tend to be longer term fuel cycle options, requiring additional development. The section starts with a description of those unique fuel cycles that exploit the high neutron economy of the HWR by not separating the fissile material in spent PWR fuel before recycle in the HWR. The direct use of spent PWR fuel in CANDU (DUPIC) fuel cycle uses only dry processes to convert spent PWR fuel into a form useable in CANDU. Without purposeful removal of any isotopes, this recycle option has a very high degree of proliferation resistance. The TANDEM fuel cycle is a variant of conventional reprocessing in which the uranium and plutonium are co-precipitated, after some degree of removal of fission products. Both DUPIC and TANDEM are illustrative of recycle technologies that are potentially cheaper and simpler than conventional reprocessing, while having a higher degree of proliferation resistance. The last two advanced fuel cycle options described in this section utilize the products of conventional reprocessing: HWR MOX uses the plutonium, while actinide burning in an HWR using an inert matrix carrier offers a means of burning the true actinide and fission product waste. Development of the inert matrix carrier fuel is required for this cycle. Alternatively, existing HWRs could be used for this purpose.

Section 6.7 describes a fuel cycle option that is not likely to have widespread use for a very long time, but which may see use where security and independence of fuel resources are dominant considerations — recycling of self-generated plutonium from spent HWR fuel.

A unique 'swords to ploughshares' fuel cycle opportunity, employing HWRs to disposition plutonium recovered from dismantled warheads is covered in Section 6.8. This is a specific niche for CANDU MOX fuel that is technically a nearer term option, but whose deployment will be driven largely by political considerations. The HWR fuel cycle flexibility is uniquely suited to this mission, and the MOX fuel designed for this purpose can be used in 100% of existing cores, with no changes to the reactor.

Section 6.8 illustrates how the fuel cycle flexibility of the HWR allows a single reactor type to achieve a very different fuel cycle objective — plutonium annihilation in an inert matrix. This is similar to the actinide burning application discussed in Section 6.5.

Thorium is a nuclear fuel in which there is potential widespread interest in the longer term, and short term interest in those countries possessing extensive thorium resources, but lacking indigenous uranium supplies. Recycling in HWRs of the ^{233}U produced by irradiating ThO_2 can significantly reduce mined uranium requirements. Complete independence from uranium is theoretically possible in an HWR, with the self-sufficient equilibrium thorium fuel cycle, which in equilibrium produces as much ^{233}U as it consumes. Full exploitation of the energy potential of thorium requires recycling, which will not be economically justified for many years. Since commercial thorium fuel recycling facilities have not yet been built, there exists an opportunity to develop a new, cheaper, proliferation resistant technology for recycling. While the high burnup and recycle options are long term, the 'once through' thorium (OTT) cycle provides a bridge, in the short term, between current uranium based fuel cycles and a long term thorium fuel cycle based on the recycle of ^{233}U. The optimal OTT cycle is economic nowadays, in terms of both money and uranium resources. This cycle creates a resource of valuable ^{233}U, safeguarded in the spent fuel, for future recovery predicated by economic or resource considerations. Section 6.9 discusses this wide range of thorium fuel cycle options available to the HWR.

In the long term, the HWR reactor is synergistic with FBRs, with a few expensive FBRs supplying the fissile requirements of cheaper, high conversion ratio HWR reactors operating on the thorium cycle. Direct recycle of fuel between the two reactor types is theoretically feasible. This is explored briefly in Section 6.10.

This section covers the full gamut of advanced fuel and fuel cycle options, and concludes with a summary of the fuel cycle options that are likely to be employed in the next ten years, and the strategies that could be followed in the longer term.

Again, because the fuel cycle drivers vary over time, and from country to country, there is no unique 'optimal' fuel cycle path that is appropriate for all countries. Many local and global factors will affect the best strategy chosen for an individual country. Advances in HWR development will continue to follow an evolutionary path, but for the present day, and in the long term, the reactor offers a fuel cycle which can be adapted to fit local requirements. Its unsurpassed fuel cycle flexibility can accommodate the widest range of fuel cycle options in existing reactors to enable any country to optimize its nuclear strategy to suit its own needs.

The focus of this section is on the pressure tube HWR of CANDU design; specific reference to the Atucha 1 pressure vessel reactor will be noted where appropriate.

In general, the CANDU 6 reactor will be used as the reference, with exceptions noted for other reactors.

6.1. THE NATURAL URANIUM FUEL CYCLE

6.1.1. Overview

The HWR fuel bundle is relatively small (0.5 m in length, 10 cm in diameter), and is easily handled. The CANDU 37 element bundle is typical of that used in HWRs, but other fuel bundle designs are in use. The 37 element bundle weighs about 24 kg, of which more than 90% is uranium oxide fuel. It uses two main materials (Zircaloy and UO_2) and has a total of only seven components: pellets, sheaths, CANLUB coating, end caps, spacers, bearing pads and end plates. Hence, it is an easily manufactured product and one that countries using HWRs have found straightforward to manufacture locally (Section 6.1.3). The use of natural uranium fuel itself simplifies manufacture and handling, as well as the sourcing and diversification of fuel supply.

After more than three hundred reactor-years of operation, the failure rate of CANDU fuel remains very low—less than 0.1% bundle failure rate (Section 6.1.4). The capability to locate the infrequent defects that do occur, and to remove the failed fuel during normal on-power refuelling operations, minimize coolant system contamination and the economic effect of fuel defects. This is in contrast to other reactor types, where a fuel defect must be left in the core for an extended period, or an extensive shutdown undertaken. Reactivity mechanisms are not part of the fuel bundle assembly, again simplifying fuel manufacture and facilitating good fuel performance. Any dissolved neutron absorber that might be used for reactivity control is confined to the moderator, precluding the possibility of precipitation onto the fuel from the coolant.

The uranium requirements (mined uranium required per unit of electricity generated) are the lowest of any commercial reactor (Section 6.1.4.5). The use of natural uranium generates no depleted uranium enrichment plant tails waste and as a concequence is a more environmentally friendly front end of the fuel cycle.

A consequence of these factors is that fuel cycle costs in HWRs (per unit of electricity generated) are the lowest of any commercial reactor system (Section 6.1.7).

Section 6.1.8 summarizes some of the reactor physics core characteristics of the HWR that impact on, or are affected by, the use of advanced fuel cycles.

The lower HWR fuel burnup is not a disadvantage in the back end of the fuel cycle (Section 6.1.9). An extensive assessment of the Canadian concept for geological disposal has been completed and has confirmed its technical soundness [115]. The concept is based on deep disposal in an underground vault excavated in plutonic rock. The density of fuel emplacement in such a facility is determined primarily by the heat load of the spent fuel. The larger the quantity of spent natural uranium CANDU fuel compared with higher burnup PWR fuel is offset by its lower heat load. The simplicity and the small size and weight of the CANDU bundle also reduce the cost of the emplacement system. The overall disposal cost per unit of electricity produced is similar for spent natural uranium CANDU fuel and spent PWR fuel. This is borne out in the NEA assessment of disposal costs [116]. Also, the size of the repository is small, considering the electricity produced.

The final section deals with non-proliferation and safeguards aspects of the HWR fuel cycle.

Given the many advantages offered by the natural uranium fuel cycle, any advanced fuel or fuel cycle for HWRs will need to offer *compelling* benefits before it is introduced. The rest of this section will identify some of these benefits.

6.1.2. Fuel design

Natural uranium fuel is simple to use and easy to process into fuel assemblies. The short, simple fuel assemblies for the HWR (fuel bundles) are easily produced and the Republic of Korea, India, Argentina and Romania all have independent fuel fabrication facilities sufficient to meet their demands. China will build a fuel fabrication facility to meet the requirements of its CANDU reactors now under construction. Furthermore, HWR fuel cycle costs are low (because natural uranium is relatively inexpensive), uranium utilization (amount of energy produced from the mined uranium) is good, and fuel bundle design and manufacture are simple. There are also many worldwide suppliers of natural uranium, including Canada, Australia and the Russian Federation, ensuring security of supply to an HWR utility that may not have access to a large indigenous source of uranium.

While several fuel bundle designs are in use in HWRs, a typical design is the CANDU 6, 37 element bundle (Fig. 127). The fuel consists of natural uranium dioxide (UO_2) pellets contained in Zircaloy 4 sheaths that are resistance welded at both ends. The end caps serve three purposes by providing:

- A seal for the contents of the element,
- Effective element termination for attachment to the end plates,
- The structural component for interfacing with the fuel handling system.

FIG. 127. The 37 element CANDU 6 fuel bundle.

A thin layer of graphite (CANLUB) is applied to the inside surface of the sheathing to reduce the pellet/sheath interaction. The elements are held together by Zircaloy 4 end plates. The desired separations at the transverse midplane of the fuel bundles are maintained by spacer pads brazed onto the fuel elements. The fuel elements have Zircaloy 4 bearing pads brazed onto the outside surface to provide support in the reactor and to prevent damage to the sheaths and the pressure tubes during service.

The following summarizes some prominent features of the CANDU 6, 37 element fuel bundle design:

- Use of high density natural UO_2 pellets, which ensures dimensional stability and bundle dimensional compatibility with the fuel channel and fuel handling systems.
- Use of thin walled collapsible Zircaloy 4 cladding for neutron economy and improved heat transfer. Neutron economy ultimately leads to reduced electricity costs, and improved heat transfer leads to low temperature and high fission gas retention within pellets.

- Absence of gas plenum. Extensive operating experience confirms that no plenum is necessary to accommodate fission gases, thus maximizing the fissile content per bundle. This experience includes extended burnup experiments and post-irradiation examination of HWR power reactor fuel.
- Use of high integrity resistance welding of end caps, which results in good fuel reliability.
- Provision of CANLUB graphite layer between the UO_2 pellets and Zircaloy cladding. This has eliminated fuel failures due to power ramping under normal operating conditions.
- Use of induction brazed spacer pads, which maintain separation of the fuel elements without the need for complex, expensive spacer grids (Indian HWR fuel uses welded appendages).
- Employment of a simple bundle structure. This is possible because the pressure tube supports the fuel bundle, and all reactivity control mechanisms are external to the fuel bundle and fuel channel.

These features ensure low fuelling costs, good uranium utilization, high capacity factors and good fuel performance.

The HWR fuel design is qualified by out-reactor tests as well as by test irradiation of elements and bundles in experimental loops. Various out-reactor qualification tests have been performed, such as strength, endurance, impact and fuel handling compatibility tests. Some of these tests were performed in test sections that were representative of full size fuel channels. In order to allow for design margins, the test conditions were designed to be more severe than those predicted as occurring in the reactor. Subsequent examinations of the fuel and fuel channel components confirmed that the fuel design meets its design basis at exit burnup. Experimental irradiations of prototype elements and bundles were also performed in in-reactor experimental loops at high powers and to high burnups.

The present fuel bundle designs have evolved substantially from the 19 element design that was used in the early phase of the Douglas Point prototype reactor in Canada. At present, there are several HWR fuel designs that are being developed. In addition to the CANDU 6 37 element fuel bundles, there are the 28 element fuel bundles which are being used in the Pickering reactors. There is also another 37 element bundle design that is being used in the 850 MW(e) Bruce and Darlington reactors. The differences between the two 37 element bundle designs are related to the end cap profiles and the locations of the bearing pads, and are dictated by differences in the fuel channels in these reactors.

In India, the current 19 element fuel bundle design used in the operating plants represents an improved version of the wire wrap fuel bundle formerly used in the Douglas Point reactor. This fuel was first installed in Rajasthan 1 in 1972. At that time, a conscious decision was taken to fabricate half the initial charge in India, in order to establish the fuel fabrication capability. Simultaneously, the fuel test facilities were also constructed to carry out tests, such as pressure drop, strength, endurance and fuelling machine compatibility tests. These facilities were very useful subsequently for the out of pile qualification of later designs and for trouble shooting. In the wire wrap design, the gap between the elements and the gap between the pressure tube and the elements were maintained by wire wound helically around the elements and spot welded. In order to avoid the possible fretting damage being caused by the wires to the neighbouring element sheath surfaces, a split spacer design was adopted. Skewed split spacers maintain the gap between the elements. The spacer and bearing pads are attached to the elements by resistance welding, in contrast to resistance brazing which is used elsewhere for appendage attachment to CANDU fuel sheaths. To improve the power ramp capability of Indian fuel, graphite coating of the inner surface of the fuel pin has been adopted as a regular feature. The end cap design parameters are specified according to compatibility with the fuelling machine, welding, or minimum thickness considerations. A modified end cap design has been provided with a scooped-out conical portion, in contrast to the flat surface employed in the original design. This modification provides extra volume for the fission gases. Some aspects of Indian HWR fuel design and fabrication are described in Refs [117–119].

A description is given in Table XIV of the CANDU 6 37 element bundle, and the Indian 19 element HWR fuel bundle.

TABLE XIV. DESCRIPTION OF THE CANDU 6 37 ELEMENT BUNDLE AND THE INDIAN 19 ELEMENT HWR FUEL BUNDLE

Parameter	Fuel bundle type: CANDU 6 37 element bundle	Fuel bundle type: Indian 19 element bundle
Fuel material	Natural UO_2	Natural UO_2
Number of fuel rods in fuel bundle	37	19
Sheath material	Zircaloy 4	Zircaloy 4
Diameter of fuel pellet (cm)	1.215	1.44
Outer diameter of sheath (cm)	1.3075	1.52
Cladding thickness (cm)	0.042	0.038
Pitch circle diameter of first ring (6 pins) (cm)	2.977	3.30
Pitch circle diameter of second ring (12 pins) (cm)	5.751	6.36
Pitch circle diameter of third ring (18 pins) (cm)	8.661	
Inner diameter of Zircaloy pressure tube (cm)	10.34	8.26
Outer diameter of Zircaloy pressure tube (cm)	11.15	9.10
Inner diameter of calandria tube (cm)	12.90	10.8
Outer diameter of calandria tube (cm)	13.17	11.1
Calandria tube material	Zircaloy	Zircaloy

6.1.3. Fuel fabrication

The basic simplicity of HWR fuel has led to extremely low fabrication costs, because only a small industrial facility is required and personnel needs are moderate. The use of natural uranium also eliminates dependence on foreign sources of enrichment (although fuel cycle costs are actually reduced with SEU in the HWR). Thus, a country operating HWRs has the possibility to become independent of foreign nuclear fuel vendors within a fairly short period of time.

Fuel fabricators typically start with the components listed below and fabricate them into finished fuel:

- UO_2 powder for the production of UO_2 pellets,
- Zircaloy wire for the production of spacer pads and bearing pads,
- Zircaloy strip for the production of end plates,
- Zircaloy bar for the production of end caps,
- Zircaloy tubing for the production of fuel element sheathing,
- Beryllium metal for the production of braze joints,
- Graphite and organic solution for the production of the CANLUB coating on the internal surface of the sheaths.

The starting materials are purchased and the fuel bundles manufactured in conformance with the relevant technical specifications.

The conversion of U_3O_8 concentrate into UO_2 powder is performed at the uranium refinery and is not a fuel fabrication process. However, for completeness, the general approach to this process is also described in this section.

Figure 128 illustrates a typical line production flow of materials and inspection leading to a completed HWR fuel bundle. The quality standard that applies to CANDU fuel manufacturing is CSA Z299.2-85 (or its equivalent). This standard is used by all Canadian manufacturers of CANDU fuel.

6.1.3.1. Conversion of uranium concentrates into UO_2 powder

The UO_2 is produced at a refinery from yellow cake, typically (though not exclusively) by the ammonium diuranate process. Yellow cake, which is about 80% U_3O_8, is dissolved in nitric acid and the resulting uranyl nitrate solution purified in ion exchange columns. Uranate crystals are precipitated using ammonium hydroxide and the crystals are then filtered, washed and reduced.

Two methods of reduction are used: stationary bed and rotary calcining. The resulting oxide, which is a blend of the two reduction processes, is conditioned over a period of several weeks by the addition of dry ice, and is then delivered to the fuel fabricators.

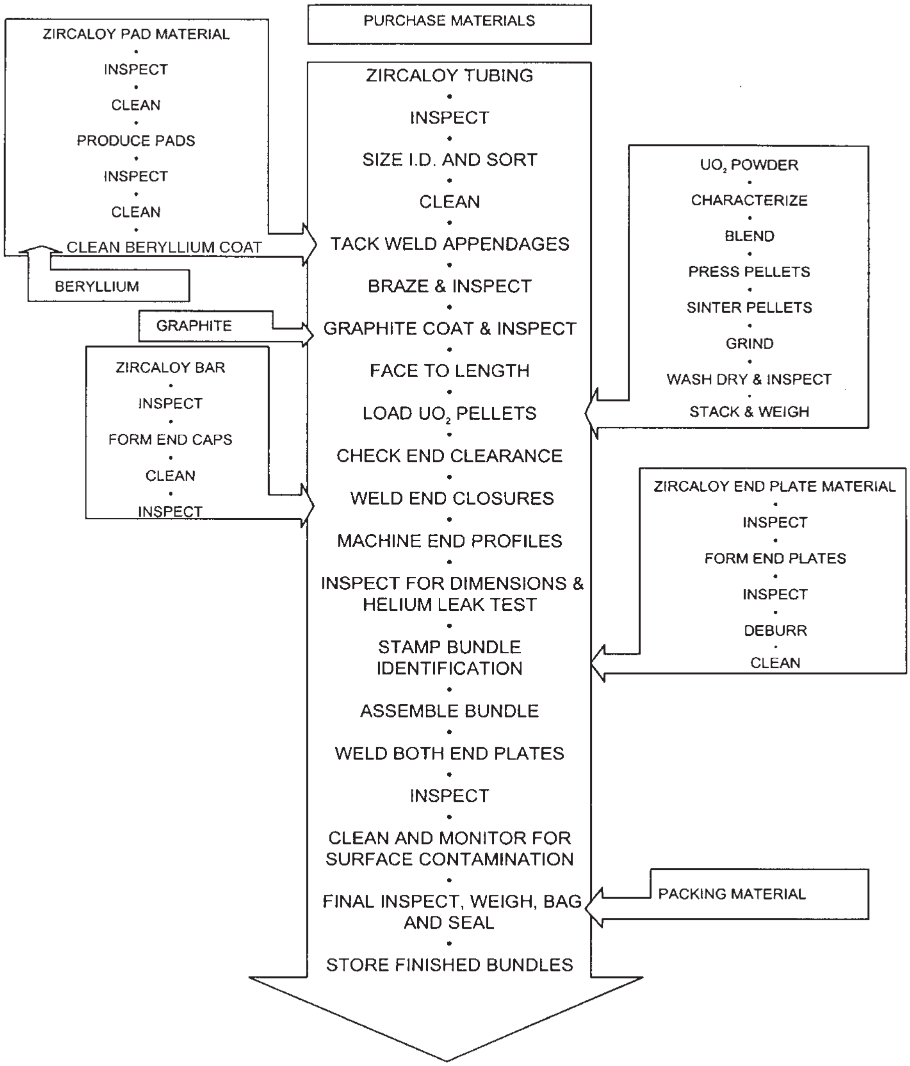


FIG. 128. The HWR fuel bundle fabrication process.

6.1.3.2. UO_2 pellet production

Uranium dioxide powder is received from the supplier in lot quantities that are suitable for planned production rates. Prior to using a specific lot, an evaluation sample is processed into pellet form and the pellets tested to ensure that they conform to the specification. In some cases, the parameters of the process may have to be altered to ensure that the pellets conform to the specification. If the pellets made from the evaluation sample meet the specification, the lot is accepted and the UO_2 powder is processed. The following procedure is typical.

The UO_2 powder is compacted on a press into wafers or slugs. The slugs are then broken up and passed through a fixed size sieve. This process increases the bulk density and improves the flow properties of the powder.

The granular material is fed into a pill press, which forms the powder into green (unsintered) pellets of predetermined weight, diameter and length. A statistical sampling plan is used to confirm that the process is producing green pellets that conform to the appropriate specification.

The green pellets are then placed in containers or boats which are passed continuously through a sintering furnace. The atmosphere in this furnace is hydrogen or cracked ammonia and the peak temperature is 1650–1700°C. Samples of the sintered pellets are checked for diameter, length, density and physical defects. Sample pellets are also subject to both chemical analysis and metallographic examination.

The sintered pellets are then ground to the required diameter on a centerless grinder, washed to remove the grinding coolant and sludge, dried, and again inspected dimensionally and for surface imperfections.

In contrast to reactor systems requiring enriched fuel, all manufacturing stages of natural UO_2 CANDU fuel can be accomplished without special criticality restrictions.

6.1.3.3. Component fabrication

(a) Spacers and bearing pads

When the Zircaloy material for the spacers and bearing pads is received, it is inspected for thickness, width, flatness and surface defects. The mill certificate (provided by the vendor) is checked for compliance with the material specification.

The material is then prepared (degreased or cleaned, dried, etc.) for loading into the beryllium coating machine, after which it is formed into spacers and bearing pads. The thin layer of beryllium is deposited by vacuum deposition. The pads and spacers are then stored, ready for brazing onto the sheaths.

It should be noted that Indian HWR fuel uses fully welded split spacers and bearing pads, without brazing.

(b) End caps

Zircaloy bar for the end caps is received and is inspected visually and dimensionally, and the mill certificate provided by the vendor is checked for compliance with the specification.

The bar is ultrasonically tested for defects, the most likely of which may take the form of a longitudinal hole in the centre. Ultrasonic standards have been developed that enable such holes to be identified, even if they are only 0.025 mm in diameter. The end caps are then machined, normally on an automatic screw machine. A visual and dimensional inspection is then carried out after which the end caps are cleaned, rinsed and dried. The end caps are then ready for use.

(c) End plates

The strip for the end plates is inspected visually and dimensionally, and the mill certificate provided by the vendor is checked for compliance with the specification.

The end plates are usually stamped from the strip and deburred. Dimensional and visual inspections and cleaning operations then follow.

6.1.3.4. Fuel element fabrication

On receipt in the shop, Zircaloy fuel sheaths are fully inspected, both visually and dimensionally, and the mill certificate from the vendor is checked against the specification. The sheaths are then ultrasonically tested for transverse and longitudinal defects, sorted according to their inside diameter, and then cleaned.

The bearing pads and spacers are tack welded to the sheaths, after which the brazing operation is performed (except in the case of Indian HWR fuel, where the bearing pads and spacers are welded). Brazing is conducted by induction welding in a vacuum chamber. After the brazing operations, another set of visual and dimensional inspections is performed. A destructive metallurgical examination to test for completeness of braze and to assess the amount of eutectic formed is also carried out on a statistical basis.

To provide the CANLUB coating for the elements of the CANDU 6 37 element fuel bundles, the insides of the sheaths are coated with graphite solution, dried in air and cured in a degassing vacuum furnace. The graphite coating is then inspected.

The sheaths are then cut to length, and the ends prepared for welding the end caps. The pellets are stacked to a specific length, visually inspected for cracks and chips, and loaded into the sheath. The loaded sheaths are then moved to the closure welder. There, the elements are backfilled with a mixture filling gas (80% helium) and the end caps welded in place with an automatically controlled resistance weld. The integrity of these welds is ensured by the careful preparation of the components and by micro-examination of the welds.

The elements are then placed in a profile lath to remove the weld flash. All elements are then cleaned, and a helium leak test performed on all of them.

6.1.3.5. Bundle assembly

The next operation is the assembly of the elements into bundles. This is done in a jig which ensures the correct placement and orientation of the elements and allows the end plates to be resistance welded to each element. The bundles are inspected visually and dimensionally and are then packed for transportation to the site.

As a control procedure, a percentage of the bundles is inspected in more detail to confirm that the bundle meets all specification requirements.

6.1.4. Fuel performance

6.1.4.1. Canadian fuel performance data

The in-reactor performance of CANDU fuel has been proven by its continuing successful use in operating CANDU reactors. Of the 1.4 million fuel bundles irradiated in Canada up to 1996, more than 99.9% of them have performed as designed. About half of the 0.1% of defects can be attributed to a single cause — stress corrosion cracking sheath failure, which was caused by power boosts during the early refuellings of Pickering 1 and 2 in 1972, and by overpowering of the core of unit 1 in 1988. Since the introduction of the graphite CANLUB sheath coating in 1973, there have been very few confirmed power boost defects during normal operations. Improved fuel management and adjuster rod sequencing, developed through operational experience, are also partly responsible for this reduction in defect rate.

Table XV shows the cumulative bundle defect rate over a ten year period to 1996 for 37 element fuel bundles discharged from the 14 large CANDU reactors in Canada.

The cumulative bundle defect rate for 37 element fuel in Canada is about 0.04 %. Less than half of these failures are attributed to fabrication and unknown causes. Each CANDU fuel bundle consists of 37 elements or rods, and usually, one

TABLE XV. CANDU 37 ELEMENT FUEL DEFECT EXPERIENCE IN CANADA 1985–1995

Reactor	Bundle defect rate (%)
CANDU 6s (Point Lepreau and Gentilly 2 (1983–1995)	0.06 (in 114 000 bundles)
Bruce A (1985–1995)	0.04 (in 164 000 bundles)
Bruce B (1985–1995)	0.04 (in 213 000 bundles)
Darlington (startup to 1995)	0.03 (in 74 000 bundles)
Total for above CANDUs	0.04 (in 565 000 bundles)

element per bundle failed. Therefore, in the period 1985–1995, the annual defect rate for fuel elements among the 14 large CANDU reactors in Canada that use 37 element bundles was about 0.65×10^{-5} [120].

The 1997 annual bundle defect rate for CANDU 6 reactors was 0.073% in 25 000 bundles discharged from the six operating CANDU 6 reactors (Point Lepreau, Gentilly 2, Wolsong 1 and 2, Cernavoda 1 and Embalse). About half of these bundle defects are attributed to the debris fretting among the newer units that have been placed in service over the last two years. Construction debris in the primary circuit tends to become trapped within fuel bundles and causes defects during the initial years of operation.

CANDU fuel reliability and experience have, therefore, been comparable to those of the PWR. Furthermore, the operational implications of fuel defects are significantly less in CANDU reactors since on-power refuelling (which is a routine operation) and the on-power failed fuel detection and location systems allow the removal of defected fuel without having to shut down the reactor.

6.1.4.2. Indian fuel performance data

Failed fuel detection in Indian HWRs is conducted by monitoring of various fission gas isotopes of iodine and the noble gases. A delayed neutron system has been provided in all reactors and is operated once a day to check for fuel failure. Close monitoring of ^{131}I activity is done. Even though technical specifications for operation permit 100 μCi/L of ^{131}I in the PHTS, a significantly lower alert level of 10 μCi/L is adopted by operations. The design and fabrication of, and changes to, operational guidelines have brought the iodine levels down to around 2 μCi/L. The current fuel failure rate is 0.06% [121].

6.1.4.3. Romanian fuel performance data for Cernavoda 1 (1997–1998)

The first fuel charge for the Cernavoda 1 reactor consisted of 4560 bundles, which were supplied by Zircatec Precision Industries (ZPI) of Canada. However, a limited quantity (66 natural bundles) of domestic fuel, manufactured by FCN-Pitesti, was included in the first charge for testing purposes. After the onset of reactor refuelling in January 1997, almost all the fuel loaded into the core was domestic.

A total of 597 refuellings were performed in 1997. ‘Swing 8’ was used for the first visit at 274 channels and the normal eight bundle fuelling shift was used for 320 refuellings. The remaining three refuellings were performed with a four bundle shift scheme, this scheme being used in addition to a single refuelling with eight bundles to unload a suspect defective bundle initially located towards the channel ends.

Out of a total of 4764 bundles loaded into the core in 1997, only 141 were ZPI fuel, the remainder being Romanian fuel. The average burnup of the 4764 bundles discharged in 1997 was 5.99 MW·d/kg HE. A total of 4560 bundles (equivalent to a fuel charge) have been discharged up to full power day 391. The average burnup of the first 4560 bundles unloaded was 5.94 MW·d/kg HE, including the 160 depleted bundles loaded in the initial core which had an average of only 5.02 MW·d/kg HE. The last depleted bundle was discharged after 278 full power days.

During 1997, eight bundles suspected of being defective were discharged from the core (five supplied by ZPI and three Romanian). Seven of these defects were very small, and four of them were not even detected by the delayed neutron monitoring system. However, a small increase in gamma activity was detected during refuelling (the coolant in the fuelling machine is monitored for fuel defects). These defects could have been small 'pinholes' through the sheath, not large enough to allow a release of halogens or to be detected by the delayed neutron monitoring system. In 1997, there was only one fuel defect which was considered 'serious', and this was detected by monitoring the gamma activity of the coolant in the refuelling machine.

In 1998, 632 refuellings were performed, all using a regular eight bundle shift fuelling scheme. The number of bundles loaded into the core totalled 5056 (all supplied by FCN-Pitesti). The average burnup on the 5056 bundles discharged in 1998 was 7.12 MW·d/kg HE.

The number of suspect defective bundles discharged from the core in 1998 was four (one supplied by ZPI and three Romanian). All these defects were very small. Only two depleted bundles were fuelled/refuelled from the core.

In summary therefore, by the end of 1998, after more than two years of commercial operation, the fuel performance at Cernavoda 1 reactor was found to be very good, as illustrated by the following statistics:

- Number of refuelled channels: 1231.
- Number of loaded/discharged bundles: 9836.
- Number of irradiated bundles: 14 396.
- Number of suspect defective bundles discharged: 12.
- Bundle defect rate: 0.083%.
- Average fuel discharge burnup: 6.53 MW·d/kg HE (157 MW·h/kg U, or 2989 MW·h/bundle).

6.1.4.4. Republic of Korea fuel performance data

Since it commenced commercial operation, Wolsong 1 has operated for about 4847 effective full power days, with a lifetime capacity factor of around 85%. A total of 82 016 fuel bundles had been loaded by the end of 1998. The first fuel charge for Wolsong 1 consisted of 4560 bundles, supplied by Canadian General Electric of Canada. In 1984 and 1985, a limited quantity of domestic fuel (48 bundles manufactured by KAERI) was loaded for testing purposes. A total of 40 615 KAERI

TABLE XVI. FUEL DEFECT EXPERIENCE IN WOLSONG 1 (1983–1998)

Period	Fuel defect rate (%)	
	Domestic fuel	Canadian fuel
1983–1994	0.08	0.09
1995–1996	0.84*	0.0
1997–1998	0.0	0.0

* Domestic fuel defects during 1995–1996 were attributable to excess hydrogen being generated by a large scale manufacturing incident.

bundles had been loaded into the core by the end of 1998, while a total of 41 401 Canadian General Electric bundles had been loaded into the core. Of the 82 016 fuel bundles irradiated in the Republic of Korea up to 1998, around 99.8% have performed as designed.

The cumulative fuel bundle defect rate is about 0.2% (0.33% for domestic fuels and 0.07% for Canadian fuels (see Table XVI)). If the fuel defect excursion which occurred in 1996 as a result of a manufacturing problem were excluded, the cumulative domestic fuel bundle defect rate would be less than 0.1%; almost the same as for Canadian built fuel. Since 1996, the in-core flux detector assemblies in Wolsong 1 have been replaced, some improvements have been made in the refuelling practice, and a new fuel management system has been developed, all of which have resulted in an improvement in fuel performance. Since 1997, there have been no fuel defects in Wolsong 1, even though more than half the core has been reshuffled for spacer location and repositioning (repositioning the garter springs that separate the pressure tube from the calandria tube). The average accumulated burnup is 7.08 MW·d/kg HE.

Wolsong 2 has operated for about 509 effective full power days and of the 10 892 fuel bundles that had been loaded up to the end of 1998, 10 844 bundles were ZPI fuels, the remainder being domestic fuels for testing purposes. The cumulative fuel bundle defect rate is about 0.02% (ZPI fuel only).

6.1.4.5. Argentine fuel performance data

In Argentina, the fuel manufacturing company is CONUAR SA, which is a subsidiary of CNEA. CONUAR manufactures fuel assemblies for both operating reactors — the pressure vessel type Atucha 1 and the CANDU Embalse reactor – in spite of the large differences in geometry, design and mounting of the two types of fuel assembly. It should be noted that the Atucha 1 fuel assembly is much closer to standard PWR fuel, though with a 6 m long cluster fuel rod configuration. The manufacturing plant currently receives zirconium cladding tubes and other fuel assembly components from FAE SA, while UO_2 powder is provided by DIOXITEK SA, both of which are CNEA subsidiaries.

This industrial chain has been performing very well for several years, after the normal adjustments were made during the early stages. This has been reflected in the irradiated fuel performance data, which fluctuates from the fixed targets during the early operation of each reactor. A steady, low, failure rate has been observed both for Atucha 1 and for Embalse over quite a long period. In the case of the Embalse reactor, the fuel bundle failure rate during 1998 was 0.04%, while the historical failure rate over the past five years (1994–1998 inclusive) was 0.15%.

In the case of the pressure vessel type Atucha 1 reactor, the equivalent figures are quite similar. In both cases, on-line coolant activity monitoring and post-irradiation examinations are adequate tools for fuel failure detection and analysis.

The total number of irradiated fuel bundles in dry storage to the end of 1998 was 31 860.

6.1.5. Load following capability

Canadian utilities normally meet the base load requirements of the grid with the most inexpensive source of electrical energy, primarily with hydroelectricity and nuclear power. Since the nuclear component of energy production has increased in recent years for some utilities, HWRs may be expected to operate in various modes of load following in the future. These operations may require fuel to be power cycled on a weekly, daily or even hourly basis, depending on the grid. This requirement is especially important for those utilities whose reactors are connected to comparatively small grids.

One consideration in a postulated load following operation is its effect on the performance and integrity of the fuel bundle. There are several references dealing with the load following performance of HWR fuel [122–126]. A current summary is given below, primarily featuring information from power reactors at Bruce B and Wolsong stations, from experimental irradiations at Chalk River Laboratories and from analytical assessments that cover operational conditions that are beyond the existing database [127].

For nine months during 1986, the three commissioned reactors in the Bruce B station achieved extensive load following. The frequency, duration and nature of these manoeuvres varied considerably; typically the frequency varied from nil to three per week, with the duration of reduced power being approximately eight hours. The station was operated with deep load following (power reduction ≥40%) of up to 19 cycles, plus up to 65 comparatively shallower cycles (between 0% and 40% reduction in power) and 11 trips. If the trips are included, the above manoeuvres add up to a total of 95 cycles.

During the load following period, the fission product levels in the coolant increased in Bruce B unit 6 and 17 ‘new’ fuel defects were detected at the station. Each case of fuel failure was thoroughly followed up with root cause determination in the spent fuel bay, and in hot cells in some cases. The investigations revealed that 15 of the failures were caused by debris fretting (not uncommon in a newly commissioned station); the remaining two failures were caused by manufacturing flaws (porous end caps). Thus, although fuel failures were detected during the load following operation in Bruce B, their root causes were not directly related to the reactor power manoeuvres. The above operating experience gained with Bruce B is regarded as being very positive.

Load following operations have also been conducted at the KANUPP reactor in Karachi, Pakistan (about 90 cycles) and at the Embalse reactor (about 30 cycles). In addition, fuel is frequently exposed to repeated power cycles in the vicinity of liquid zone controllers because of fluctuations in the water levels in the liquid zone controller units brought about by specific refuelling patterns. Some fuel failures have occurred. However, fuel performance under the above conditions has not been followed up by the same degree of post-irradiation fuel examination as has been the case for the Bruce B fuel noted above. Hence, the effect of power cycles on detailed fuel integrity parameters in these reactors is not currently known with full confidence.

Fuel operating experience gained from the Wolsong 1 reactor shows that limited load following (e.g. many shallow cycles over a few days) does not cause fuel failure. Likewise, the following experimental irradiations at the Chalk River Laboratories showed no fuel failures during repeated power cycles: X-218 (490 cycles, 0–40 kW/m), X-411 (95 cycles, 0–70 kW/m) and U-900 (up to 103 cycles, power reduction >25%).

In some situations, different sets of operating or design conditions, or both, need to be considered. For example, the average residence period for natural uranium fuel in a CANDU 6 reactor is about 250 full power days in the inner core and about 270 full power days in the outer core. The corresponding maximum residence periods are about 330 and 700 full power days respectively. Therefore, during daily load following, natural uranium fuel in a CANDU 6 reactor can be expected to encounter a few hundred power cycles—considerably more than the Bruce B cycles, noted above. Further, the nature of the refuelling power ramps differs between the Bruce B and CANDU 6 reactors because of differences in their fuelling schemes. Moreover, the operating power levels also differ between the Bruce B and CANDU 6 reactors. All these parameters influence the integrity of the fuel sheath during load following.

In the case of a full duty cycle consisting of power cycles and refuelling ramp(s), the loads and the defect thresholds are expected to be influenced by a variety of parameters, including the severity of the refuelling ramp(s), the size(s) and number of the power cycles, the hold period(s) at low power, the power level and its history, fuel burnup, details of the fuel design and manufacturing to an acceptable quality level. These parameters interact in a complex manner, and hence extrapolation of the available operating and design data to other operating and design conditions requires the assistance of suitable scientifically based computer codes.

The analytical assessments performed to date, have focused on determining the risk to fuel integrity caused by stress corrosion fatigue of the sheath, arising from expansions and contractions of the pellet. The failure mechanism was considered to be the combination of stress corrosion cracking caused by the refuelling ramps, plus the additional damage caused by corrosion assisted fatigue from the power cycles. The assessments concentrated on conditions prevailing at the circumferential ridge of the sheath (between fuel pellets).

The amount of static and cyclic hoop strain at the circumferential ridge of the sheath was calculated by using the ELESTRES code [128]. The effect of sheath bending at the ridge was considered by using the FEAST code [129]. Sines law was used to incorporate the effect of multiaxiality in stresses/strains. The level of corrodent concentration was also calculated by the ELESTRES code. The effects of irradiation damage on the microstructure of Zircaloy, and on the fission product corrodent, were obtained by combining the existing failure criteria for stress corrosion cracking with the available 'S–N' curves for fatigue of Zircaloy. The Palmgren–Miner law was used to sum the cumulative damage from the refuelling ramp and the load following cycles of various amplitudes.

The results of the assessment indicate that the fuel in CANDU 6 reactors is expected to survive daily or weekly load following. Of course, load following, in combination with refuelling ramps, will reduce operating margins. If needed, the performance margins can be increased by a variety of options, such as reducing sheath stresses by the use of improved pellet shape and modified clearances, and reducing corrodent concentration by employing the CANFLEX fuel design to reduce the element ratings.

CANDU stations are equipped to deal with fuel failures should the need arise. First, different regions of the core impose different loads on the fuel bundle, and at the same time, as-built fuel varies in strength. Therefore, only some fuel, if any, would initially exceed the operational limit. Any initial fuel failures will give the operator advance warning, and time to react and contain the situation. Second, most CANDU stations are equipped with gaseous fission product and delayed neutron monitoring systems for detecting and locating failed fuel. Careful and frequent monitoring will help detect early indications of any impending problems, and help prevent the situation from overwhelming the gaseous fission product/delayed neutron systems. Third, any suspect or failed fuel can be removed promptly by the on-power fuelling system. Fourth, the coolant purification system can be used to clean up any released activity.

The extent of these capabilities differs in different CANDU stations, e.g. delayed neutron versus gaseous fission product and differences in fuel operating margins. Thus, even if load following beyond the existing database should lead to erosion of fuel performance margins, the above features, where available, can help deal with the initial consequences, especially with careful operation of the reactor. This monitoring can be done more effectively in stations that have a stronger capability to detect, locate and discharge failed fuel, and which have higher operational margins with regard to fuel performance.

In conclusion, operational feedback from three Bruce B reactors shows no evidence of fuel failure from stress corrosion fatigue during the performance of 0–3 reactor power manoeuvres per week for nine months. Fuel irradiation experience gained from the research reactors is also encouraging. Analytical assessments for daily load following during operation at a high power envelope in a CANDU 6 reactor suggest that the fuel will likely survive this load following operation, albeit with reduced margins to failure. Thus, CANDU fuel is expected to show good performance in both base load and in load following modes of operation.

6.1.6. Uranium utilization

The excellent neutron economy of the HWR stems from a number of design features dictated by the use of natural uranium fuel. Neutron absorption is minimized by the choice and strategic use of reactor structural materials, as well as by the heavy water moderator and coolant. In addition, on-power refuelling minimizes the excess reactivity required to maintain criticality. This eliminates the need for any neutron absorbing materials which are used to suppress the initial excess reactivity associated with refuelling an LWR core.

A consequence of the high neutron economy of the HWR is high uranium utilization. Uranium utilization is a measure of the amount of electrical energy extracted from a given amount of natural uranium, and has units of MW·d/Mg U (U as U_3O_8). Uranium consumption is a measure of the amount of natural uranium required to produce a given amount of electrical energy, and has units Mg U/GW·a (U as U_3O_8). Although natural uranium HWR fuel achieves an apparently low average burnup of only about 20–25% of enriched PWR fuel, HWRs extract about 38% more energy per Mg of U_3O_8 than do PWRs [130]. The HWR, therefore, makes efficient use of natural uranium resources. The use of natural uranium fuel also avoids the production of enrichment tails.

The HWR's efficient use of neutrons also contributes to its fuel cycle flexibility and can be a key to a country being self-sufficient in energy. In countries rich in thorium, fuel based on ThO_2 offers a potential HWR fuel cycle that would provide assurance of nuclear fuel supply long into the future. Other HWR fuel cycles include the use of SEU, recycled uranium from the reprocessing of PWR spent fuel, MOX fuel, DUPIC and inert matrix fuel. The HWR, therefore, allows a country to become independent of foreign fuel supply and technology. HWR design features leading to fuel cycle flexibility are discussed in Section 6.2.

Uranium utilization for various fuel cycles and PWR/HWR reactor systems is illustrated in Table XVII, while the fuel cycle data used to calculate the characteristics are given in Table XVIII.

6.1.7. Fuel cycle costs

Excellent uranium utilization and a simple fuel bundle design help minimize the HWR fuel cycle unit energy costs, in absolute terms, and relative to other reactor types [130]. The bundle, with its limited number of components, is easy to manufacture and there are no criticality concerns associated with the handling or transporting of natural uranium. At the station, on-power refuelling contributes to high capacity factors. Upon discharge from the reactor, criticality is again of no concern, and the lower average burnup results in lower decay powers and shorter cooling periods. This translates into higher packing densities for ultimate disposal, which offset the greater amount of spent fuel produced (see Section 6.1.9). Fuel cycle costs per unit of energy for the HWR are about half those of LWRs [130].

6.1.8. HWR physics core characteristics

6.1.8.1. Summary

The HWR design offers a number of economic, performance and safety advantages from the point of view of reactor physics. Overall, HWR physics is characterized by: high neutron economy, reactivity characteristics which remain relatively constant throughout operating life and power manoeuvring, and reactivity coefficients which are generally very small and which facilitate simple reactor operation with a high degree of control automation.

Figure 129 is a schematic of a CANDU reactor assemby, while Figs 130 and 131 are top and front views of the reactor, which show the location of channels and reactivity devices.

6.1.8.2. Reactor neutron efficiency

The HWR has excellent neutron economy. This derives from the choice of heavy water as coolant and moderator, which has a very small absorption cross-section for neutrons (the neutron absorption cross-section of deuterium is 0.0016 that of hydrogen), as well as from the use of low neutron absorbing structural materials and on-power refuelling. The neutron economy of the HWR allows the use of natural

TABLE XVII. URANIUM CONSUMPTION/UTILIZATION DATA

Fuel cycle option	Natural U consumption Mg U/GW(e)·a	Uranium utilization MW(e)·d/Mg (U as U_3O_8)	Equivalent natural U burnup MW(th)·d/kg	Spent fuel arisings Mg HE/GW(e)·a
PWR with enriched U	218	1670	5.1	33
PWR with re-enriched recycled U, recycled	182 (-16%)	2000 (20%)	6.1 (19%)	28
PWR with Pu recycled	185 (-15%)	1970 (18%)	6.0 (18%)	28
PWR with re-enriched recycled U, recycled and Pu recycled	160 (-27%)	2280 (37%)	7.0 (37%)	24
CANDU reactor with natural U	157 (-28%)	2320 (39%)	7.5 (48%)	157
CANDU reactor with 0.9% SEU	119 (-45%)	3070 (84%)	9.9 (95%)	84
CANDU reactor with 1.2% SEU	116 (-47%)	3150 (89%)	10.2 (101%)	56
PWR/CANDU (2.8/1)*, with recycled U, recycled in a CANDU reactor	161 (-26%)	2280 (37%)	7.0 (38%)	25
PWR/CANDU (2.9/1)*, with recycled U, recycled in a CANDU reactor, and Pu recycled in a PWR	143 (-34%)	2550 (53%)	7.9 (55%)	22
PWR/CANDU (1.4/1)*, with recycled U plus Pu recycled in a CANDU reactor (TANDEM)	129 (-41%)	2830 (69%)	8.8 (73%)	20
PWR/CANDU (2.4/1)*, with DUPIC fuel in a CANDU reactor	154 (-29%)	2370 (42%)	7.4 (45%)	24

Note: Figures in parentheses are percentage change from reference PWR. System uranium requirements and fuel disposal requirements for recycling options refer to an equilibrium system in which the fresh fuel requirements of the ‘receiving’ reactor (either CANDU or PWR) are exactly met by the spent fuel discharge rate of the ‘supplying’ PWRs.
* Ratio of PWR/CANDU electrical generation in equilibrium.

uranium fuel and this results in great fuel cycle flexibility. Good neutron economy is maintained if the reactor is operated with enriched fuel or with advanced fuel cycles, and this is reflected in high fuel utilization levels compared with other reactor types for a range of fuel cycles (Table XVII).

TABLE XVIII. FUEL CYCLE DATA AND FORMULAS

CANDU reactor:	
Net thermal efficiency (η_1)	0.31
Burnup with natural U (B_1) (MW·d/kg HE)	7.5
Burnup with recycled U from PWR spent fuel (0.9% U-235) ($B_{1,\ RU}$) (MW·d/kg HE)	13
Burnup with 0.9%/1.2% SEU ($B_{1,\ SEU}$) (MW·d/kg HE)	14/21
Natural U feed for 0.9%/1.2% SEU (F) (kg U/kg HE)	1.41/2.06
Burnup with recycled U and Pu from PWR spent fuel (TANDEM) ($B_{1,\ TANDEM}$) (MW·d/kg HE)	25
Burnup with DUPIC fuel ($B_{1,\ DUPIC}$)	15
PWR reference:	
Net thermal efficiency (η_2)	0.33
Burnup (B_2) (MW·d/kg HE)	33
U-235 content in fresh fuel (% U-235 in U)	3.25
U-235 content in enrichment plant tails (% U-235 in U)	0.25
Natural U feed (F) (kg U/kg HE)	6.51
U-235 content in spent UO_2 fuel (% U-235 in U)	0.92
U-236 content in spent UO_2 fuel (% U-235 in U)	0.41
PWR with re-enriched recycled U:	
Burnup (B_2) (MW·d/kg HE)	33
U-235 content in re-enriched recycled U (%)	3.62
U-236 content in re-enriched recycled U (%)	1.23
U-235 content in enrichment plant tails (%)	0.25
Recycled U feed (NR_{RU}) (kg recycled U/kg HE) (also equal to the number of PWRs required to provide re-enriched recycled U for 1 PWR)	5.03
PWR with recycled MOX:	
Number of PWRs required to provide Pu for 1 MOX PWR (NR_{Pu}) (33 MW·d/kg)	5.65

Formulas: Equilibrium natural U consumption (Mg U/GW(e)·a)

PWR or CANDU reactor with enriched U:

$365F/(\eta B)$, with the appropriate values of F, η and B

PWR with re-enriched recycled U: $365NR_{RU}F/[\eta_2 B_2(NR_{RU} + 1)]$

PWR with Pu recycle: $365NR_{Pu}F/[\eta_2 B_2(NR_{Pu} + 1)]$

PWR with re-enriched recycled U and Pu recycle:

$$365NR_{Pu}F/[\eta_2 B_2(NR_{Pu} + 1 + NR_{Pu}/NR_{RU})]$$

TABLE XVIII. (cont.)

CANDU reactor with natural U:	$365/[\eta_1 B_1]$
PWR/CANDU, with recycled U recycled in a CANDU reactor (the factor 0.956 is the fraction of U present in the PWR spent fuel):	$365F/(0.956\ \eta_1 B_{1,\ RU} + \eta_2 B_2)$
PWR/CANDU with recycled U recycled in a CANDU reactor and Pu recycled in PWR:	$365NR_{Pu}F/[0.956\eta_1 NR_{Pu}B_{1,\ RU} + \eta_2 B_2(NR_{Pu} + 1)]$
PWR/CANDU (TANDEM) (the factor 0.966 is the fraction of U and Pu in the PWR spent fuel):	$365F/(0.966\eta_1 B_{1,\ TANDEM} + \eta_2 B_2)$
PWR/CANDU (DUPIC):	$365F/(0.985\eta_1 B_{1,\ DUPIC} + \eta_2 B_2)$

Note: $F = (E-0.25)/(0.711-0.25)$, where E is the percentage ^{235}U (3.25% for PWR, 0.9% or 1.2% for CANDU).
The ^{236}U concentration in spent PWR fuel is assumed to be 0.41%, and in the enriched recycled uranium product is 1.23%. An additional 0.3% ^{235}U enrichment is needed for every 1% ^{236}U in the final product: $NR_{RU} = \{[(3.25+1.23)0.3]-0.25\}/(0.92-0.25) = 5.03$ (RU = recycled uranium).

6.1.8.3. Characteristics of the HWR core

The HWR maintains very low excess core reactivity. Core reactivity characteristics change very little during fuel residence in the core. This is made possible by on-power refuelling, since reactor criticality can be maintained indefinitely by replacing fuel on a daily or quasi-daily basis. This avoids the necessity of having large amounts of soluble poison in the moderator or burnable poisons in the fuel. Typically, the excess reactivity ‘hold down’ in an HWR is about 20 mk, compared with over 200 mk at the start of a fuel cycle in a PWR. Only small amounts of excess reactivity are needed in the light water zone control compartments for bulk reactivity and spatial control, and a desired level of xenon override capability can be designed in the adjuster rods. The small excess reactivity is a safety advantage, as the amount available to be added to the core in accidents is limited.

The HWR lattice features a long neutron migration length. This is characteristic of the moderating properties of heavy water and explains the use of a relatively large lattice pitch (28.6 cm for a CANDU reactor), which in turn makes possible the pressure tube design and the consequent capability for on-power refuelling.

The large lattice pitch and the long neutron diffusion time in the HWR lattice yield a very long prompt neutron lifetime, approximately 1 ms in the case of a CANDU reactor, which slows the evolution of power in transients, especially power excursions that are near prompt critical.

Some of the characteristics of a typical CANDU 6 core and the Indian HWR cores are shown in Table XIX.

6.1.8.4. Other HWR physics aspects

In the pressure tube design, the high pressure, high temperature coolant is separated from the moderator. As a result, the moderator can be maintained at a relatively low temperature (which benefits the neutron economy) and pressure. The reactivity devices operate in the benign moderator environment, in the space between

TABLE XIX. GENERAL CHARACTERISTICS OF NATURAL URANIUM HWRs

Parameter	CANDU 6	Indian HWRs	
		500 MW(e)	220 MW(e)
Nominal reactor power (MW(th))	2061	1830	802
Number of fuel channels in the core	380	392	306
Number of fuel bundles per channel	12	12	10
Lattice pitch (cm)	28.575	28.575	22.86
Number of elements in reference fuel	37	37	19
UO_2 weight per bundle (kg)	21.8	21.6	15.2
U weight per bundle (kg)	19.1	19.0	13.4
U-235 weight per bundle (g)	135.6	135.4	95.1
Core average discharge burnup (MW·d/kg HE)	7.5	6.7	6.3
Maximum time average channel power (kW)	6600	5500	3080
Maximum time average bundle power (kW)	800	642	420
Maximum instantaneous channel power (kW)	7000	6215	3300
Maximum instantaneous bundle power (kW)	880	736	462
Zone control system worth/regulating rods (mk)[a]	7.0	6.7	5.7
Adjuster rod system worth (mk)	16	11.5	9.0
Mechanical control rods/shim rod worth (mk)	11	10	7
SDS1 worth (mk)[b]	87	72	33.9
SDS1 (one or two rods not available) worth (mk)	57	50	28.4
SDS2 (liquid poison injection) worth (mk)[c]	>300	300	32.2
Fuelling scheme (bundles per shift)	8	8	8
Fuelling rate (bundles per full power day)	15	15	10
Delayed neutron fraction	0.006	0.006	0.006

[a] In the CANDU 6, the 7 zone controller units provide 14 compartments; in the Indian HWR, there are 4 absorber regulating rods which provide reactor regulation.

[b] In the 220 MW(e) HWR, the SDS1 comprises 14 shim rods.

[c] In the 220 MW(e) HWR, the SDS2 comprises 12 liquid poison tubes.

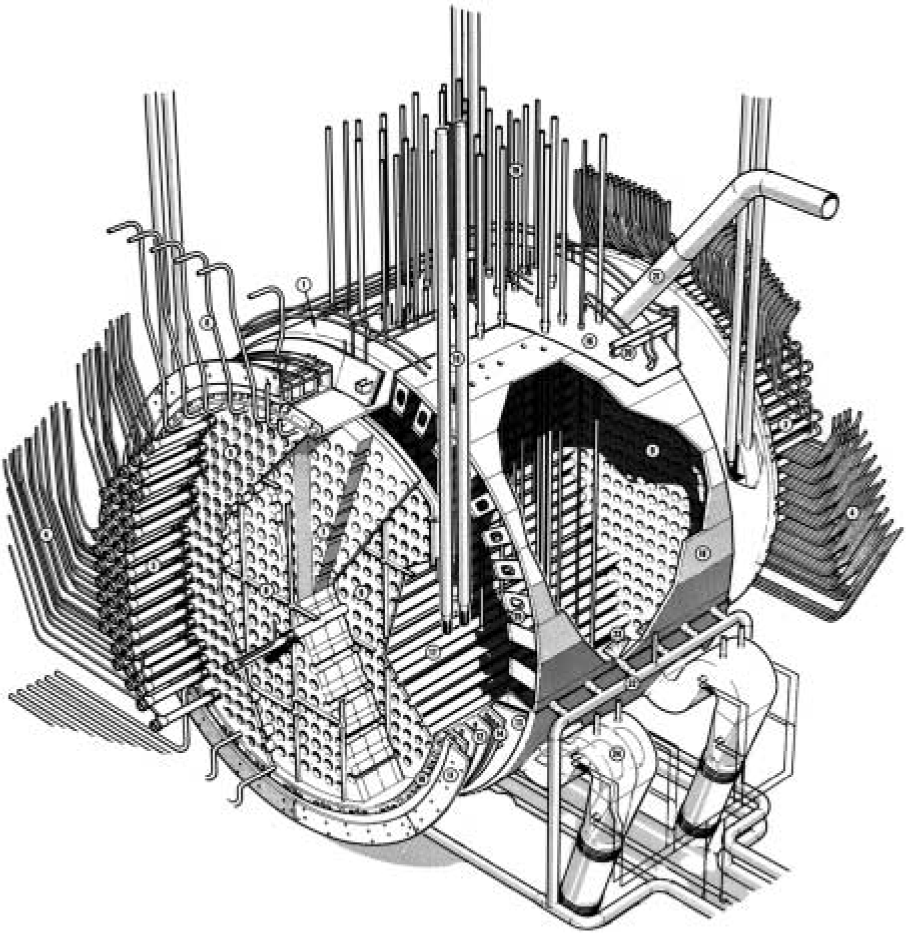


FIG. 129. Schematic of a CANDU reactor assembly.

fuel channels, thereby making the reactor insensitive to pressure assisted rod ejection accidents. The low temperature moderator also acts as an ultimate heat sink.

On-line refuelling provides flexibility in fuel management. It is possible to vary the number and type of bundles added to a channel, the location of the channel to be refuelled, the frequency of refuelling, and even the axial location along the channel where the new fuel bundles are inserted. Both axial and radial power distributions can thus be shaped and controlled, as can the amount of reactivity added to the reactor during refuelling.

The on-power refuelling system can also be used to remove defective fuel, in the unlikely event that a fuel defect develops. There are effective reactor systems to

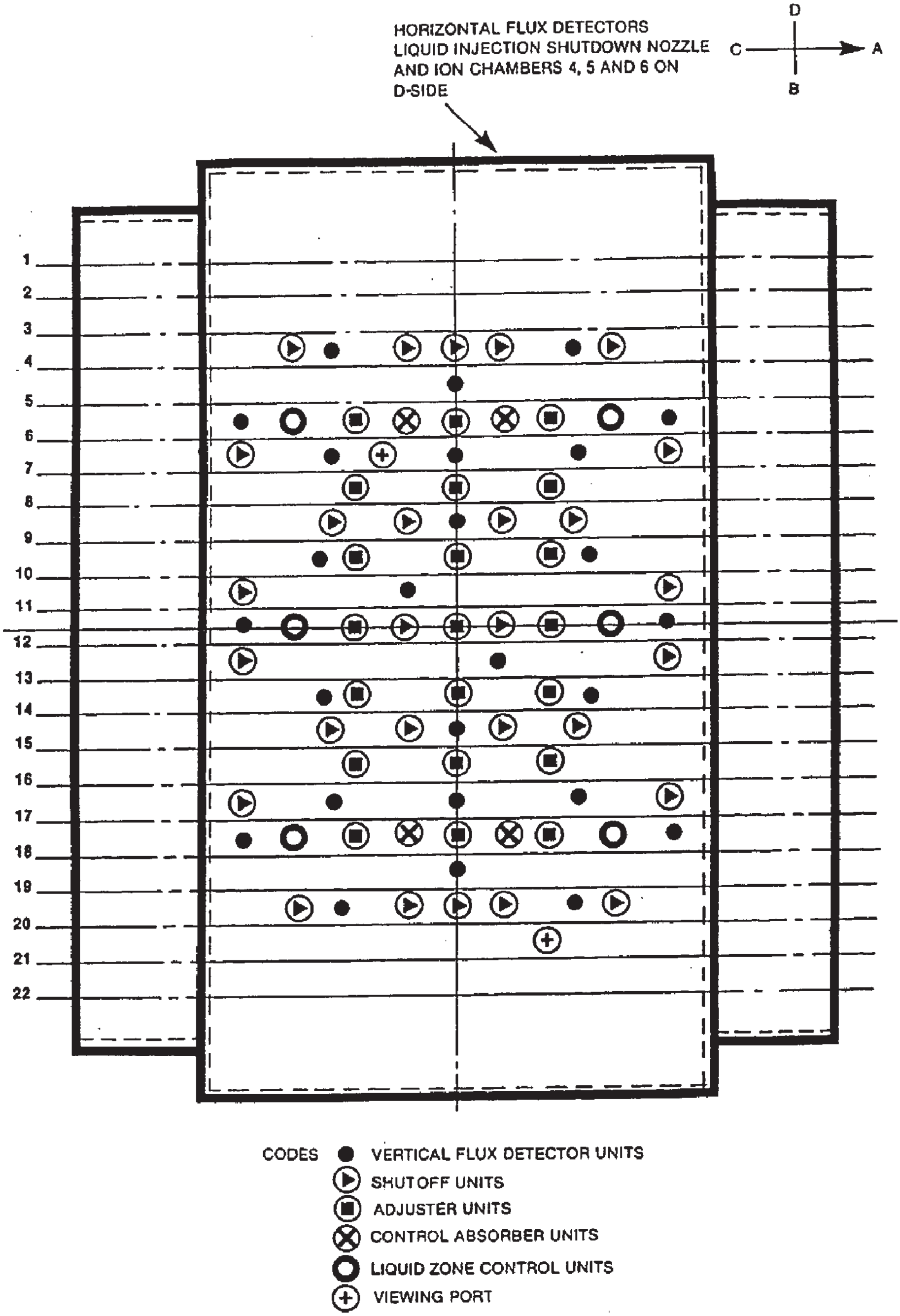


FIG. 130. Top view of a CANDU 6 reactor.

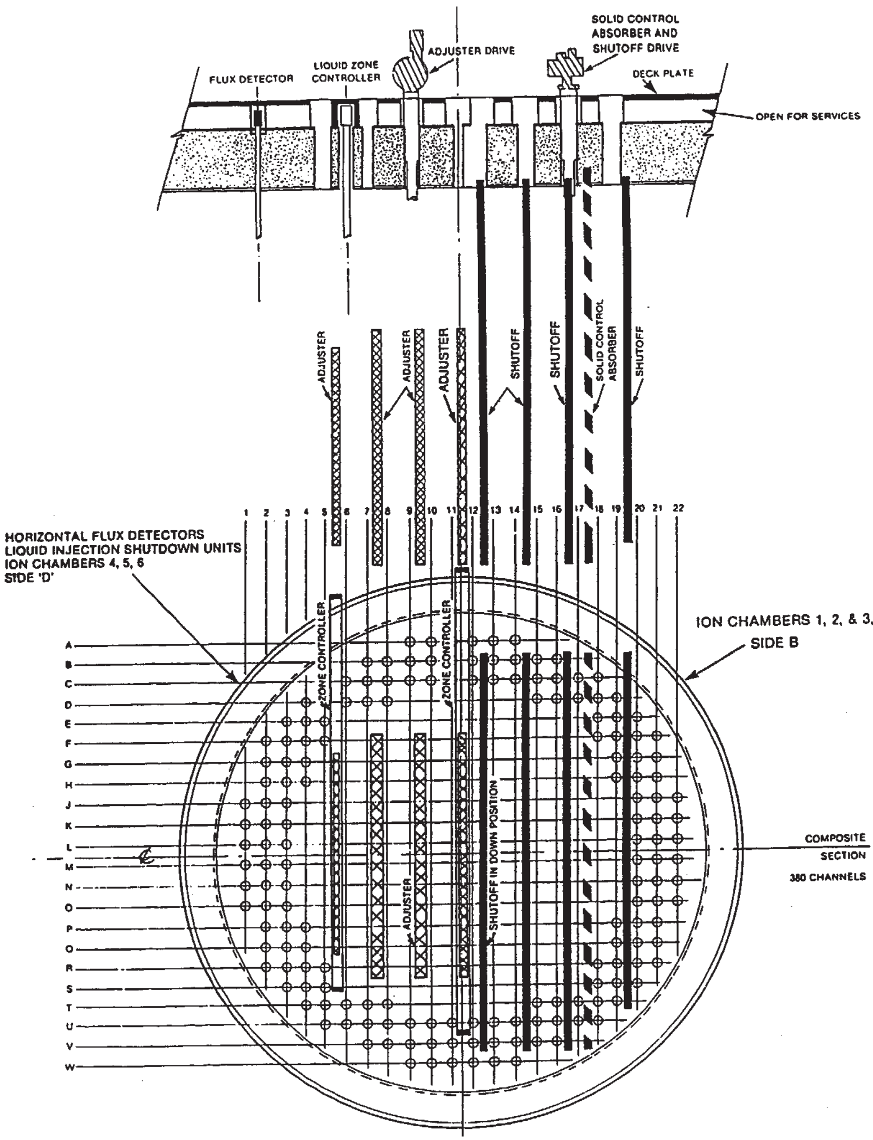


FIG. 131. Front view of a CANDU 6 reactor.

identify and locate defective fuel, so that the defect can be removed immediately, in contrast to other reactor types where a fuel defect must be left in the core for an extended period, or an extensive shutdown undertaken.

The CANDU reactor is provided with two, special, dedicated shutdown systems. These are additional to, and separate from, the shutdown capability of the reactor regulating system. The special shutdown systems are also physically, logically and functionally separate from one another. Each shutdown system is fully capable, acting on its own, of shutting the reactor down in the event of any accident condition.

Each shutdown system is testable on-line and must have an availability of at least 999/1000, i.e. an unavailability factor of less than 0.001, or 10^{-3}. As the shutdown systems are physically and logically separate, the probability of an unterminated power excursion occurring is extremely low (in fact, lower than the probability of an 'anticipated transient without scram' in comparable LWR designs).

Each special shutdown system is actuated by a triplicated protection system. Neutronic actuating signals come from in-core self-powered detectors and out of core ion chambers.

A separate system of in-core, vanadium detectors provides a flux mapping capability to reconstruct the spatial flux distribution in the reactor on an on-going basis. Flux mapping guards against local overpowers, helps calibrate the zone control system detectors, and provides information for optimizing power output and fuel management.

6.1.8.5. HWR reactivity coefficient characteristics

With currently used fuel, the coolant void reactivity (or void coefficient) in HWRs is positive. The basic reason for this lies in the pressure tube design, which separates the coolant from the moderator. As the volume of coolant is much smaller than the volume of moderator, the coolant does not play a very large role in the moderation of neutrons in the CANDU lattice. As a consequence, the loss of coolant is not at all equivalent to a loss of moderator (as in pressure vessel reactors), which would result in a large loss of reactivity.

The main physics effects which contribute to increasing the lattice reactivity following a drop in coolant density are a decrease in the resonance absorption and an increase in fast fission.

It should be noted, however, that a positive void coefficient means that fast, reliable neutronic signals can be (and are) used to identify quickly a large loss of coolant and to actuate one (or both) of the shutdown systems to turn off the fission chain reaction.

The large LOCA presents the greatest challenge to HWR shutdown systems in terms of the rate of positive reactivity insertion. In fact, the shutdown systems' response speed is many times that required for any other accident. The CANDU 6 reactor and shutdown systems design build on experience gained with earlier HWR designs, and include large safety margins against power transients for large break LOCAs. Highly conservative calculations form the design basis for shutdown system response time. In all design basis accidents in CANDU 6 reactors, each shutdown system is tripped by at least two, effective, diverse initiating signals.

AECL has evaluated large LOCA reactor physics for many years, using sophisticated three dimensional coupled neutronic and thermohydraulic computer programs, backed up by reactor physics measurements taken in the ZED-2 reactor at AECL's Chalk River Laboratories. The uncertainty allowances used in accident analysis modelling are updated as improved experimental results on CANDU fuel and improved computer models become available.

The neutronic dynamic response to a change in reactivity is governed by the kinetics parameters. The HWR delayed neutron fraction is increased by the production of delayed photoneutrons created by the break-up of deuterium nuclei (induced by the action of delayed gamma rays) in the heavy water moderator. The very long prompt neutron lifetime, the increased delayed fraction and the long time constants of the delayed photoneutron precursors slow the evolution of power transients, and result in a slower response to a reactivity insertion than in other reactor types.

The relationship between reactivity and reactor period (the time constant for power increase in a transient) is shown as a function of the neutron lifetime in Fig. 132.

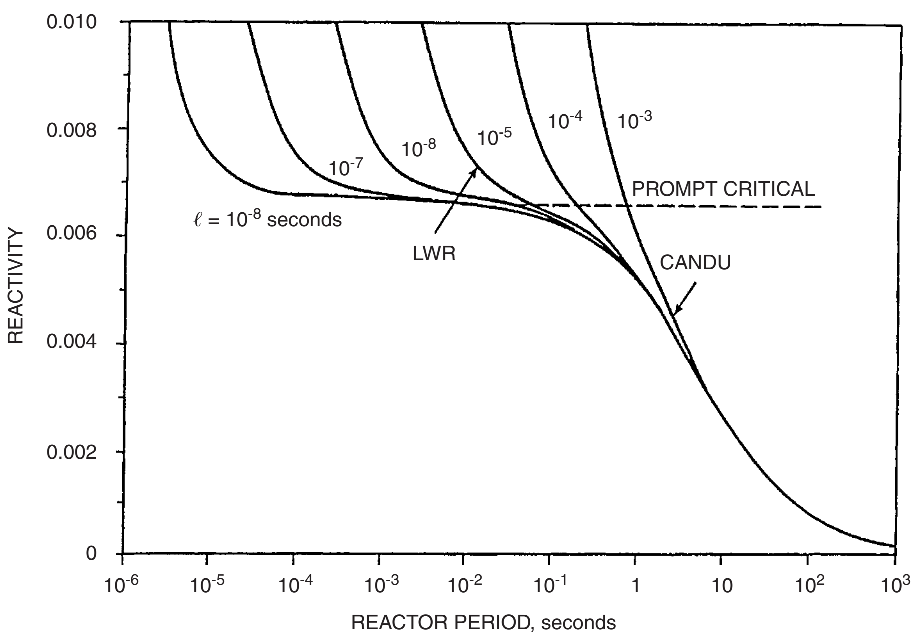


FIG. 132. Relationship between reactor period and reactivity for various neutron lifetimes.

The sensitivity of a near prompt critical power excursion (reactivity insertion is defined in Fig. 133) to the neutron lifetime is shown in Fig. 134. This shows that an HWR lattice provides a much longer response time in which to control or shut down the reactor than would otherwise be the case with a shorter neutron lifetime.

The initial inherent response of a CANDU reactor to a reactivity increase is prompt, negative reactivity feedback. The increase in fuel temperature gives a rapid negative reactivity feedback through the Doppler effect (temperature broadening of absorption resonances). This prompt negative feedback ensures that the reactor is stable during normal operation and allows control by mechanical or hydraulic devices. A reactivity increase can also lead to positive feedback through the coolant void coefficient; the time-scale is longer than the fuel temperature feedback, since it is controlled by heat transfer through the fuel to the coolant. It is less important in periods of normal control but becomes more important in accidents.

The power and coolant temperature reactivity coefficients are small, which greatly simplifies operator control. Further, the overall benign characteristics of CANDU reactivity coefficients eliminate any reactivity concerns specific to reactor cooldown, because coolant density increase from cooldown actually decreases reactivity.

6.1.9. Spent fuel disposal

6.1.9.1. Introduction

The simple fuel design of the HWR offers many benefits in spent fuel handling, storage and disposal. Over twenty years of experience gained in the development and

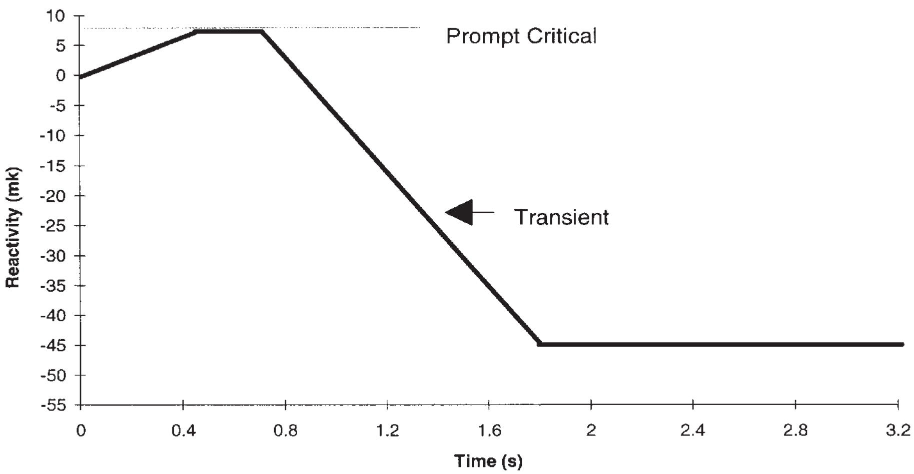


FIG. 133. Hypothetical near prompt critical reactivity excursion.

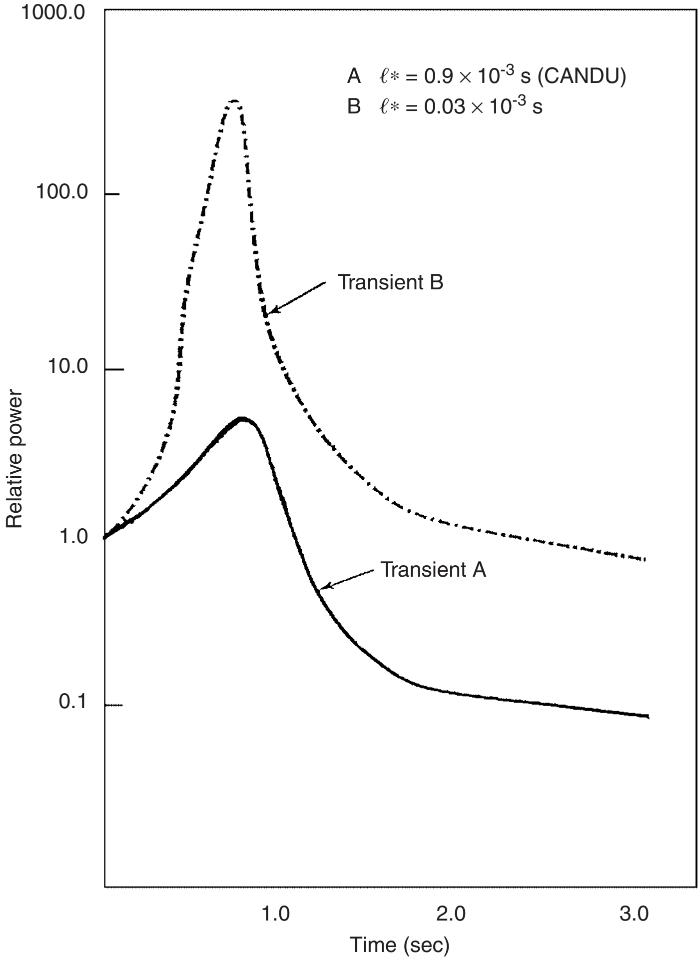


FIG. 134. Sensitivity of power excursion to neutron lifetime.

application of medium term storage have resulted in technology that is well proven and economical, and which has an extremely high degree of public and environmental protection. In fact, both the short term (water pool) and medium term (dry canister) storage of spent HWR fuel are lower in terms of cost per unit of energy than for other reactor types.

AECL's extensive research into permanent spent fuel disposal has resulted in the development of a defined concept for Canadian spent fuel disposal in crystalline rock. This concept was recently confirmed as being "technically acceptable" by an independent environmental review panel [115]. Much of the technology behind the

Canadian concept can be adapted to permanent, land based disposal strategies chosen by other countries. In addition, the Canadian development has established a baseline for HWR spent fuel permanent disposal costs. Canadian and international work has shown that the cost of permanent CANDU fuel disposal is similar to the cost of LWR fuel disposal, per unit of electricity produced.

6.1.9.2. Wet storage of spent HWR fuel

After an HWR natural uranium fuel bundle is discharged from the reactor, having spent about one year of irradiation, it is removed to a pool system for interim storage. The water in the pool removes the residual heat produced by the spent fuel and provides radiation shielding for workers. The compact design of the HWR fuel bundle and the impossibility of criticality occurring for HWR natural uranium and SEU spent fuel bundles under storage conditions in water pools make for extremely simple pool storage [131, 132].

To ensure protection of the environment and public health, the spent fuel pool in CANDU reactors is provided with double concrete walls, designed such that any leakage through the inner wall would enter drains located between the walls and would flow to the cleanup system. The water acts to shield personnel from the radiation emitted by the spent fuel, and the heat generated by the radioactive decay is transferred to the water. The water is cooled by circulating it through heat exchangers and is purified by passing it through filters and ion exchange systems that remove any dissolved and suspended radionuclides.

6.1.9.3. Dry storage of spent HWR fuel

After spent HWR fuel has been out of the reactor for about six years, its activity and rate of heat generation will have decreased sufficiently to allow the fuel to be transferred to dry storage if desired. Compared with wet storage, dry storage is considered to offer several advantages:

- Reduced amounts of radioactive waste, such as filters;
- Smaller potential for contamination of the storage facility;
- Little or no corrosion of fuel sheaths;
- Less radiation exposure to operating personnel (with triple containment of the spent fuel radioactivity);
- Minimal maintenance;
- Low operating costs.

Dry storage is simple to implement and modules can be added as needed. AECL instigated the study of dry storage for spent nuclear fuel in the early 1970s. Structures similar to silos (termed concrete canisters) were first developed for the storage of research reactor enriched uranium fuel and then perfected for the storage of spent CANDU natural uranium fuel. By 1987, concrete canisters were being used for the safe and economical storage of all spent fuel accumulated during the operation of AECL's decommissioned prototype reactors. Each canister contains a stack of spent fuel baskets.

The same basic technology was then applied to on-site dry storage of spent fuel generated by operating CANDU power generating plants. The spent fuel baskets have air as the fuel cover gas. The basket is dried and welded in a shielded work station and transferred by a shielded transfer flask into the dry storage canister. The concrete canisters hold 9 baskets of 60 bundles (except at one station where one basket holds 36 bundles), making a total of 540 bundles. The concrete canisters are licensed for standard CANDU 6 bundles which have had a minimum of seven years' cooling, corresponding to a heat release of approximately 5.4 W per bundle.

Both New Brunswick Power and Korea Electric Power Company selected AECL's concrete canister technology for their CANDU 6 nuclear generating stations at Point Lepreau (1989) and Wolsong 1 (1990).

As regards the Embalse plant in Argentina, a dry storage facility was designed locally and erected by Argentinian companies headed by CNEA, following the general concepts of AECL's dry concrete canister storage. Up to the end of 1998, the total number of irradiated fuel bundles in Embalse's dry storage facility was 31 860. In the case of the Atucha 1 reactor, the irradiated spent fuel assemblies are stored in two fuel pools, again quite similar to those used for PWRs. It is expected that with a suitable rearrangement of the already existing stored fuel assemblies, no additional fuel storage capacity will be needed up to the end of the plant's operational lifetime.

In 1989, AECL began development (in co-operation with Transnuclear Inc.) of a monolithic, air-cooled, concrete structure for dry storage termed MACSTOR. MACSTOR modules require less land area than do concrete canisters for the same amount of spent fuel and are suitable for the storage of spent fuel assemblies from other reactor types (PWR, BWR, WWER) as well as CANDU. The MACSTOR modules store 12 000 bundles in 20 storage cylinders, each holding 10 baskets of 60 bundles. The MACSTOR module is also licensed for a seven year cooling period (although it is expected that the MACSTOR module will soon be licensed for storage of CANDU 6 natural uranium bundles requiring six years' cooling, corresponding to a marginally higher heat release of 6.08 W per bundle at a reference burnup of 7.8 MW·d/kg). In 1995, Hydro Québec built the first such system for dry storage at the Gentilly 2 generating station [133].

In India, after irradiation, the HWR fuel bundles are stored in the spent fuel storage bay. These bays are designed to store the discharged fuel for 10 years. The water in the pool acts as a cooling medium and also provides the required shielding. The fuel bundles from the Rajasthan 1 reactor have been under such storage for more than 25 years and show no signs of degradation. The storage of fuel bundles in concrete casks in a dry air atmosphere is being practised at Rajasthan Atomic Power Station as a means of enhancing the spent fuel storage capacity. Each such concrete cask has the capacity to store 187 fuel bundles. The atmosphere in the concrete cask is regularly monitored to ascertain the integrity of the fuel.

Dry storage costs for spent fuel from a CANDU reactor design have been compared with those of a typical LWR, both normalized to a gross power of 1000 MW(e) [134]. The results indicated that the dry storage costs for the CANDU system are about 30% lower than those for the LWR system.

Table XX lists some data on the CANDU spent fuel currently in dry storage. Figure 135 shows a basket in which 60 spent CANDU natural uranium fuel bundles would be stored (either in a concrete canister or in MACSTOR). Figure 136 is a cross-section of a MACSTOR facility while Fig. 137 shows the MACSTOR facility at the Gentilly 2 reactor.

6.1.9.4. Disposal of spent HWR fuel

Spent fuel from HWRs is currently stored by its owners in water filled pools or in dry storage concrete containers. Current storage practices at HWR sites have an excellent safety record, permit easy monitoring and retrieval, and could be continued for many years.

Storage, while an extremely effective interim measure, is not considered to provide a permanent solution. The objective of permanent disposal is to manage spent nuclear fuel in a way that does not depend on the use of institutional controls to maintain safety in the long term. This does not mean that society would not use long

TABLE XX. CANDU SPENT FUEL CURRENTLY IN DRY STORAGE

Reactor unit	Fuel quantity (Mg U)	Number of canisters/ modules
Whiteshell Research	17	11
Gentilly 1	67	11
Douglas Point	298	47
NPD	75	11
Point Lepreau	2790 (lifetime)	275 (lifetime)
Wolsong 1	2790 (lifetime)	275 (lifetime)
Gentilly 2	2790 (lifetime)	13 (lifetime)

Note: In the case of Point Lepreau and Wolsong 1 reactors, 275 canisters (silos) are required to accommodate the fuel discharged over the lifetime of the reactors. For Gentilly 2, 13 MACSTOR modules are required.

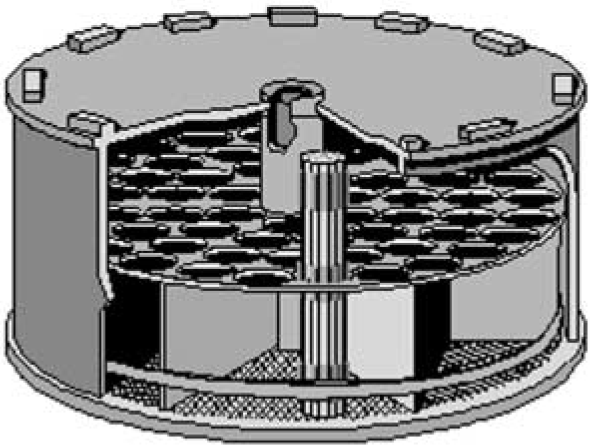


FIG. 135. Basket holding 60 CANDU spent fuel bundles.

term institutional controls, but rather that, even if such controls should fail, human health and the natural environment would still be protected.

Thus, Canada and other countries with mature nuclear power programmes have, for many years, been developing the technology for the permanent disposal of spent nuclear fuel and reprocessing waste. There is an international consensus among waste management experts that the preferred method for the long term management of spent nuclear fuel and reprocessing waste is land based geological disposal [135].

During the past two decades, extensive R&D has taken place throughout the world on land based geological disposal. Land based geological disposal would involve placing containers of spent fuel or reprocessing waste in sediment or rock at a depth of hundreds of metres, with access gained from the land surface. The advantages of land based geological disposal are that most nations have, within their borders, rock types potentially suitable for use in disposal, and that land based disposal concepts can be based on existing mining and engineering technology. Research has concentrated on disposal media (rock types) having one or more of the characteristics commonly considered favourable for disposal. The decision to focus on a particular rock type or types is made in each country on the basis of the geological conditions within that country and a variety of other relevant factors. International research on land based geological disposal of radioactive waste has concentrated on five disposal media: crystalline rock, salt, clay (or shale), tuff and basalt.

It is not currently economic to recycle the fissile material remaining in spent HWR fuel, and Canada currently has no plans for doing so. Along with other countries, Canada is basing its plans for the disposal of spent nuclear fuel on deep

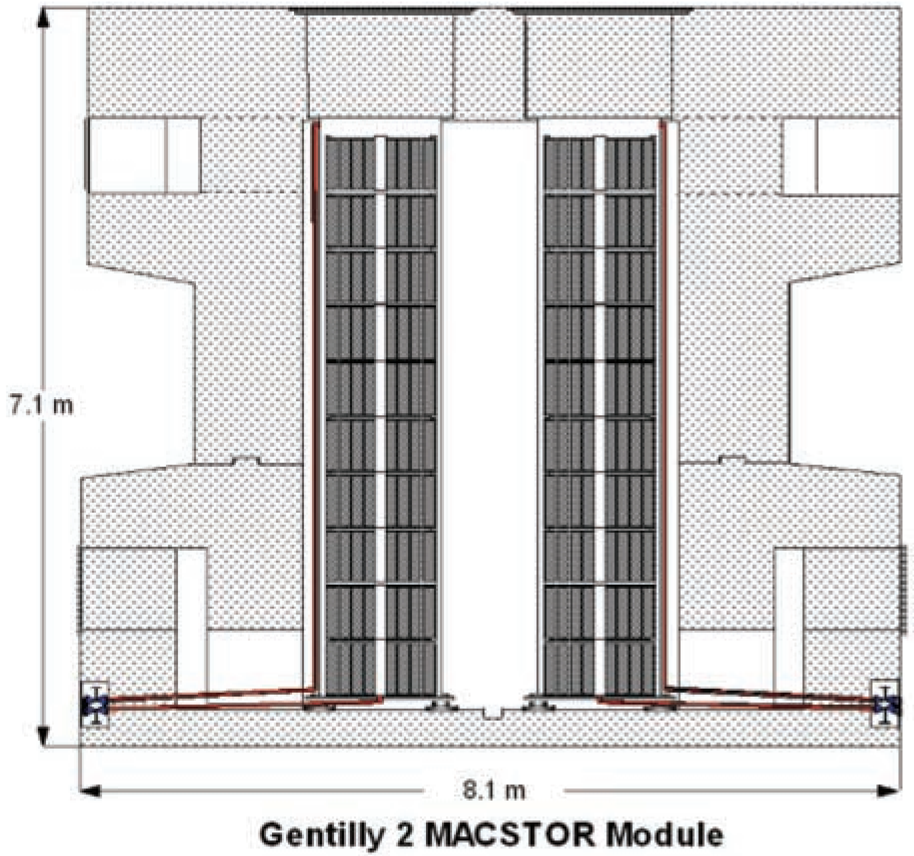


FIG. 136. The MACSTOR facility.

geological disposal; in the Canadian case, in stable crystalline rock of the Canadian Shield. In common with the approach adopted in other countries, the disposal concept developed by AECL entails isolating the spent nuclear fuel from the biosphere by a series of engineered and natural barriers [136]. These barriers include: the spent fuel itself (in particular, the ceramic UO_2 matrix); sealed, long lived containers in which the spent fuel is placed; buffer materials to separate the containers from the surrounding rock and to control the movement of water to, and corrosion products away from, the container; the use of seals and backfill materials to close the various openings, tunnels, shafts and boreholes; and the rock mass in which the repository is located (the geosphere). The biosphere, although not a barrier, is an important part of the overall system. As it contains the pathways through which direct exposure of humans and other organisms to contaminants could occur, it must be studied as part of any spent fuel management programme.

FIG. 137. The MACSTOR facility at the Gentilly 2 reactor.

In the Canadian concept, as in most countries, spent nuclear fuel (or reprocessing waste) would be emplaced in a repository excavated below the water table in stable rock. Hence, the principal concern from the point of view of long term safety is that groundwater could eventually become contaminated with radioactive or other hazardous materials and ultimately make its way to the surface and pose a risk to future human health or to the environment. The multibarrier system developed for the Canadian concept will prevent this by the combined effects of radioactive decay and the containment, retardation, dispersion and dilution that will take place as the contaminants move through the disposal system (from the waste form, through the container, buffer, backfill and geosphere to the surface). Thus, human health and the

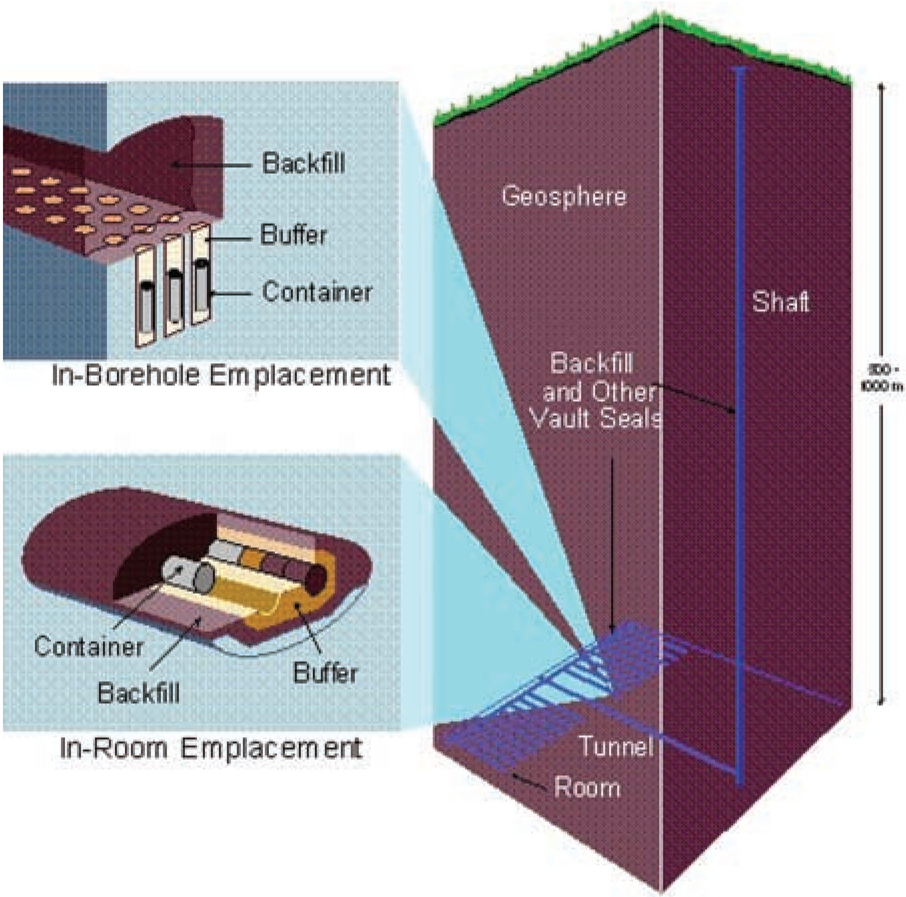


FIG. 138. Geological disposal showing two emplacement options.

environment will be protected. Figure 138 is a schematic diagram of the Canadian concept for a spent fuel repository showing two emplacement options: borehole and in-room emplacement.

Considerable efforts have been made internationally to evaluate the behaviour of deep geological repositories with time and their long term safety. There is an international consensus among waste management experts that “appropriate use of safety assessment methods, coupled with sufficient information from proposed disposal sites, can provide the technical basis to decide whether specific disposal systems would offer to society a satisfactory level of safety for both current and future generations” [137].

Several countries, including Canada, have carried out quantitative assessments of the risk associated with disposal [136, 138–144]. These analyses indicate that the amount of contaminants moving from a repository to the surface would be very small and that the radiological impact would be many orders of magnitude less than that from naturally occurring radioactivity in the surface environment.

Although Canadian research has focused on disposal in crystalline rock, much of the Canadian research on waste forms, spent fuel containers, buffer materials and repository sealing systems would be relevant to disposal in other rock types. In addition, most of the methods developed or evaluated by AECL for the characterization of sites and for the assessment of the performance of disposal systems would also be applicable to other rock types and surface environments. The Canadian experience also provides confidence in estimates of the direct costs of permanent disposal of CANDU fuel.

6.1.9.5. Technical and economic issues associated with the direct disposal of spent CANDU fuel

A key consideration in assessing the long term performance of a disposal system is the stability of the radioactive material, since it represents the source term for safety and performance assessments. Spent UO_2 fuel exhibits excellent behaviour under the reducing groundwater conditions expected at depth in saturated rock. The long term stability of spent fuel has been assessed in studies of natural analogues such as the Cigar Lake uranium ore deposit [145]. This ore deposit formed 1.3 billion years ago and has been in contact with groundwater since its formation. Despite this, the uranium has remained stable and very little dissolution has occurred under the reducing groundwater conditions prevailing in the deposit. Solidified waste from reprocessing is also an excellent waste form, and therefore both spent CANDU fuel and the solidified waste from reprocessing the spent fuel can be disposed of safely.

As noted, although the reprocessing of spent fuel to extract useful material for recycling is possible, it is not currently undertaken in Canada and there are no plans to do so. If spent fuel were reprocessed, almost all the radioactive material that remained (the high level waste) would be solidified. The quantities of the radionuclides present in the spent fuel are not changed by reprocessing (other than by the removal of the uranium and plutonium), nor are the total activity or heat generated by these radionuclides. Thus, reprocessing in itself would not reduce the amount of fission products, per unit of electricity generated, to be disposed of. For the first few hundred years, when the activity is dominated by fission products, the total activity per unit of electricity generated would be roughly the same for vitrified high level waste as for the spent fuel from which it was derived. Similarly, for the first few hundred years, the rate of heat production would be nearly the same for both types of waste. To meet the thermal constraints for underground disposal, the size of repository required for the disposal of the vitrified high level waste alone would be about the same as that for the spent fuel from which it was derived. Therefore, both options — direct disposal of fuel bundles, or disposal of reprocessed, vitrified waste — are equally economic.

As an illustration, Canadian studies of the direct disposal in granite of spent natural uranium CANDU fuel indicate a repository requirement of about 400–700 m^2 per TW·h of electricity [146, 147]. Swiss studies on the disposal of vitrified reprocessing waste, also in granite, indicate a comparable or somewhat larger repository requirement of about 600–1200 m^2/TW·h. Finnish and Swedish studies of the disposal of used BWR and PWR fuels indicate a similar requirement of about 500–1000 m^2/TW·h [116, 148].

This result comes about because the quantity of heat generating material per unit volume of a repository is limited by the maximum acceptable heat load of the container, the repository sealing systems such as clay based buffer materials, and the surrounding host geological formation. In the case of spent fuel that has been out of the reactor for a given time, the amount of heat that the fuel generates depends on the fuel burnup, which normally corresponds to the amount of electricity that was generated by the fuel (DUPIC spent fuel being a notable exception (see Section 6.5.2.5)). Thus, the size of a repository is, to a first approximation, dependent only on the amount of electricity that was generated in the production of heat generating spent fuel (or reprocessing waste (see Table XXI)), and is not a strong function of the volume of the heat generating waste.

Reprocessing operations also produce streams of low and intermediate level wastes that contain long lived radionuclides. Efforts are under way to reduce the volumes of these wastes [149]. However, they represent a waste stream that will need to be disposed of, and many countries are basing their plans on the use of deep geological disposal to isolate these wastes from the biosphere. Direct disposal of CANDU fuel bundles eliminates this need.

The overall cost of deep geological disposal depends not only on the size of the repository, but also on the costs associated with site characterization, construction of shielded spent fuel handling facilities, support of R&D, safety assessments, etc., all of which represent more or less fixed costs. These costs do not depend so much on spent fuel volumes (or reprocessing waste forms) as on factors such as geological setting, the scale of the nuclear programme and details of the system design (such as design of the disposal container).

These factors, together with the fact that the size of a repository is more a function of the radioactivity and heat produced by the spent fuel or reprocessing waste rather than its volume mean that the costs of disposal (not including the cost of reprocessing) per unit of electricity produced are comparable for the direct disposal of spent fuel and for the disposal of the long lived waste arising from reprocessing.

TABLE XXI. COMPARISON OF SPACE REQUIREMENTS FOR THE DISPOSAL OF HIGH LEVEL WASTE

Country	Waste form	Burnup of fuel (MW·d/kg HE)	Storage period before disposal (a)	Waste emplacement method	Plan area of repository (m^2/TW·h)*
Canada	Used natural U CANDU fuel	7.9	10	In boreholes From rooms	400
		8.3	10	In-room	660
Finland	Used BWR fuel	35	20–40	In boreholes From rooms	500–900
Sweden	Used BWR and PWR fuels	35 (BWR) 39 (PWR)	~40	In boreholes From rooms	500
Switzerland	Vitrified waste from reprocessing			In-room	600–1200

* This figure assumes an ideal geometry. The actual plan area will be larger because rooms will be laid out to avoid important geological features such as fracture zones and faults.

The estimated costs of disposing of spent CANDU and LWR fuels, and reprocessing wastes (not including the costs of reprocessing) have been compared [147]. The estimated costs of direct disposal of spent CANDU fuel, for a given electricity output, are comparable or less than those incurred in the direct disposal of spent LWR fuel and reprocessing wastes. Typical results are shown in Fig. 139.

Differences in estimated costs are more related to factors other than the waste form, i.e. such factors as the host geology, whether or not an overpack is used, and the size of the national nuclear programme.

6.1.9.6. Indian perspective of spent fuel management

The Indian nuclear power programme incorporates spent fuel reprocessing and recycling in thermal and fast reactors as integral parts of the fuel cycle. The spent fuel obtained from HWRs, after sufficient cooling, is transported to reprocessing plants. Two reprocessing plants, the Power Reactor Fuel Reprocessing Plant at Tarapur and the Kalpakkam Reprocessing Plant at Kalpakkam, are currently reprocessing power reactor spent fuel.

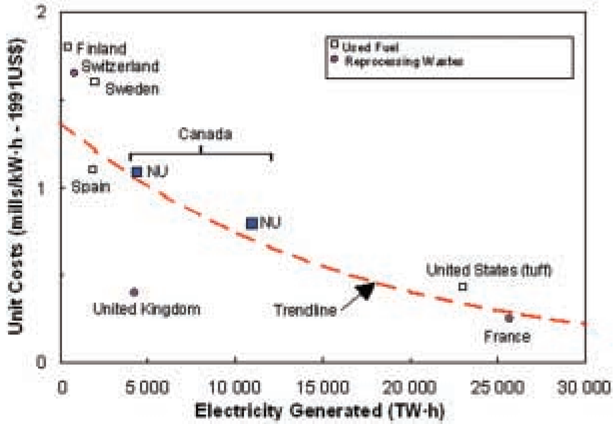


FIG. 139. Costs of packaging and disposing of nuclear fuel as a function of generated electricity.

6.1.9.7. Summary

The HWR design is backed by well established, economic means of short and medium term storage, and by more than twenty years experience gained in defining and proving concepts for permanent disposal. In the case of HWRs, wet pool storage has the advantages that fuel criticality concerns are eliminated (for natural and slight enrichments), and that pool fuel storage baskets can be moved directly to dry storage facilities. On the basis of current studies, HWR dry spent fuel costs are expected to be lower than those for PWRs, partly as a consequence of the more easily handled HWR fuel bundles. As regards permanent disposal, the Canadian programme to develop a technically acceptable disposal concept provides an international demonstration of feasibility. It also establishes cost benchmarks, showing that permanent disposal costs for CANDU spent fuel would be comparable or less than the costs for PWRs.

Costs do not include the costs of site screening, site evaluations and supporting R&D. Data are taken from Ref. [116], supplemented with additional results from Canada for disposal of used SEU CANDU fuel [150]. Except for France and the USA, the costs relate to disposal in crystalline rock. No host rock is specified for France because a variety of geological media have been considered in the estimates.

6.1.10. Non-proliferation and safeguards considerations

The natural uranium fuel cycle currently used in HWRs supports proliferation resistant technology, and HWRs such as CANDU reactors have been safeguarded to

IAEA requirements. The use of natural uranium fuel eliminates the need for enrichment technology, allowing a State to operate a nuclear power programme without requiring proliferation sensitive technology, such as enrichment or reprocessing facilities. Natural uranium is accepted as having a lower proliferation risk than HEU or plutonium because it cannot be used directly for the manufacture of nuclear weapons. It must either be enriched, or used to generate other, direct use, material. The technologies for enrichment, or for generation and extraction of direct use material from natural uranium, are not required to support a natural uranium fuel cycle.

6.1.10.1. HWR features relevant to non-proliferation and safeguards

HWRs contain unique features that affect non-proliferation regimes and the application of safeguards. In the following discussion of non-proliferation and the general principles of safeguarding HWRs, the CANDU reactor will be used as the primary reference.

(a) Small bundles

HWRs typically utilize natural uranium bundles that are small compared with LWR fuel assemblies. Studies such as Ref. [151] conclude that in the presence of an appropriate safeguards regime, the small bundle configuration does not affect proliferation. Undeclared proliferation requires the diversion of a large number of highly radioactive bundles, and the same chemical processing as that used for irradiated LWR fuel.

(b) On-line refuelling

Unlike LWRs that are refuelled annually, most HWRs are refuelled daily while the reactor is operating at full power. A typical CANDU reactor is refuelled with approximately 40 fresh natural uranium fuel bundles per week. Extremely reliable and secure monitoring systems have operated successfully for more than 15 years to verify these daily fuel movements and confirm that HWRs are operated in compliance with non-proliferation treaties. Monitoring equipment includes core discharge monitors to count irradiated bundles removed from the core, spent fuel bundle counters to count bundles as they are transferred to the spent fuel bay, and surveillance cameras to monitor for unusual or undeclared fuelling operations.

(c) Spent fuel storage

Many HWRs make use of dry storage (see Section 6.1.9.3). This involves making transfers of irradiated fuel from the spent fuel pond that requires continuous safeguards monitoring to dry storage canisters that can be sealed and safeguarded in a more efficient manner. Typically, CANDU reactors transfer fuel to dry storage after spending 6–10 years in storage in the spent fuel bay. Filled canisters are sealed by IAEA inspectors and checked regularly.

6.1.10.2. Current safeguards

The IAEA verifies that all CANDU reactors worldwide are only being used for the peaceful civilian purposes for which they were intended, through a combination of strictly adhered to safeguards measures including inspections by IAEA officials and installation of equipment.

According to Article 28 of INFCIRC/153, the objective of these measures is "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection."

The basis of the IAEA's verification system for both LWRs and HWRs is identical, and consists of three major components:

- Accountancy, which comprises reporting by States on the whereabouts of the fissionable material under their control, on stocks of fuel and spent fuel, etc. Each CANDU fuel bundle has a unique identifier that enables it to be tracked through the entire fuel cycle;
- Containment and surveillance techniques, such as seals which provide confirmation that no material has disappeared, and film and TV cameras which record any action occurring in a particular area of a nuclear installation;
- Inspection by IAEA inspectors, checking instruments and seals installed, verifying accounting records, and confirming physical inventories of fuel and spent fuel.

6.1.10.3. Safeguards approach

The manner in which the IAEA designs safeguards activities is referred to as the safeguards approach and is tailored to the design of the facility. The safeguards approach for CANDU reactors was established by the IAEA in the 1980s after detailed review of the CANDU design and has been successfully applied at CANDU stations worldwide. It is based on an IAEA analysis of all technically possible diversion paths at a facility and on criteria specified in the particular safeguards agreement. The approach is also designed to counter the possible undeclared production of direct use material. The safeguards approach for other

HWR designs is generally a combination of the safeguards approaches used for CANDU and LWR designs.

(a) Inspection goals

In order to implement the safeguards approach, performance targets or inspection goals are specified in the IAEA safeguards criteria. The inspection goal for a facility consists of a quantity component and a timeliness component (see Tables XXII and XXIII). The quantity component relates to the scope of the inspection activities necessary to provide assurance that there has been no diversion of a significant quantity of nuclear material over the material balance period. The timeliness component relates to the frequency of inspection activities necessary to provide assurance that no abrupt diversion has taken place. The inspection goal for each facility is regarded as having been attained if all the criteria relevant to the material types and categories present at the facility have been satisfied.

(b) CANDU safeguards approach

The challenge in safeguarding a CANDU reactor lies in the tracking of a very large number of items (fuel bundles) that are, in effect, continuously flowing through the reactor core (i.e. on-power refuelling). A CANDU core cannot be ‘sealed’ in the way that is commonly done to safeguard an LWR reactor pressure vessel.

The safeguards approach for CANDU uses the same two basic tools to achieve inspection goals as those used for an LWR:

- *Item accountancy:* This includes item (fuel bundle) counting and identification, non-destructive measurement and examination to verify the continued integrity of the item.
- *Containment and surveillance measures:* These are used to complement the accountancy verification methods for safeguarding the spent fuel. Since fuelling occurs daily, permanently installed instrumentation is used continuously to monitor and track fuel movements. A surveillance system in the spent fuel bay allows the IAEA to detect the undeclared removal of spent fuel from storage.

A comprehensive set of containment and surveillance measures was installed during the construction of all CANDU reactors commissioned since the late 1980s (i.e. Darlington, Cernavoda, and Wolsong 2, 3 and 4). Earlier CANDU reactors (i.e. Point Lepreau, Gentilly 2 and Embalse) were retrofitted with containment and surveillance measures during the 1980s.

TABLE XXII. SIGNIFICANT QUANTITIES AND TIMELINESS GOALS RELEVANT TO NATURAL URANIUM (CANDU) AND LEU (PWR) FUEL CYCLES

Category	Type	Significant quantities (kg)	Timeliness goals (months)
Direct use material	Pu in spent fuel	8 (Pu)	3
Indirect use material	LEU and natural U*	75 (U-235)	12

* Less than 20% ^{235}U (includes natural and depleted uranium).

The following list summarizes the activities performed at CANDU reactors to achieve IAEA inspection goals (see also Table XXIV and Figs 140 and 141):

- Audit of accounting records and comparison with reports to the IAEA.
- Examination of operating records and reconciliation with accounting records.
- Verification of new fuel on an annual basis. In order to detect possible diversion of new fuel, the verification is carried out by item counting, serial number identification and non-destructive assay.
- Verification of fuel in the core by the continuous monitoring of spent fuel discharges using a core discharge monitor or optical surveillance.
- Counting and monitoring of spent fuel bundles as they are transferred from the vault to the fuel bay using a spent fuel bundle counter. Where required, other vault penetrations are monitored to verify that all spent fuel is transferred to the bay via the route containing the bundle counter.

TABLE XXIII. NUMBER OF FUEL ITEMS[a] REQUIRED TO OBTAIN ONE SIGNIFICANT QUANTITY

Fuel type	Reactor type		
	LWR		CANDU
	Demountable PWR pins[b]	Assemblies (PWR, BWR)[c]	Bundles
Fresh fuel	1125–1445	5–15	550
Irradiated fuel	250–578	2–6	100

[a] CANDU fuel is natural uranium; LWR fuel is low enriched ^{235}U (3–5%).
[b] Range is from PWR assemblies containing 225–289 pins.
[c] Range is from PWR to BWR.

TABLE XXIV. CANDU SAFEGUARDS EQUIPMENT

Equipment	Location	Description
Core discharge monitor	Reactor vault	A combination of neutron and gamma radiation detectors in the reactor vault is used to count irradiated fuel discharges from both reactor faces.
Spent fuel bundle counter	Irradiated fuel discharge path from vault to bay	A set of gamma detectors is used to count irradiated fuel bundles as they are transferred through the irradiated fuel discharge port in the vault to the spent fuel bay.
Multiplex closed circuit television surveillance system	Spent fuel bay and some vault penetrations	Up to 16 video cameras monitor for undeclared fuel movements. All CANDUs have cameras in the spent fuel bays. Cameras may also be located in the vault to monitor fuelling machine movements and outside the airlock to monitor for the undeclared removal of irradiated fuel in flasks.
AECL random coil sealing system	Spent fuel bays	Irradiated fuel is stored in tamper indicating enclosures with a lid fastened using IAEA approved AECL random coil seals to ensure that the bundles are not removed.
Yes/no radiation monitors	New fuel port, auxiliary port, two pipes in spent fuel bay	Thermoluminescent devices or other electronic gamma radiation detectors are used to detect the discharge of irradiated fuel through vault penetrations other than the irradiated fuel discharge port, specifically, the fresh fuel port and the auxiliary port. These are also used at some stations to detect the removal of fuel from the spent fuel bay through pipes.
Spent fuel verifier	Spent fuel bays (only where AECL random coil sealing is not used)	A collimated gamma spectrometer is lowered into the spent fuel bay to verify the authenticity of spent fuel during IAEA inspections. This instrument is used at some stations that do not use the AECL random coil sealing system.
VACOSS seals	Dry storage canister	A fibre optic bundle is threaded through the structure to be sealed and formed into a loop using a closure that crimps the cable. The light pattern created by the crimping forms the identity and integrity element for the seal.
E type seal	Dry storage canister	A wire is threaded through the structure to be sealed and formed into a loop using a mechanical closure that cannot be opened without incurring permanent recognizable damage. A random mechanical marking inside the seal and the one time closure mechanism form the integrity elements.

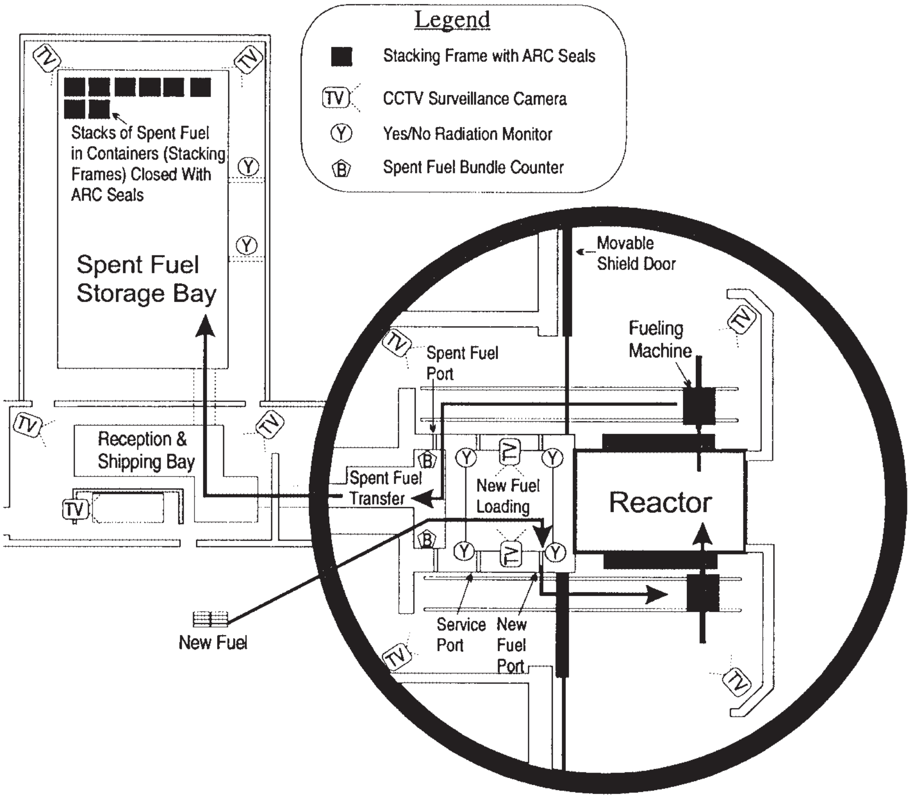


FIG. 140. Location of safeguards equipment in a typical CANDU 6 reactor.

- Surveillance in the spent fuel bay. In addition, at some stations, containers of fuel are closed with tamper indicating AECL random coil seals which complement the surveillance system. At other stations, non-destructive assay instruments are used as a backup to the surveillance system. Darlington, Gentilly 2, Cernavoda, and Wolsong 2, 3 and 4 use, or are expected to use, the tamper indicating containers with AECL random coil seals. Embalse, Wolsong 1, Point Lepreau, Pickering and Bruce use non-destructive assay techniques in conjunction with surveillance. In view of the effort required to re-verify spent fuel prior to sealing, it is generally not cost effective to implement AECL random coil sealing once a station has a significant inventory of spent fuel.
- Sealing in dry storage canisters. Dry storage canisters containing irradiated fuel in baskets with welded lids are sealed using two diverse seals. Typically, one is based on a fibre optic cable and the other is mechanical, based on wire. The

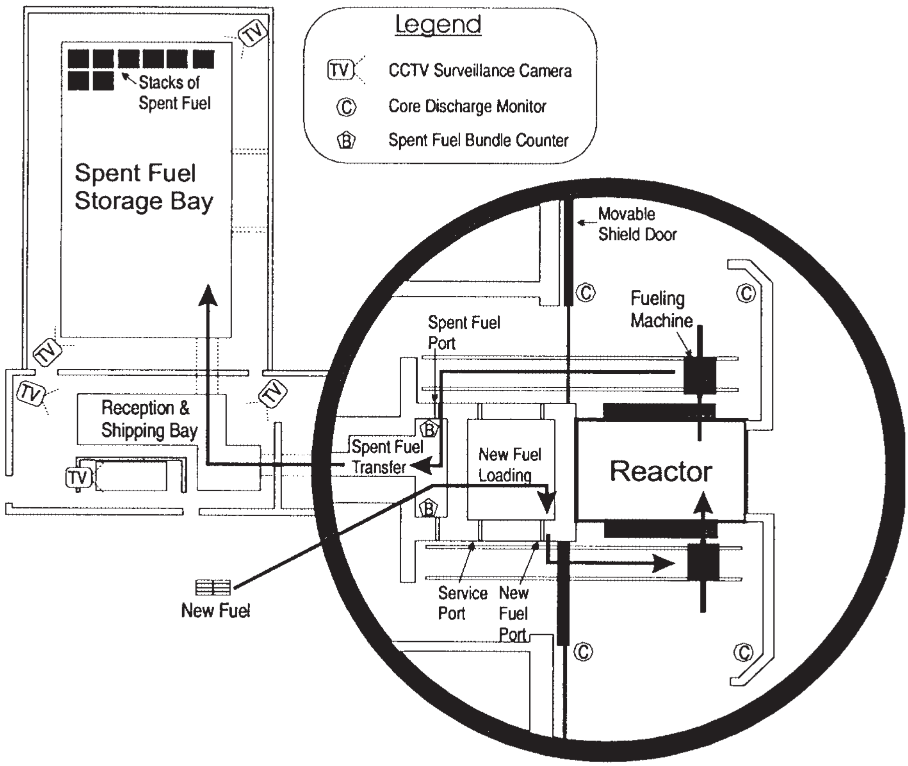


FIG. 141. Location of safeguards equipment in a typical CANDU 9 reactor.

seals are routed through the canister structure so as to detect both removal of the top cover and access through the concrete walls. The canisters are verified quarterly by checking one of the seals. In addition, these canisters contain verification tubes buried in the concrete walls. Detectors are lowered through these tubes to provide a gamma profile of the canister. Current profiles can then be compared with reference gamma profiles to verify the contents of the canister without having to open it.

6.1.10.4. Proliferation resistant HWR fuel cycles

Owing to their neutron economy, HWRs provide a unique opportunity for the implementation of proliferation resistant advanced fuel cycles. An HWR can support fuel cycles based on PWR/HWR synergism or on the exploitation of unique materials such as thorium, using proliferation resistant recycling technology.

In HWR/PWR fuel cycles, spent fuel from PWRs is used to fuel an HWR reactor. As a result of the neutron economy of the HWR, it is not necessary to add fissile material to the spent PWR material. Moreover, it is not necessary to separate the fissile material from the fission products. This enhances the proliferation resistance of the overall fuel cycle. This technical report discusses several advanced fuel cycles that could be developed which have a high degree of proliferation resistance.

6.1.10.5. Advanced/improved HWR safeguard approaches

New approaches to safeguards arising from the Protocol approved at the IAEA General Council in 1997 are expected to change significantly the manner in which safeguards are applied worldwide. The Protocol provides for expanded declarations and greater transparency; hence it provides greater assurance that States are not engaged in undeclared activities. This redirects the safeguards' focus at nuclear power plants from being one of verification of detailed accounting for spent fuel to one of verification that States are not engaged in any undeclared use of spent fuel. New safeguards approaches being explored include satellite surveillance, short notice random inspections and environmental monitoring. All of these are primarily aimed at detecting undeclared nuclear activities within a State. In the long term, these new approaches are expected to transfer safeguards' effort from nuclear power plants to entire States and hence reduce the costs associated with safeguarding nuclear power plants.

Individual States that are voluntarily party to the Non-Proliferation Treaty must each accede to the Protocol. Changes to safeguards approaches arising from the Protocol are expected to be phased in gradually over a period of many years in order to maintain the level of confidence provided by the current safeguards regime. The traditional safeguards' focus on spent fuel accountancy at nuclear power plants is expected to remain important for the next decade as new approaches based on the Protocol are phased in and tested.

Traditional safeguards approaches are also being investigated as the means of reducing safeguards' cost and increasing their effectiveness at HWRs. One of these is fuel bundle amalgamation whereby a quantity of spent fuel bundles is placed in a sealed container and thereafter treated as a single item. This change in the configuration of the nuclear material is similar to that where pellets are combined into elements and elements into bundles, but which occurs later in the fuel cycle. Amalgamation reduces the cost of safeguards by reducing the number of items to account for and to verify. Advanced safeguards technology includes increased use of remote monitoring and remote surveillance, use of monitoring systems that combine video surveillance and radiation detection, and use of seals that can be opened and closed by the nuclear power plant staff without an IAEA inspector having to be present. Advances such as these will be integrated into future HWR safeguards approaches.

6.2. HWR FUEL CYCLE FLEXIBILITY

6.2.1. Features leading to fuel cycle flexibility

An inherent feature of the HWR is its very high degree of fuel cycle flexibility. Several key features contribute to the fuel cycle flexibility of the HWR: high neutron economy, channel design, on-power refuelling and bundle design.

High neutron economy is a feature of the HWR, and in the CANDU reactor results from:

- The use of heavy water as both coolant and moderator;
- The use of low neutron absorbing structural materials (such as zirconium alloys);
- A simple fuel bundle design that requires a minimum of structural material;
- On-power refuelling;
- The use of a low pressure moderator in which the reactivity devices are located between the fuel channels, thereby reducing the mechanical requirements of these control devices.

High neutron economy is reflected directly in excellent fuel utilization in the HWR, which, with natural uranium fuel, is about 39% better than that of LWRs, and with enrichment is even higher (Table XVII).

High neutron economy is required for the use of natural uranium fuel. It also allows the use of other low fissile content fuels and makes possible a unique synergism with LWRs, one that offers the potential for fuel recycling with a high degree of proliferation resistance, using simpler and potentially cheaper technologies than conventional reprocessing. High neutron economy also means that about double the thermal energy can be derived from burning fissile material in a CANDU reactor as opposed to a PWR, regardless of whether the fuel is enriched uranium, MOX, or recycled uranium. High neutron economy also results in high conversion ratios that can approach unity in the self-sufficient equilibrium thorium cycle where as much fissile material is produced as is consumed. The high conversion ratio will also result in a natural synergism with future FBRs.

The basic reactor design consists of fuel channels that are separated by relatively large amounts of heavy water. The moderating properties of heavy water result in a long neutron migration length and the use of a relatively large lattice pitch. This, in turn, allows implementation of the pressure tube design and provides the capability for on-power refuelling. The neutrons entering a channel are very well thermalized and the spectrum is largely insensitive to the fuel type.

Reactivity control mechanisms are located in the low pressure, low temperature moderator. They are not part of the fuel bundle. This precludes various fuel performance issues from arising, ones that must be addressed with LWR fuel, such as:

- Deposition of dissolved boron in the coolant onto the fuel (the so-called ‘axial offset’ phenomena),
- Severe distortion of control rod guide tubes (resulting from having to push the burnup to very high levels in order to reduce fuel cycle costs),
- Hydriding of control rod guide tubes (resulting from large temperature gradients in the fuel assembly between the fuel and the guide tubes),
- Corrosion of the fuel sheath (resulting from pushing fuel burnup to very high values).

On-power refuelling contributes both to the excellent neutron economy of the HWR and to its fuel cycle flexibility. The HWR maintains very low excess core reactivity, and core reactivity characteristics change very little throughout the fuel cycle. On-power refuelling enables reactor criticality to be maintained indefinitely by replacing fuel on a daily or quasi-daily basis. This avoids the need to use large amounts of soluble poison in the moderator or of burnable poisons in the fuel. Typically, the excess reactivity hold down in a CANDU is about 20 mk, compared with over 200 mk at the start of a fuel cycle in a PWR. Only small amounts of excess reactivity are needed in the light water zone control compartments for bulk reactivity and spatial control, and the desired level of xenon override capability can be designed in the adjuster rods. The small excess reactivity is also a safety advantage, as the amount available to be added to the core in the event of accidents is limited.

On-power refuelling provides a great deal of flexibility in fuel management. Fuelling is bi-directional, meaning that adjacent fuel channels are refuelled in opposite directions. This method of fuelling results in both a flattening of the axial flux distribution and a symmetrical axial flux distribution. The axial power distribution along the channel is mainly determined by the variation of reactivity along the channel, which is itself determined by the fuel type (particularly the initial enrichment), the fuel management scheme and the location of reactivity devices in the moderator (e.g. the adjuster rods). The variation of reactivity along the channel can be controlled in the simplest instance by varying the rate of refuelling; in most cases this provides sufficient shaping of the axial power distribution and results in similar axial power profiles for a wide variety of fuel types. The consequence of this is that SEU, MOX, thorium and even inert matrix fuels (containing no fertile material) can all be utilized in existing CANDU reactors.

Moreover, the axial power distribution with enriched fuels peaks towards that end of the channel at which new fuel is added, and decreases along the length of the channel. In the case of CANDU 6 and CANDU 9 reactors, in which fuelling is in the direction of coolant flow, the peak bundle power occurs towards the coolant inlet end of the channel. This axial power distribution results in higher thermohydraulic margins than those obtained with the more symmetrical axial power distribution arising from the use of natural uranium fuel, and the declining power history with burnup facilitates good fuel performance.

Ultimately, bundles can be removed from the channel during refuelling, reshuffled and then reinserted in any order. This axial shuffling provides nearly unlimited capability for shaping the axial power distribution, if necessary. Adjuster rods are located between fuel channels, in the low pressure moderator. They flatten the power distribution using natural uranium fuel, a function not required with enriched fuel, and provide xenon override capability. With the use of enriched fuel, the adjuster rods can be easily replaced, if desired, or even eliminated, providing further flexibility in the accommodation of advanced fuel cycles.

The fuel management scheme can also shape the radial channel power distribution across the core. With enrichment, the extra burnup potential can be traded off for increased power in the outer channels by flattening the channel power distribution, and obtaining more power from a given sized core without increasing maximum bundle and channel powers. Alternatively, the flatter channel and bundle power distributions available with enrichment can be used to reduce the peak fuel ratings for a given reactor power, thereby increasing operating and safety margins. Fuel management flexibility also provides many options in the transition from one fuel type to another.

Most CANDU reactors have failed fuel detection systems; on-power refuelling enables prompt removal of any failed fuel. This reduces the risk to a utility of introducing a new fuel type. An extensive array of in-core flux detectors has always been a feature of CANDU reactors, and this ensures that the flux and power distributions are well known, regardless of the fuel type or fuel management strategy employed.

Finally, the basic CANDU fuel bundle design lends itself to fuel cycle flexibility. The fuel composition can be easily varied from ring to ring. Again, with the channel design and the separation of channels from each other using large volumes of heavy water, there is a 'sameness' in the neutron spectrum entering the fuel channel, regardless of the details of the fuel design. Hence, new fuels can be accommodated within operating reactors without changes needing to be made to the fuel bundle geometry.

6.2.2. Fuel cycle drivers

The fuel cycle path chosen by a particular country or utility will depend on many local and global factors, or criteria, key amongst them being:

- Overall energy economics,
- Resource considerations,
- Environmental impact,
- National and international policies and goals.

Public opinion and acceptance will also continue to influence fuel cycle decisions. Safety is not listed as a criterion for selecting fuel cycle options because any fuel cycle strategy must meet the highest safety standards. However, it is acknowledged that some advanced fuels or fuel cycles might be chosen specifically to address safety or licensing issues. For example, low void reactivity fuel (discussed in Section 6.5.2) can reduce void reactivity; the CANFLEX fuel bundle features higher critical channel power, increasing the margins to fuel dry out; and enrichment and new bundle designs can be used to reduce peak linear element ratings, increasing operating and safety margins.

Thus, there are many factors that influence the choice of fuel cycle, and criteria with which to optimize the cycle. These factors are discussed in the literature [152, 153].

These criteria are not always self-consistent (the use of SEU in HWRs is an example of a fuel cycle that simultaneously optimizes several criteria; the more usual case being a trade off between conflicting criteria). The importance assigned to these criteria will vary from country to country, and over time. In particular, the relative weight assigned to each of the criteria will differ between the developing and the industrialized countries, and it is in the developing countries that the greatest increase in energy demand, and electricity in particular, will take place. Within a country, local electrical utilities will be largely driven by the short term, economic imperative to produce electricity at lowest cost, particularly in the face of deregulation and privatization of the electricity supply industry. National organizations may have broader perspectives and agendas encompassing international commitments to reduce greenhouse gases, and issues of national security and diversity of energy supply.

There is, thus, no single fuel cycle strategy that is optimal for all countries at all times. Given the historical difficulty involved in predicting the availability and cost of energy resources and fuel cycle technologies, and the large uncertainties and variability associated with many of these criteria, fuel cycle flexibility must be a cornerstone on which to base any fuel cycle strategy. This flexibility will enable a country, or utility, to optimize its fuel cycle strategy according to its own unique circumstances, and to respond to the ever changing external and internal environments. The HWR has the inherent ability to utilize a wide range of fuel cycles. This section describes, in more detail, some of those drivers and the optimization criteria that will influence the choice of fuel cycle.

6.2.2.1. Economics

Minimization of the overall electricity generation cost will be the principal driver in at least the short and medium terms. The traditional cost advantage of nuclear power has been eroded, and competitive pressures will continue to be exerted by independent power producers, and by deregulation and privatization of the electricity supply sector. In particular, natural gas turbines have low capital costs, which can be recovered within a few years, regardless of the fuel cost. To compete, the nuclear power industry will have to reduce costs, and the capital cost of new plants, in particular, will have to be reduced.

Operating plants will be forced to 'seize' every opportunity to reduce operation, maintenance and administration costs. One of only few opportunities for cost reduction open to operating utilities is the reduction of fuel cycle costs. This will favour the use of SEU (or recycled uranium) within the current decade.

However, there is an even greater opportunity for the use of advanced fuels for maintaining or increasing operating margins, particularly in the face of ageing phenomena. Advanced fuel designs (such as CANFLEX) that provide enhanced thermohydraulic performance, can also provide a large economic benefit to the utility by maintaining thermohydraulic operating margins. Use of the CANFLEX fuel bundle as the vehicle for introducing SEU enhances the economic benefits of each. This will be the most important driver for the introduction of advanced fuels in the next five years.

In new plants, there is very strong pressure to reduce capital costs and this will favour the use of SEU. Section 6.5.2 details how SEU can be used to reduce capital costs in new plants.

6.2.2.2. Resource considerations

Historically, concern over the availability and price of natural uranium has been the main fuel cycle driver. The short and long term availability (both globally and nationally), cost, security and diversity of energy resources are still important considerations in many countries. The commercial reprocessing industry was established on the premise that FBRs would be required to meet the demand for nuclear fissile resources, as uranium resources became exhausted. Plutonium, which was once viewed as a valuable energy resource, is now increasingly portrayed as a threat, as is its increasing accumulation, whether separated through reprocessing or in spent fuel. Fast reactors are nowadays discussed more often in terms of their role in 'managing' separated plutonium by destroying it, rather than in producing it.

The current perspective is very different from that of two decades ago. This change, of course, is a result of many things, including a relative abundance of fossil fuel resources, a slowdown in the growth of nuclear power generation, and the low cost of uranium, due in part to discoveries of new high grade uranium deposits. This perspective might change equally dramatically over the next two decades.

A recent publication assessed uranium supply and demand for several nuclear growth scenarios up to 2050 [154]. Figure 142 shows three nuclear growth scenarios developed in Ref. [154] to illustrate the wide range of plausible future options, ranging from a high growth scenario reflecting high economic growth and significant

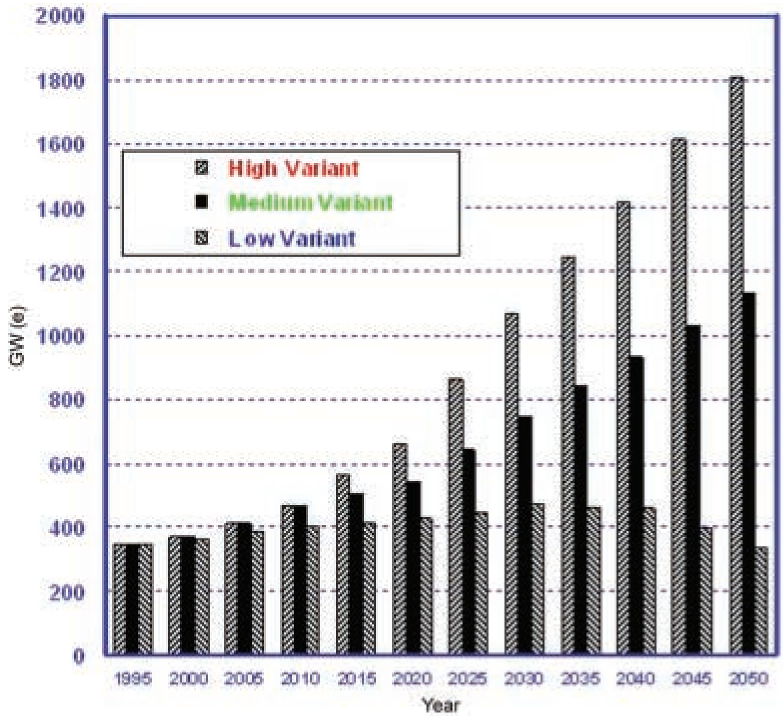


FIG. 142. World nuclear capacity (GW(e)).

development of nuclear power, to a medium variant corresponding to moderate economic growth and sustained development of nuclear power, to a low variant corresponding to a phase out of nuclear power by 2100. The background assumptions and calculational methodologies are given in Ref. [154].

Figures 143 and 144 illustrate the resultant uranium requirements for the high and medium growth scenarios, respectively, for a range of fuel cycle options: OT/HT corresponds to a once through strategy with high uranium tails enrichment (0.3%), and S2/LT corresponds to an aggressive recycle strategy in which all the spent fuel from LWRs would be reprocessed and fissile materials recycled in LWRs using MOX fuel for 30% of the core. This scenario assumes the commissioning of FBRs at a significant rate by 2030.

Also shown in Figs 143 and 144 are the known uranium reserves recoverable at less than US $80/kg U (2.12 million t, not including existing civilian and military inventories); the estimated known uranium resources recoverable at less than US $130/kg U (4.5 million t); and the total conventional uranium resource recoverable, including speculative resources, at less than US $130/kg U (15.5 million t) .

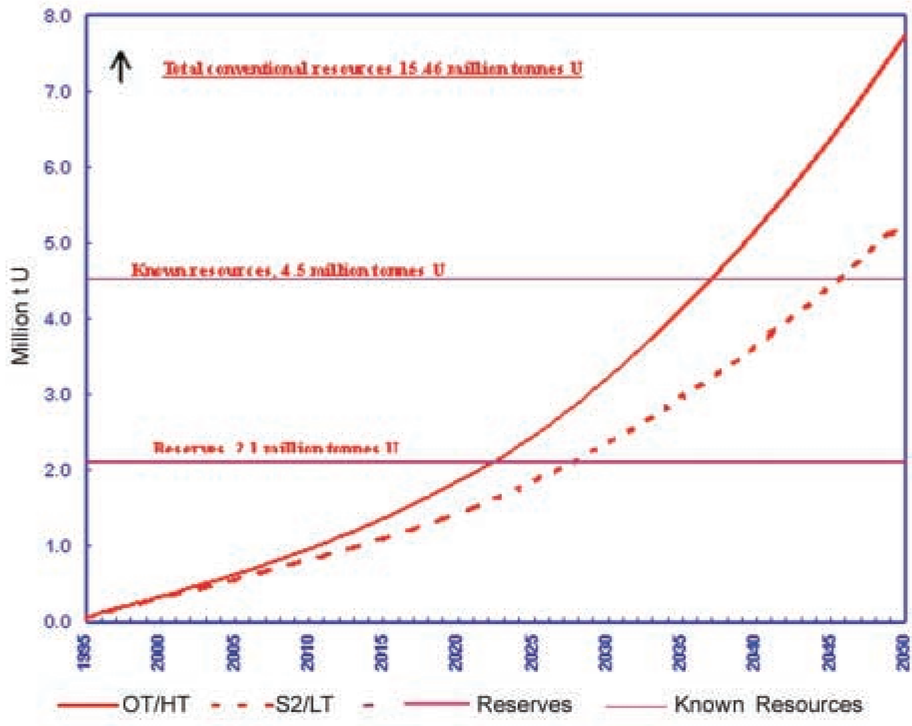


FIG. 143. Cumulative uranium requirements: High variant (million t U).

According to the high growth scenario, the known conventional uranium resources recoverable at less than US $130/kg would be exhausted sometime between 2035 and 2045, depending on the fuel cycle strategy employed. Even under the medium growth scenario, known conventional uranium resources recoverable at less than US $130/kg would be exhausted by 2045 in the once through, high tails fuel cycle. From Ref. [154]:

"Additional conventional resources could be at least partly discovered and economically exploited within the time frame considered … taking into account the lead times for exploration and implementation of mining and milling facilities .... Taking into account all the conventional and unconventional resources, the uranium resource base is large enough to support nuclear power development to 2050 and beyond in the three nuclear variants for all strategies ... With the progressive exhaustion of known resources recoverable at low cost, uranium prices are likely to increase and fuel cycle strategies aiming at reducing uranium requirements will become increasingly attractive from sustainability as well as competitiveness viewpoints."

In summary, uranium resources for thermal reactors are expected to be adequate to 2050 and beyond, but the cost of production may increase significantly as the

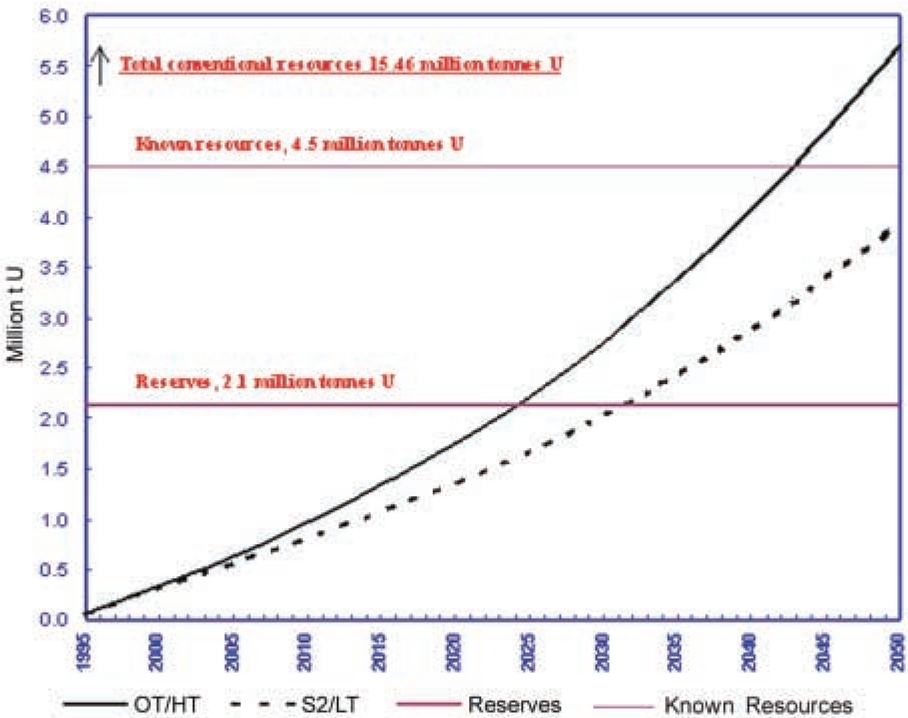


FIG. 144. Cumulative uranium requirements: Medium variant (million t U).

lowest cost resources are depleted. Hence, availability of natural uranium will not, in general, be a strong fuel cycle driver in the short term. Issues such as future security, self-sufficiency and independence of energy resources are perceived as being less critical when there is a recognized abundance and diversity of resources (particularly if there is an abundance of domestic resources).

Nonetheless, in some countries, security and independence of fuel supply, and issues concerning the use of domestic resources and the localization of the nuclear fuel supply industry are still important considerations. In these countries, and for those governments taking a more strategic, longer term view of energy supply, these issues may be factors to take into account when considering uranium efficient fuel cycles, fuel recycle options, or thorium fuel cycles. In this context, various CANDU/PWR synergistic fuel cycles, such as DUPIC, may assume greater importance. At the very least, such governments might invest on an ongoing basis in fuel cycle research, so that the options are available when needed in the future.

In Canada, the localization issue gives rise to a different consideration — the impact on the Canadian uranium mining industry of the domestic use of more uranium efficient fuel cycles, such as SEU.

In the broader sense, the availability of enriched uranium can be included as a resource consideration. The use of natural uranium, and independence from foreign suppliers of enrichment, have been factors in the adoption of HWR technology. However, enrichment is now widely available from several suppliers. Recycled uranium from reprocessing spent LWR fuel is also becoming increasingly available from several suppliers. This reduces the reliance on any one country or technology for enrichment, and will generally favour the use of SEU in the HWR.

Another important consideration, given how quickly the ‘environment’ in which decisions are made can change, concerns the imprudence of taking decisions that would preclude future energy options. This impacts on decisions taken with regard to the use of plutonium.

6.2.2.3. Environmental impact

Environmental considerations must encompass the entire fuel cycle, from mining and refining, to fuel fabrication, use in-reactor (fuel failures increase the potential of radioactive exposure to operating staff and to the public), interim storage, recycling and final disposal. Improving uranium resource efficiency reduces the amount of mined uranium required to support a given level of power generation (and the amount of mining tails produced), reduces the amount of depleted uranium tails from the enrichment plant (if the fuel is enriched), and reduces the quantity of spent fuel produced. Therefore, there is a cascading environmental benefit accruing from improved resource efficiency.

Back end considerations include the total radioactive inventory in the spent fuel or high level waste requiring permanent disposal, and the various impacts of the permanent disposal facility, i.e. environmental, availability, cost and public acceptance. Much misinformation pervades the back end of the fuel cycle; the most common misconception being that owing to its lower burnup and higher quantity of spent fuel, the natural uranium fuelled HWR is at a cost or environmental disadvantage relative to other reactor types. Actually, the radiological burden is mainly a function of the energy produced and not the quantity of spent fuel or radioactive waste generated. The repository size (and cost) is also largely determined by the heat generated by the spent fuel or radioactive waste, which in turn is mainly determined by the energy generated by the fuel rather than by the quantity of spent fuel or its burnup.

Nonetheless, if reducing the volume of spent fuel helps win public and stakeholder acceptance of nuclear power, then that can be considered as being a fuel cycle driver, given that there are economic and other advantages to using SEU in the HWR.

Another misleading view relates to the use of ‘radiotoxicity’ as a measure of merit in assessing the hazard from the spent fuel or reprocessing waste. This measure ignores the natural and engineered barriers in a geological repository that keep the waste isolated from humans and the biosphere. The more appropriate criterion would be the dose incurred by the public from the spent fuel in the repository, rather than the source term. The use of radiotoxicity as the measure of merit can lead to incorrect conclusions being drawn about the relative merits of alternative fuel cycles. It is, therefore, very important that environmental decisions be based on sound fact and not on misinformation.

There do not appear to be direct environmental considerations that would favour reprocessing over direct disposal (other than the general benefits of recycling).

Technical advances can lead to improvements in 'environmental friendliness', as well as in other aspects of the fuel cycle. A good example is advanced enrichment technology, such as atomic vapour laser isotope separation, which could lead to lower enrichment costs (further favouring the use of SEU in the HWR), and to lower enrichment tails becoming economic. A decrease in the level of enrichment tails would reduce the amount of mined uranium required to produce a certain amount of enriched fuel, as well as reducing the amount of depleted uranium waste produced.

6.2.2.4. National and international policies and goals

Non-proliferation is becoming an increasingly important international fuel cycle consideration. Reprocessing is particularly sensitive, since it produces separated plutonium. This is in spite of the fact that the international safeguards regime has been successful in preventing the spread of nuclear weapons, and in ensuring that plutonium derived from the reprocessing of civilian spent nuclear fuel is not diverted for military purposes. The HWR is uniquely suited to exploit the recycle technologies that have been designed to have a high degree of proliferation resistance. The DUPIC cycle is one such example (Section 6.5.2). A high degree of proliferation resistance does not obviate the need for a strong safeguards system; in fact, in some cases it can complicate aspects of safeguards. However, it does make it inherently more difficult to divert fissile material.

The disposition of plutonium retrieved from the dismantling of surplus warheads is another example of a fuel cycle development that is driven by international strategic and political considerations. Hence, the consideration given to MOX fuel in CANDU for the dispositioning of this plutonium, or to incineration in inert matrix carriers (Sections 6.7, 6.8).

Another consideration is the state of industrial development. A country without a strong technical infrastructure may choose natural uranium as a simple, 'entry level' fuel cycle. On the other hand, the development of advanced nuclear fuel cycle technology could be a national priority to further high technology development in its own right, and for the spin-off benefits it may provide. One example of this is in the development of safeguards technology for the DUPIC cycle. The same technology could be applied to other fuel cycles for commercial gain; remote handling technology could be applied to other industries.

In summary, the fuel cycle path chosen by a particular country will depend on a range of local and global factors. No single fuel cycle strategy will be appropriate for all countries. The HWR provides the flexibility to enable any country to optimize its fuel cycle strategy to meet its current and future needs.

6.3. ADVANCED HWR FUEL DESIGNS

6.3.1. Introduction

Running in parallel with the development of specific fuel cycles and advanced HWR fuel designs is the development of generic advancements in fuel design and performance that can be applied to any of these advanced fuels, including SEU, MOX or thorium, or natural uranium. These include the following:

- Generic high burnup element design (some of the features of which might be applied to natural uranium fuel to enhance the load following capability);
- Advanced CANLUB coating that provides protection against stress corrosion cracking at high burnup;
- Enhanced thermohydraulic performance (lower pressure drop, higher critical channel power);
- Tailored reactivity coefficients (low void reactivity fuel is one example);
- Low temperature fuels (such as graphite disc fuel, annular pellets).

Any particular fuel cycle might employ some, or all, of these features, as needed. In addition, advanced characterization techniques will help elucidate the relationship between fuel properties and fuel performance for advanced fuels. Examples of such innovative techniques include the measurement of thermal diffusivity, porosity, density and the oxygen potential of irradiated fuel, and the use of advanced techniques for the measurement of the diffusion coefficient of fission gases and methods for the accurate determination of the plutonium distribution in MOX fuel [155].

6.3.2. Experiences of Canada and the Republic of Korea: CANFLEX

6.3.2.1. Introduction

Canadian CANDU fuel has evolved from 7 element fuel bundles in the NPD reactor, through 19 elements in the Douglas Point reactor, 28 elements in the Pickering plant, to the current 37 element bundle in CANDU 6 and in the Bruce and Darlington plants. The 43 element CANFLEX bundle is a logical step in this evolution. AECL has been developing CANFLEX since 1986 [156, 157]. Since 1991, AECL and KAERI have pursued a collaborative programme to develop, verify and prove the CANFLEX design. In September 1998, New Brunswick Power began a demonstration irradiation of 24 CANFLEX fuel bundles at the Point Lepreau generating station (PLGS) which took place over a two-year period and which will act as a final verification of the CANFLEX design prior to full-core conversion.

6.3.2.2. CANFLEX features

The principal features of CANFLEX are enhanced thermohydraulic performance and lower peak linear element ratings, which provide CANDU plant operators with greater operating flexibility through improved operating margins [158–160]. Critical heat flux enhancement appendages on the CANFLEX bundle enable a higher bundle power to be achieved before critical heat flux occurs, leading to a net gain in the critical channel power of more than 6% over the existing 37 element fuel. The 20% lower maximum linear element rating in a CANFLEX bundle, compared with a 37 element bundle, reduces the fuel operating temperatures, which leads to lower fission gas release and a reduction in the consequences of most design basis accidents. The lower element rating is achieved by adding extra elements and by using larger diameter elements in the two centre rings and smaller diameter elements in the two outer rings [161]. In addition to providing greater operating margins, the lower linear element ratings with CANFLEX facilitate the achievement of higher burnups associated with the use of SEU, recycled uranium and other advanced fuel cycles [162, 163].

CANFLEX has been designed to have hydraulic and neutronic characteristics that are similar to those of the existing fuel. This feature enables the operators to introduce CANFLEX bundles during normal on-power refuelling. No hardware changes are required when switching to using CANFLEX fuel because the bundle is fully compatible with existing fuel handling equipment. Fuel channels containing both CANFLEX and 37 element fuels, in any combination that can occur with normal fuelling, have improved or unchanged operating margins. The transition to using CANFLEX fuel can be undertaken gradually and without wasting existing fuel.

6.3.2.3. CANFLEX verification programme

The CANFLEX bundle has undergone an extensive verification programme, defined in the design verification plan [164]. All testing and analysis were performed under the quality standard CAN/CSA-N286.2 or its equivalent [165]. The verification work consisted of analysis and testing and drew on the capabilities of AECL's facilities in Canada and KAERI's facilities in the Republic of Korea.

The design verification plan identifies the performance requirements, specifies the test or analysis required to verify that the requirements are met, and identifies responsibility and procedures. It requires preparation of a test specification, a test procedure and acceptance criteria, and identifies the required documentation. The following sections summarize the main verification activities.

6.3.2.4. Critical heat flux dry out power testing

To quantify the improved thermohydraulic performance of the CANFLEX bundle for licensing in a power reactor, a series of critical heat flux and pressure drop measurements were undertaken at Stern Laboratories (Hamilton, Ontario), in water at CANDU relevant pressure, flow and temperature conditions [166, 167].

An electrically heated assembly was used, simulating a string of twelve aligned CANFLEX bundles. As one of the applications of CANFLEX is to alleviate the erosion of operating margins resulting from reactor ageing, the string was tested in three separate flow tubes that simulated: an uncrept pressure tube; and two pressure tubes having axially varying inside diameters, one having a maximum diametral creep of 3.3% and the other a creep of 5.1%. The heated part of the string was nominally 6 m long and was equipped with spacer planes, bearing pads, button planes and simulated end plates to mimic the geometry of a string of aligned CANFLEX bundles. The axial heat flux profile was a cosine, skewed towards the outlet end of the fuel string and the radial profile simulated that of natural uranium fuel. The five downstream bundles were equipped with movable internal thermocouples to measure the surface temperature and to detect the first occurrence of dry out, axially and circumferentially, along the elements. The measurements were undertaken using a similar methodology and test matrix to those which had been used previously for conducting measurements on the 37 element bundle, thus facilitating a direct comparison between the thermohydraulic performance of the two bundles.

Both single and two phase pressure drops, and critical heat flux data were taken. The flow conditions covered the following ranges: 6–11 MPa (outlet pressure), 7–25 kg/s (flow) and 200–290°C (channel inlet temperature).

Overall, the dry out power values were consistently higher for the CANFLEX bundle than for the 37 element bundle. At inlet flow conditions of interest, and with a liner exhibiting 5.1% creep, the dry out power measurements were, on average, 17% higher for the CANFLEX bundle than for the 37 element bundle (Fig. 145). Under the same channel inlet conditions, both the single and two phase pressure drops were very similar to those of 37 element fuel. Thus, there should be no significant effect on overall reactor operation during transition refuelling.

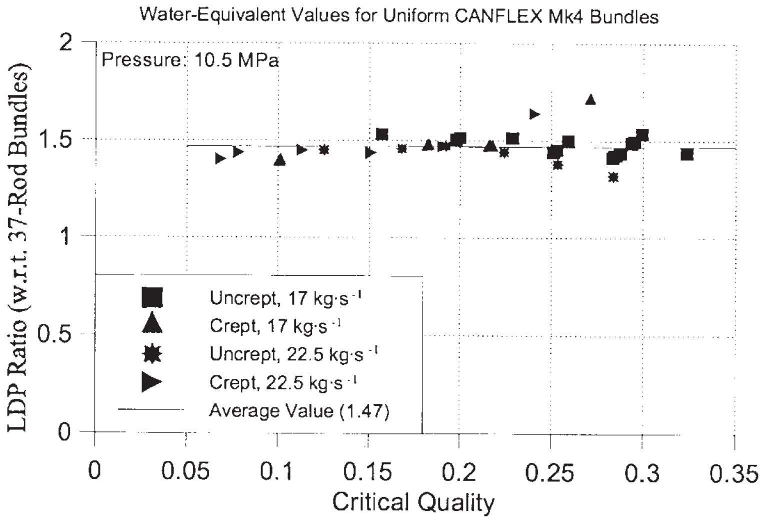


FIG. 145. Dry out power enhancement of CANFLEX over the 37 element bundle string inside an axially non-uniform channel with a maximum diameter 5.1% greater than the reference value [168].

6.3.2.5. Pressure drop testing

The pressure drop characteristics of the CANFLEX bundle were determined in both freon tests and in hot and cold water tests. KAERI tested a full string of CANFLEX bundles and 37 element bundles in its hot test loop under normal reactor operating conditions [169]. AECL has studied the axial pressure profiles of CANFLEX bundles in the freon MR-3 facility. A mixed junction of 37/43 bundles and the effect of bundle rotation were studied.

A comparison of the results obtained for the CANFLEX bundles with those obtained for the 37 element bundles is shown in Fig. 146. The results indicate that, for the same flow conditions, the CANFLEX bundle will reduce the adiabatic pressure drop by about 2% for both uncrept and crept channels. The refuelling of CANFLEX bundles in the PLGS during the demonstration irradiation confirmed that the difference in pressure drop between the two bundle types was within the measurement accuracy.

6.3.2.6. Out-reactor flow testing

Over the last several years, both AECL and KAERI have subjected the CANFLEX fuel bundle to a set of out-reactor flow tests aimed at simulating reactor

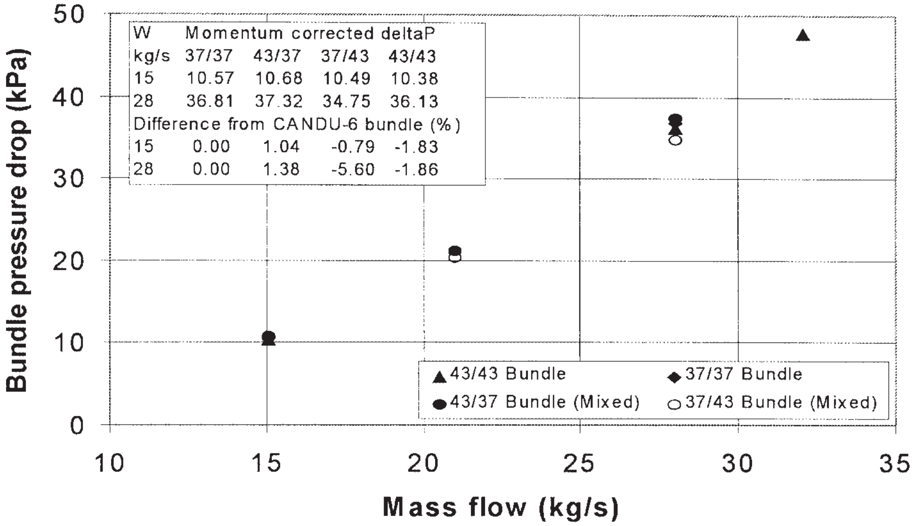


FIG. 146. Pressure drop for 37, 43, 37/43 and 43/37 element bundles.

conditions and verifying that the design is compatible with existing reactor hardware [169]. In addition to the heat transfer and pressure drop tests, the following mechanical flow tests have been successfully completed:

- Strength tests showed that the fuel can withstand the hydraulic loads imposed on it during refuelling. Post-test bundle geometry measurements showed no significant distortion.
- Impact tests showed that the CANFLEX bundle can withstand bundle impact during refuelling.
- Cross-flow tests demonstrated that during refuelling, when the bundle is in the cross-flow region in the end fitting, the bundle can withstand the flow induced vibration for a minimum of four hours.
- Fuelling machine compatibility test demonstrated that the bundle is dimensionally compatible with the fuel handling system (and in particular, with the side stops in the fuelling machine) [169].
- Flow endurance test demonstrated that the CANFLEX bundle maintains its structural integrity during operation. Fretting wear on the bearing pads, interelement spacers and pressure tube remained within design limits over the 3000 h test duration.

6.3.2.7. *In-reactor testing*

CANFLEX bundles AJK, AJM and AJN were irradiated in the U1 and U2 loops of the NRU research reactor at AECL's Chalk River Laboratories to demonstrate performance in-reactor. Peak powers experienced in the NRU irradiation were more than 25% higher than those calculated for a CANDU 6 reactor. Once the bundles were removed, a detailed post-irradiation examination was performed. The irradiation requirements and corresponding results are summarized in Table XXV. All design requirements were met.

6.3.2.8. *Reactor physics testing and analysis*

The ZED-2 facility at Chalk River Laboratories was used to measure fine structure, reaction rates and reactivity coefficients for CANFLEX natural uranium fuel bundles in order to validate the reactor physics lattice code WIMS-AECL [170]. The data showed excellent agreement with code predictions. Reactor operation for over 600 full power days was simulated to determine peak bundle powers and power changes occurring during refuelling, burnups and residence times. This information was used to define the power/burnup history for the NRU irradiation.

6.3.2.9. *Structural analysis*

The CANFLEX design was analysed for sheath strains, fission gas pressure, end plate loading, thermal behaviour, mechanical fretting, element bow, end flux peaking and a range of other mechanical characteristics. Acceptance criteria established after years of operating experience gained with 37 element fuel and previous 37 element testing were met by the CANFLEX design [171, 172].

6.3.2.10. *Formal design review*

The final stage in the design verification was a formal review of the design by a design review panel, chaired by AECL's chief engineer and consisting of experts from the various disciplines relevant to fuel design and selected from the CANDU utilities and fuel fabricators, as well as experts from within AECL not directly involved in the CANFLEX programme. The results of the testing and analysis programme were evaluated against the design requirements set out in the design verification plan. All CANFLEX performance data were summarized in a fuel design manual. On the basis of the disposition of all issues raised by the panel, the chief engineer certified the CANFLEX design for the Point Lepreau demonstration irradiation.

TABLE XXV. IN-REACTOR IRRADIATION REQUIREMENTS AND RESULTS

Item	Capability to be demonstrated	Irradiation status
Primary coolant system	Elements retain fission products	Bundle AJK successfully irradiated in NRU at ~70 kW/m to 15.5 MW·d/kg HE
	Capability of primary coolant system to withstand coolant pressure	Performance acceptable
Fuel channel	Bundle compatibility (whether expansion and pressure tube sag and creep can be accommodated)	All CANFLEX bundles removed from pressure tube in NRU without problems Profilometry acceptable
Fuelling system/physics	Accommodate change in power with refuelling	Bundle AJM subjected to power increase in NRU Performance acceptable
Fuel management	Capability to withstand continuous high power	Bundle AJK: 15.5 MW·d/kg HE, ~70 kW/m Bundle AJM: 18.75 MW·d/kg HE, ~65 kW/m Bundle AJN: 19.5 MW·d/kg HE, ~69 kW/m Performance acceptable
	Capability to withstand power changes by ripples, reactivity shims and refuelling sequences	Bundle power envelopes bound peaks Performance acceptable
	Capability to withstand end flux peaking	Post-irradiation examination showed that bundle AJK withstood end flux peaking Performance acceptable

6.3.2.11. CANFLEX demonstration irradiation: Strategy and plan

The final step in the verification of the CANFLEX bundle before full-core implementation is a demonstration irradiation in a power reactor. At the PLGS it was decided to implement a limited 24 bundle demonstration irradiation as a critical step towards a decision to implement full-core CANFLEX fuelling.

Twenty-six CANFLEX bundles were fabricated by ZPI to the quality assurance levels normally applied to the 37 element fuel supplied to PLGS. Twenty-four of

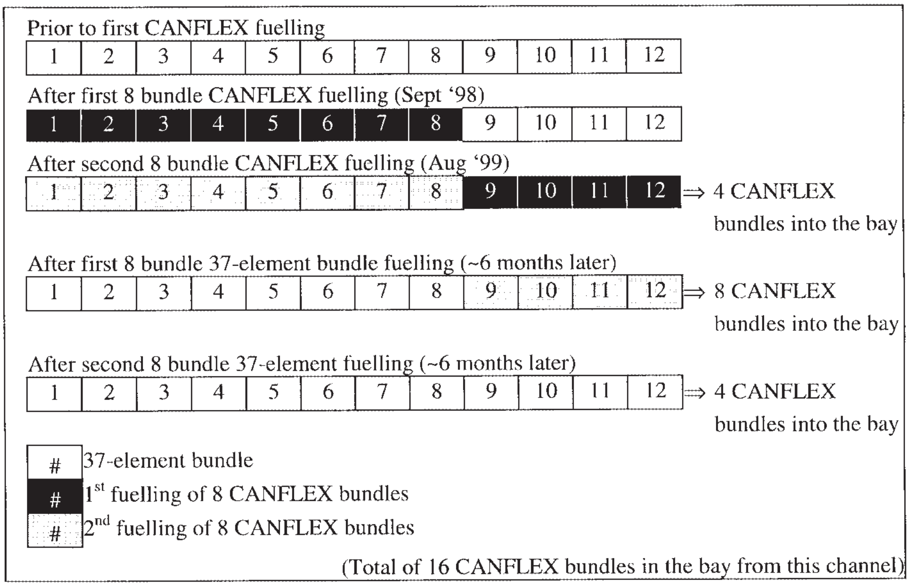


FIG. 147. Projected fuelling history for the high power channel (S08).

these bundles have been fuelled into two channels. All configurations of CANFLEX bundles mixed with 37 element bundles in a single channel during transition and full-core refuelling will be tested.

Two channels at PLGS were each refuelled with eight CANFLEX bundles in September 1998: the high power channel (S08) and the lower power, instrumented channel (Q20). Channel Q20 was subsequently (March 1999) refuelled with eight, 37 element bundles, demonstrating the successful interaction of the CANFLEX fuel bundles with the side stops in the refuelling machine during refuelling. The refuelling of channel S08 with eight more CANFLEX bundles (resulting in a full channel of 12 CANFLEX fuel bundles) took place in August 1999. The fuelling plan for each channel is indicated in Figs 147 and 148.

As the CANFLEX bundles are discharged and transported to the bays they will be subject to visual examination. Several CANFLEX bundles will be shipped to the Chalk River Laboratories for post-irradiation examination.

6.3.2.12. CANFLEX demonstration irradiation: Safety analysis

To secure approval for the demonstration irradiation, both PLGS and AECL analysed all design basis accidents to determine CANFLEX performance. The principal design differences between CANFLEX and the 37 element bundles are:

- Higher dry out power, providing greater operating margin;
- Lower maximum linear element ratings, leading to lower centre line temperatures and lower fission gas release;
- Smaller diameter outer elements, which run at lower temperatures but which could be less rigid;
- Slightly higher void reactivity because of the 5% increase in coolant cross-sectional area and greater bundle subdivision;
- Smaller Zircaloy mass but higher surface area, which could affect hydrogen production.

The safety studies have shown that CANFLEX will maintain acceptable safety margins for all postulated accidents. The results of these studies were summarized in an information report submitted to the AECB.

6.3.2.13. CANFLEX: Summary

CANFLEX fuel has been under development for over ten years. The CANFLEX fuel bundle has been verified through extensive testing by both AECL and KAERI. The demonstration irradiation of 24 bundles in PLGS is the final verification of the compatibility of this fuel type with existing reactor systems. The utility decision to implement CANFLEX fuel will depend on the economic benefits being confirmed with the water critical heat flux licensing data. Effort is now being concentrated on establishing licensing data and completing the safety assessments for full-core implementation.

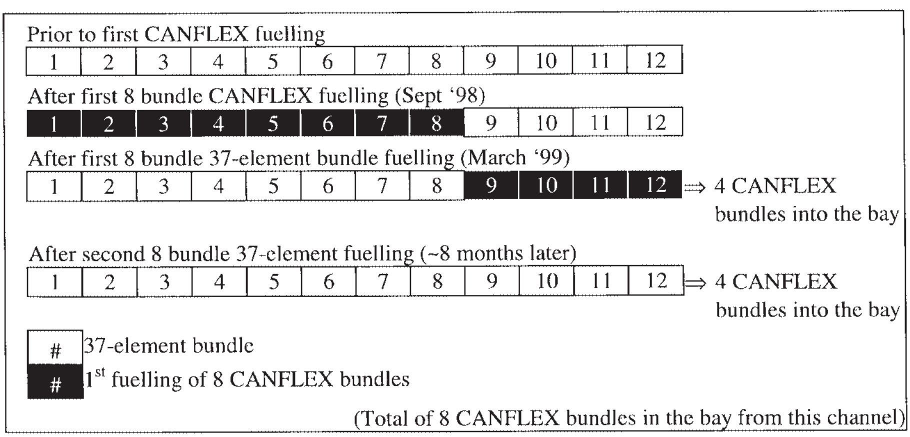


FIG. 148. Projected fuelling history for the low power channel (Q20).

6.3.3. Indian experience

6.3.3.1. The 22 element fuel bundle

The 22 element bundle design was specifically developed for higher bundle powers, compared with the 19 element fuel bundle currently in use. The bundle has three concentric rings of fuel pins, consisting of 1, 7 and 14 pins. The 14 pins of the outermost ring are smaller in diameter than the remaining 8 pins, which leads to reduced rating in the outer pins.

Figure 149 shows the cross-section of this bundle and descriptive data are given in Table XXVI. A total of 522 22 element fuel bundles have undergone irradiation at Narora 1.

6.3.3.2. The 37 element fuel bundle

The 37 element bundle is being developed for use in the 500 MW(e) HWR. This bundle has four concentric rings of fuel pins, comprising 1, 6, 12 and 18 pins,

TABLE XXVI. DESCRIPTION OF THE INDIAN 22 ELEMENT FUEL BUNDLE

Parameter	Value
Fuel material	Natural UO_2
Number of fuel rods in bundle	22
Sheath material	Zircaloy 4
Fuel pin data for the 8 inner pins:	
Diameter of fuel rod (cm)	1.44
Inner diameter of sheath (cm)	1.45
Outer diameter of sheath (cm)	1.52
Fuel pin data for the 14 outer pins:	
Diameter of fuel rod (cm)	1.22
Inner diameter of sheath (cm)	1.23
Outer diameter of sheath (cm)	1.31
Diameter of first ring (7 pins) (cm)	3.75
Diameter of second ring (14 rods) (cm)	6.58
Inner diameter of pressure tube (Zircaloy) (cm)	8.30
Outer diameter of pressure tube (cm)	9.10
Air gap thickness (cm)	0.85
Inner diameter of calandria tube (cm)	10.80
Outer diameter of calandria tube (cm)	11.10
Calandria tube material	Zircaloy
Coolant purity (D_2O, wt%)	99.7

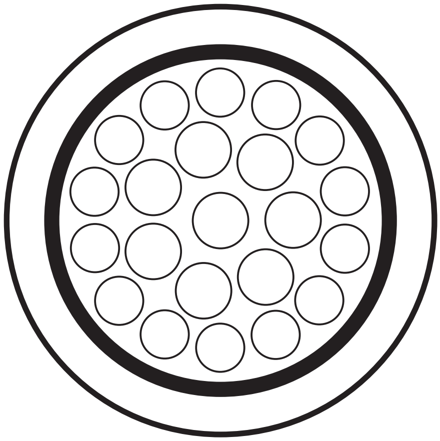


FIG. 149. Cross-section of the Indian 22 element fuel bundle.

respectively. Figure 150 shows the bundle cross-section and descriptive data are given in Table XXVII.

6.3.3.3. The AHWR fuel cluster

The AHWR, currently being designed in India, is an advanced reactor system in which special emphasis has been placed on the optimal utilization of thorium. A description of the reactor and fuel bundle assembly is given in Section 6.9.4.

6.3.4. Romanian experience: Development of Romanian SEU 43 fuel bundle [173]

6.3.4.1. Introduction

HWRs have the capability to use a wide variety of fuel types. As discussed in Section 6.4.2, the use of SEU offers many advantages [174]. While some of these benefits can be obtained by using existing technology, increased operating margins and greater confidence in extended burnup fuel behaviour are possible in a more subdivided bundle design.

In 1990, the Institute for Nuclear Research (ICN (Pitesti)), started a general research programme aimed at developing a new fuel bundle for extended burnup operation. It adopted a staged strategy in which each step was based on the results obtained in the preceding steps [175]. The design arrived at is the result of a long

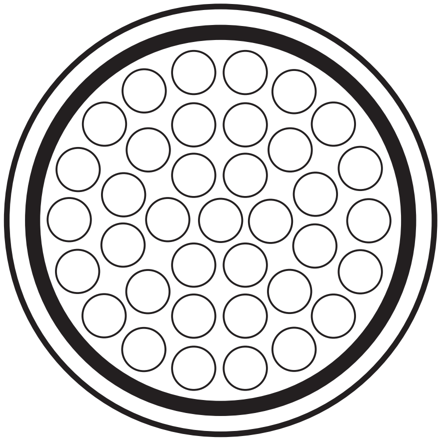


FIG. 150. Cross-section of the Indian 37 element fuel bundle for 500 MW(e) HWR.

TABLE XXVII. DESCRIPTION OF THE INDIAN 37 ELEMENT FUEL BUNDLE

Parameter	Value
Fuel material	Natural UO_2
Number of fuel rods in bundle	37
Sheath material	Zircaloy 4
Diameter of fuel rod (cm)	1.22
Outer diameter of sheath (cm)	1.30
Cladding thickness (cm)	0.038
Diameter of first ring (6 pins) (cm)	2.97
Diameter of second ring (12 rods) (cm)	5.73
Diameter of third ring (18 rods) (cm)	8.63
Inner diameter of pressure tube (Zircaloy) (cm)	10.40
Outer diameter of pressure tube (cm)	11.30
Air gap thickness (cm)	0.65
Inner diameter of calandria tube (cm)	12.60
Outer diameter of calandria tube (cm)	12.90
Calandria tube material	Zircaloy

process of analyses and improvements. This section discusses the most relevant calculations performed on this fuel bundle design, and an experimental programme aimed at verifying the operating performance of the new bundle.

6.3.4.2. SEU 43 bundle design

The major feature of the SEU 43 bundle is an increase in the number of fuel elements from the 37 present in the standard CANDU 6 bundle to 43 elements. The SEU 43 bundle consists of two fuel element sizes: 11.78 mm diameter elements in the outer ring, and 12.4 mm diameter elements in the intermediate, inner and centre rings (Fig. 151). This contrasts with the CANFLEX bundle (Section 6.3.2) which has smaller diameter elements in both of the outer two rings of fuel. The peak linear element ratings are reduced by 16% in comparison with the standard 37 element bundle [176]. The larger diameter elements in the inner rings of the bundle compensate for the fuel volume loss due to the smaller diameter outer elements. To maintain compatibility of the new bundle with the existing CANDU 6 reactor systems, the overall dimensions of the SEU 43 fuel bundle were designed to be the same as those of the 37 element bundle. The use of smaller diameter elements in the outer ring results in a slight eccentricity (cf. the standard 37 element bundle) where the fuel element end caps are welded to the end plate web. This feature makes the SEU 43 fuel bundle fully compatible with the side stop/separator assembly of the CANDU 6 fuelling machine. An important consideration is the clearance between the bundle's end plate and the fuelling machine side stops, and to ensure that there is adequate clearance during refuelling. During this operation, the engagement of the side stops with the end caps of the fuel elements has to be adequate to prevent damage to the fuel itself. These features are shown in Fig. 152.

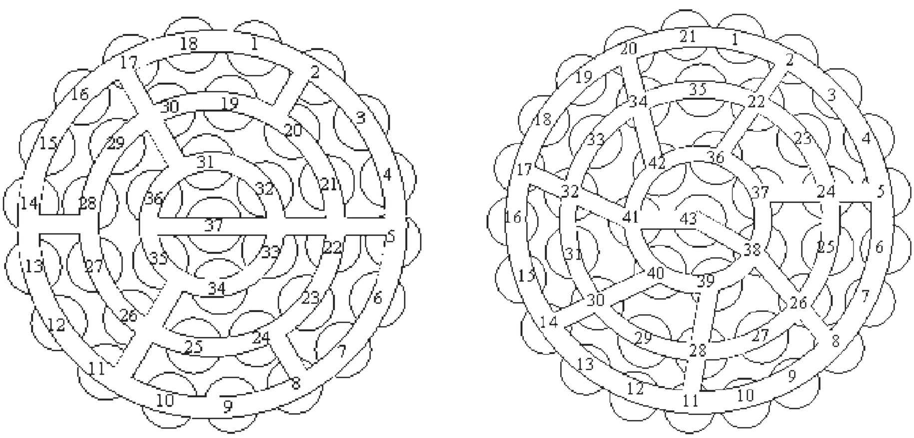


FIG. 151. Comparison of standard 37 element and SEU 43 fuel bundle designs.

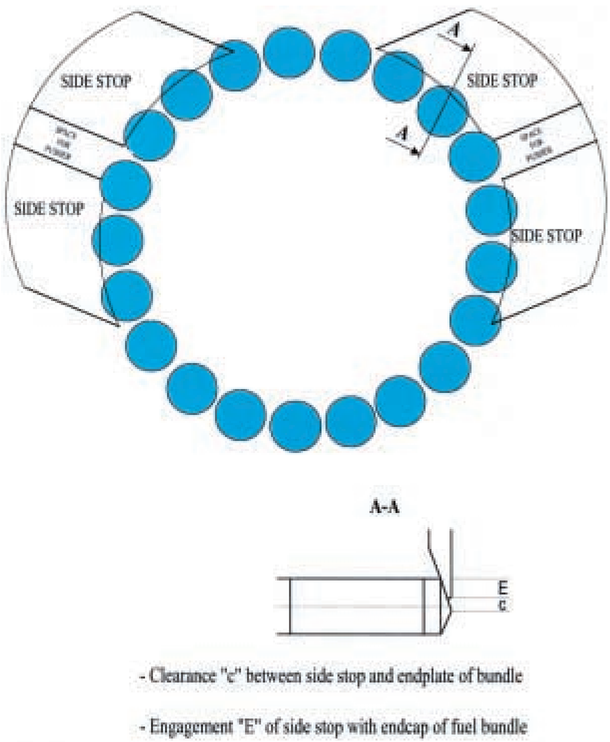


FIG. 152. Interaction between SEU 43 bundle and CANDU 6 fuelling machine side stops.

The detailed design features of the bundle have continued to evolve as a result of ongoing design analysis and thermohydraulic testing. The main specifications of the SEU 43 fuel bundle and fuel elements are given in Tables XXVIII and XXIX, respectively. Test programmes are under way to demonstrate all hydraulic characteristics of the bundle and the irradiation behaviour of the fuel elements.

6.3.4.3. Fuel element design optimization

The optimization of the SEU 43 fuel bundle design with respect to extended burnup application started with a review of the intended design objectives. In order to reduce any life limiting factors at extended burnup, a set of solutions was conceived for the SEU 43 fuel element design (Table XXX) [175, 177, 178]. The specific details of the SEU 43 optimized design are shown in Table XXIX.

TABLE XXVIII. MAIN CHARACTERISTICS OF SEU 43 FUEL BUNDLE

Parameter	Specification
Structural material	Zircaloy 4
Type of assembly	Welded bundles of 43 elements arranged in circular array with brazed appendages
Length of bundle (cm)	49.53
Diameter of bundle (cm)	10.229
Weight (kg, nominal)	23.4
End plate thickness (cm, minimum)	0.158
End plate diameter (cm)	9.08

TABLE XXIX. MAIN CHARACTERISTICS OF SEU 43 FUEL ELEMENTS

Parameter	Main specification	
	Inner ring fuel element	Outer ring fuel element
Fuel rod:		
Total length (cm)	49.265	49.265
Active length (cm)	47.730	47.685
Axial gap (cm)	0.40	0.45
Diametral gap (cm)	0.009	0.009
Filling gas	80% He	80% He
Pellet:		
Diameter (cm)	1.155	1.093
Height (cm)	1.256	1.192
Material	UO_2	UO_2
Enrichment (wt% U-235)	0.9	0.9
Density (% theoretical density)	96.7	96.7
Grain size (μm)	3.50	3.50
Cladding:		
Outer diameter (cm)	1.240	1.178
Thickness (cm)	0.038	0.038
Material	Zircaloy 4	Zircaloy 4

TABLE XXX. DESIGN OBJECTIVES AND SOLUTIONS FOR EXTENDED BURNUP FUEL ELEMENT

Design objectives	Design solution
Decrease of fuel element linear power and average fuel temperature	Decrease of element diameter in outer ring of fuel
Ensuring that fission gas release is within acceptable limits	Increase of initial pellet grain size
Minimization of local strain in sheath ridges and reduction of stress corrosion cracking failure susceptibility in power ramps	Increase of pellet dish volume Increase of pellet land width Increase of pellet chamfer
Accommodation of axial fuel stack expansion	Increase of axial gap
Reduction of stress corrosion cracking failure susceptibility	Increase of graphite layer thickness

6.3.4.4. SEU 43 fuel element design evaluation

(a) Deterministic approach

The process of defining the new fuel element design started with the identification of the extended burnup limitations in the existing fuel element design. The possible design solutions were reviewed, taking into account the proposed design objectives. The first step was the review of the design criteria and limits. As long as new, extended burnup dependent phenomena were not identified, the actual design limits for the CANDU type fuel could be maintained. Regarding the fuel element internal pressure, the actual limit (coolant pressure), which may be restrictive with respect to extended burnup operation, cannot be modified or replaced owing to the use of thin, collapsible sheathing. Consequently, significant design effort was directed towards maintaining the internal pressure below the design limit [178].

The new bundle operating conditions (maximum power, average burnup, typical power history envelope and coolant parameters) that were established during the initial stages of the design process were subsequently prepared as input data for the design codes. In assessing the influence of the selected design solutions on fuel element performance, a conservative approach was primarily used, coupled with a comparison made with the performance of the standard 37 element bundle design under similar conditions. In the cases where the results of the conservative calculations exceeded the design limit, a probabilistic alternative was used [179]. A detailed analysis was performed for each of the performance parameters and the most significant results selected [178].

The value obtained for the highest fuel central temperature was below the design limit (melting temperature) and was also almost 200°C lower than the central temperature calculated in the reference 37 element bundle for similar bundle powers. The cladding 'end of life' circumferential strain at the mid-pellet position was calculated in a conservative manner (power envelope and the most unfavourable combination of input parameters) and only amounts to 0.43%, a value which lies below the design limit and which also lies in the range of experimental results obtained with the reference 37 element bundle.

The local mechanical behaviour of the fuel cladding in typical power ramps has been analysed with the ROFEM code in various situations [178]. Parallel calculations have been performed on the reference fuel under similar bundle power conditions. For example, in the highest anticipated power ramp for the SEU refuelling scheme, the calculated values of the circumferential deformation of the cladding at the pellet interfaces for the SEU 43 element are only half those calculated for the standard 37 element design, a direct consequence of the modified pellet geometry.

(b) Probabilistic approach

Concurrent analytical and experimental programmes have been conducted to support the SEU 43 design, and knowledge about potential performance affecting phenomena associated with increased burnup has increased. Fuel design analyses using current methodology often only allow limited burnup increases. Therefore, effort is needed to find solutions that permit further increases in burnup. Besides fuel bundle and fuel element design improvements, the removal of excess conservatism from design criteria and from design calculations can contribute to this goal [180]. The main means of eliminating excess conservatism from design calculations at extended burnup involve model refinement, database improvement and the utilization of probabilistic design methods.

An example of the probabilistic methodology applied to the evaluation of a particular CANDU type fuel design is presented below. An essential prerequisite for such an analysis is a best estimate fuel element behaviour modelling code that must incorporate up to date knowledge about potential performance limiting phenomena at extended burnup. The above mentioned version of the ROFEM code includes such developments (e.g. burnup dependent degradation of UO_2 thermal conductivity, burnup dependence of radial heat generation rate) and has demonstrated good prediction capabilities in the FUMEX blind comparison exercise [181, 182]. The probabilistic methodology has been applied to the evaluation of the effect of variations of fuel element design data within their tolerances on calculated fuel performance parameters. The outer element linear power histories utilized in this set of calculations were derived from the bundle power history envelope presented in Fig. 153, which was calculated from core management simulations for 1.2% SEU and a two bundle shift refuelling scheme, giving a bundle average burnup of ~21 MW·d/kg HE.

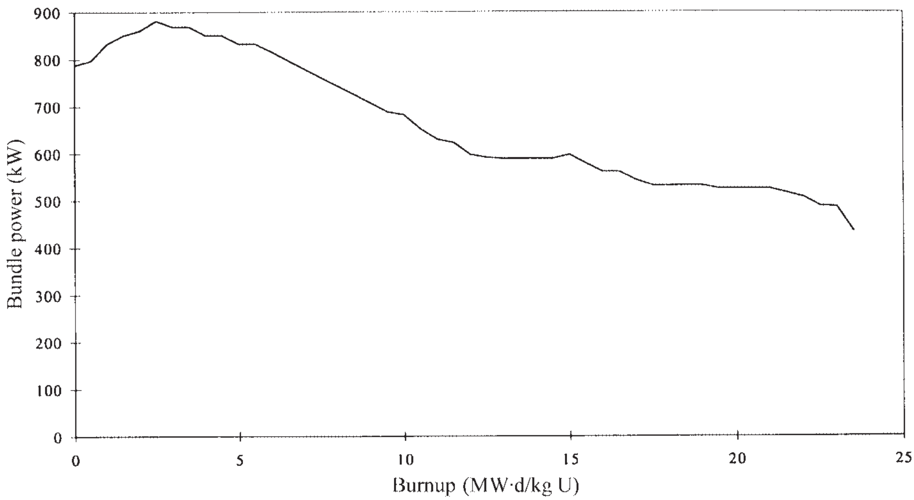


FIG. 153. Bundle power envelope used: Enriched CANDU fuel.

Figure 154 shows calculations of outer fuel element internal pressure for three different situations. The curve denoted '37 nominal' represents the calculated internal pressure in the outer element of a standard geometry 37 element bundle containing enriched UO_2. The upper limit of internal pressure for this nominal case is very close to the coolant pressure. The 'worst case' conservative calculation produced significantly higher values for the 37 element bundle, and is not shown. Also shown

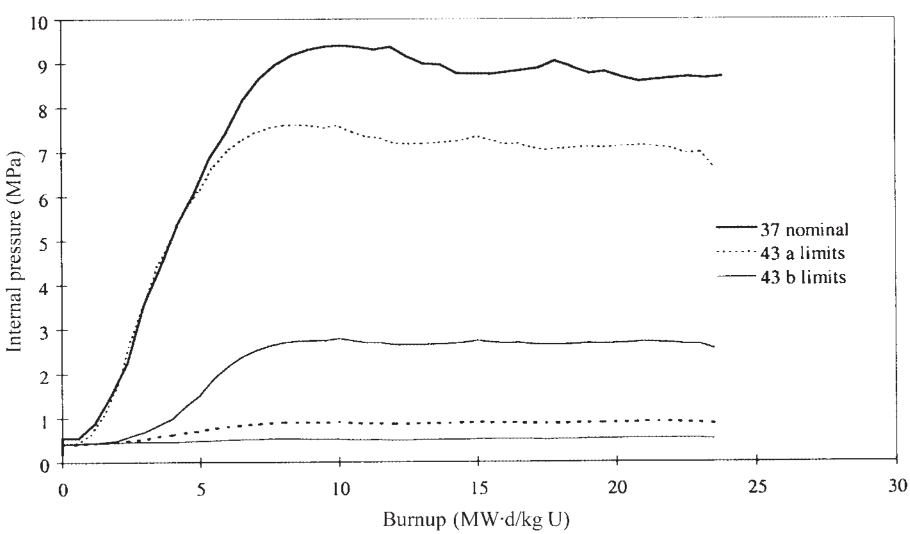


FIG. 154. Calculated uncertainty bands of internal pressure for two fuel element designs.

is the calculated upper uncertainty band limit for the outer elements of the new 43 element bundle design (denoted 43a). A version of this design, containing UO_2 pellets with a different grain size range (25–45 µm), was investigated using the same approach. The calculated upper uncertainty band limit for this version is shown as 43b; the upper limit for internal pressure is significantly reduced.

Other important results are shown in Fig. 155, which shows the probability density function of end of life internal pressure for the 43b design, calculated using two available methods of response surface equations. It also includes the cumulative probability calculated for the same parameter, a function that can be used directly for

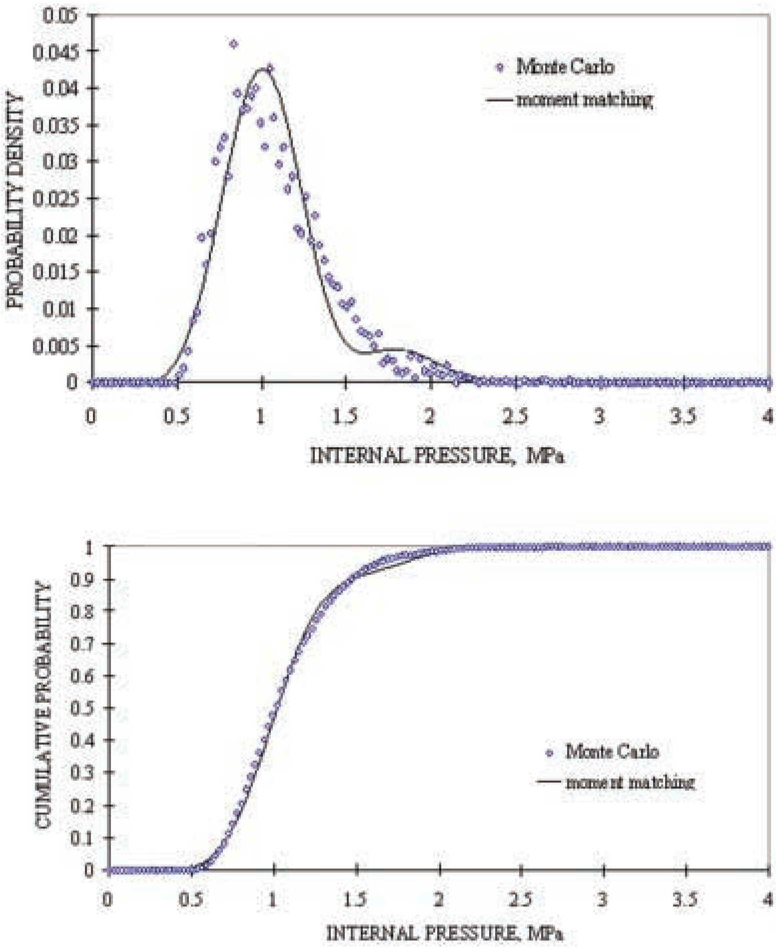


FIG. 155. Calculated probability density function and cumulative probability for end of life internal pressure (43b fuel element design).

evaluating the probability of exceeding a given value (threshold). Figure 156 shows the contribution to the variance of calculated end of life internal pressure (43b case) of five input parameters that have been selected in a preliminary RMS sensitivity analysis.

Another important advantage of this approach is the possibility of correctly establishing the worst case combination of input variables for a given case and calculating the associated probability of occurrence. This is the practical method of quantifying the conservatism of design calculations and also one which serves as a basis for comparing designs.

6.3.4.5. Fuel bundle out of pile testing

A series of pressure drop tests have been performed to verify that the SEU 43 bundle meets the acceptance criteria and to provide relevant test data for the evaluation of the new fuel design. The fuel string pressure drop was measured in light water for both SEU 43 bundles and reference 37 element bundles to enable a comparison to be made under the same test environment. The conditions for the pressure drop test comprised flow rates of 10–30 kg/s at a temperature of 100–289°C, and fuel channel inlet pressure of 5–10.4 MPa. The test results for the SEU 43 and reference 37 element bundle strings are compared in Fig. 157. From a hydraulic point of view, the major differences between the two bundle designs are in the flow area and wetted perimeter. Statistical analysis of all experimental data suggests that the pressure drop for the SEU 43 bundle (fully aligned) is about 3.8% lower than that for the 37 element bundle [183]. A pressure drop test using freon as a modelling fluid

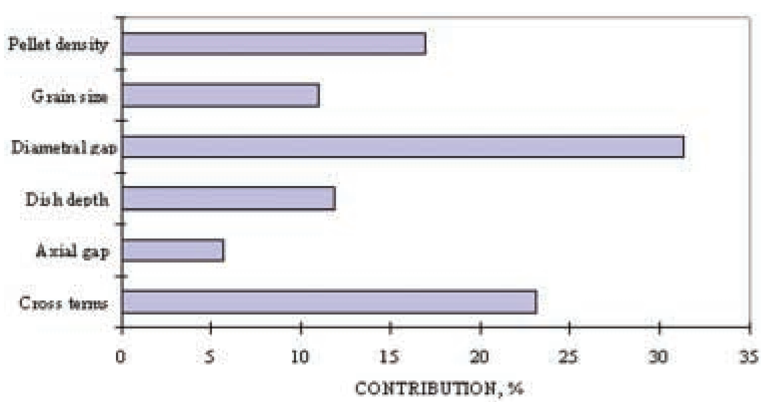


FIG. 156. Contribution to end of life internal pressure variance (43b fuel element design).

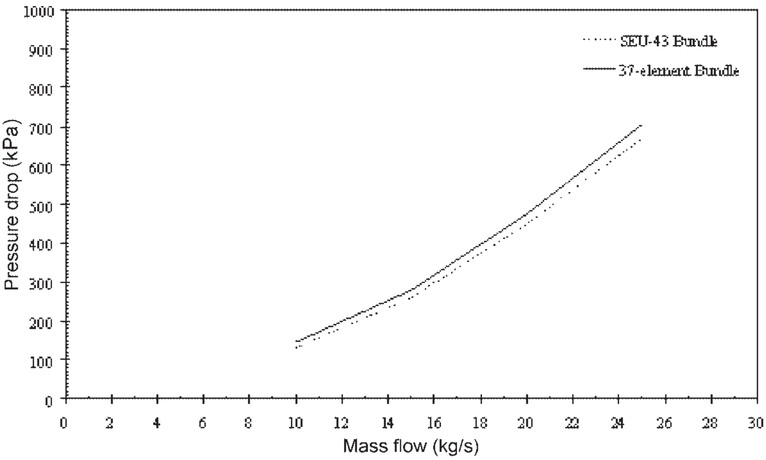


FIG. 157. Comparison of 37 element bundle and SEU 43 bundle pressure drop tests.

rather than light water offers the possibility of obtaining more precise measurements of bundle junction pressure drop at Reynolds numbers closer to actual reactor conditions.

Two flow based strength tests are required for all bundle designs for CANDU reactors: one to demonstrate that the bundle can support the hydraulic drag of a string of fuel bundles on the fuelling machine side stops without incurring damage to the bundle, and the other, a single side stop test, to demonstrate that the hydraulic force of a full channel of bundles can be carried by one side stop. Both tests have been completed at ICN (Pitesti) and the results indicate that the bundle can withstand the hydraulic forces. A refuelling impact test, simulating the impact of a new fuel bundle in coming to rest against the stationary bundles in the channel during refuelling, is also required. Also, flow excitation tests are required to investigate the vibration excitation characteristics of the smaller diameter outer elements in the SEU 43 bundle. It can be anticipated that the smaller diameter SEU 43 element has only a slightly lower natural frequency, and a slightly higher vibration amplitude, than the corresponding element of a 37 element bundle.

Special tests are planned for both the SEU 43 bundle and the standard 37 element bundle geometries to study how the bundles interact with the side stops under axial loads up to and beyond normal design load. To make these measurements and refine the design of the bundle, the design and fabrication of a special ‘side stop fixture’ is planned. With this device, a bundle could be positioned against the side stops. The final ‘acceptance’ test for fuelling machine compatibility involves the use of an actual CANDU 6 fuelling machine.

The final test required is a flow endurance test, which is used to demonstrate acceptable vibration and fretting behaviour of the SEU 43 bundles in a fuel channel operating under a specified set of flow conditions.

6.3.4.6. In-reactor experiments

In order to demonstrate the performance of the SEU 43 fuel element design and prove the adequacy of the manufacturing technologies, experimental fuel elements have been introduced in the irradiation devices of the ICN (Pitesti)–TRIGA research reactor. One of the objectives of these fuel behaviour studies is to predict, through investigation, and with reliability, the fuel element performance during power cycling.

The power cycling experiment has been performed in a specially designed irradiation device known as capsule C9 [184]. The specified variation in the fuel element linear power was obtained by the mechanical movement of the device in and out of the TRIGA reactor core. During the power cycling test, the experimental fuel element successfully experienced up to 367 power cycles, mostly between 50% and 100% of the specified linear power, underlining the role played by the CANLUB graphite coating in preventing stress corrosion cracking defects from arising during load following manoeuvres. Post-irradiation examination results indicate a maximum cladding ridge strain of 0.7% in the region having the highest linear power. There are also indications of strong axial interaction between the pellet column and the end cap.

6.3.4.7. Summary

A development programme aimed at introducing extended burnup SEU fuel in the Cernavoda CANDU 6 reactors has been started at ICN (Pitesti) and the results from the early phases are available. The following conclusions can be drawn:

- The major feature of the SEU 43 bundle is the increased number of fuel elements (43 as opposed to 37). The SEU 43 bundle consists of two fuel element sizes (with the smaller diameter element confined to the outer ring).
- A new fuel element design is included in the SEU 43 fuel bundle, and this was developed using a set of selected design solutions to reduce the probability of failure at extended burnup.
- A calculation assessment of the final SEU 43 element design has been performed, highlighting the good performance achieved at extended burnups in comparison with the standard design.
- The probabilistic approach was extremely useful in the process of fuel design development and comparison between options. It can provide a quantitative measure of the effect of design differences, and a measure of the importance of the tolerances specified for the fuel. This approach also has the capability of

characterizing the degree of conservatism in fuel design analysis by estimating the probability of occurrence of extreme values.

- As a part of the development programme, four experimental SEU 43 bundles have been fabricated and utilized in typical out-reactor tests (pressure drop and axial compression). The results of the tests met acceptance criteria.
- During the power cycling test, the experimental fuel element successfully experienced up to 367 power cycles.
- A refuelling impact test, a flow endurance test and a fuelling machine compatibility test are planned for the future.
- Fuel performance models and codes will be updated to incorporate the new experimental results and thereby ensure continuing predictive capability for further developments.

6.3.5. Argentinian experience: The CARA bundle

6.3.5.1. Introduction

Argentina has two HWR reactor types: the Embalse 648 MW(e) CANDU reactor and Atucha 1, a 360 MW(e) pressure vessel HWR. Although both reactors use 37 element natural uranium fuel, the two fuel types are quite different. The CANDU fuel is the standard 37 element bundle, 0.5 m long, simple to fabricate and with low fuelling costs. The Atucha 1 fuel is 6 m long, quite complex and characterized by high manufacturing and fuelling costs. Both fuels are made domestically by the fuel manufacturing company CONUAR SA.

In order to reduce the overall nuclear fuel cycle costs in Argentina, a new fuel bundle is being developed that can be used in both reactors—Combustible Avanzado para Reactores Argentinos (CARA) [185]. This is a good example of the exploitation of synergism between two reactor types (through employment of a common fuel bundle in both reactors), and of the optimization of the overall fuel cycle, rather than just one component of it.

The objectives of developing the CARA fuel bundle are to:

- Use the same fuel for both reactors,
- Increase the heat transfer area,
- Use a single fuel rod diameter,
- Decrease the fuel centre temperature,
- Decrease the ratio of zirconium to uranium masses,
- Retain the higher uranium density of the CANDU fuel,
- Keep the hydraulic pressure drop per channel of each reactor constant,
- Achieve higher burnup using SEU,

- Keep fabrication cost within the fabrication cost of the 37 element CANDU fuel in Argentina.

While the CANFLEX bundle achieves some of these objectives, it has two different rod diameters, and hence does not meet the third objective. Hence, a new design was developed.

If the number of fuel rods were increased, the core pressure drop being kept constant, the frictional pressure drop would also increase owing to the smaller equivalent hydraulic diameter. To compensate for the higher frictional pressure drop, a double length bundle (1 m in length) was adopted. This avoids the sizeable pressure drop due to the end plate and bundle junctions.

To check this approach, Fig. 158 shows the relationship between element radius and the number of elements for a given fuel channel, keeping either the uranium mass constant (constant mass curve), or the hydraulic pressure drop constant (constant ΔP curve). Clearly, both curves decrease monotonically as the element diameter increases, but the constant ΔP curve drops faster than does the constant mass curve. If the standard 37 element, CANDU 6 bundle is taken as a reference, both curves have the same element radius for 37 rods.

However, if a double length bundle is used, both curves cross when the bundle contains 66 rods. The smaller pressure drop due to the elimination of half of the end plates and bundle junctions, together with a new concept of spacer design employing reduced pressure drop, offset the increased frictional pressure drop due to the increased number of fuel elements. As the CARA fuel bundle must be compatible

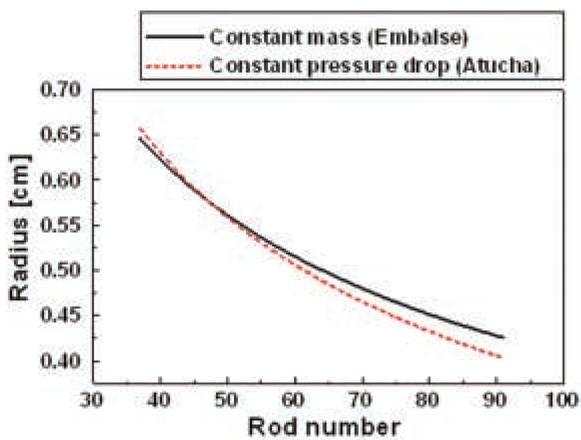


FIG. 158. Relationship between the 'mass' and 'hydraulic' radius of CANDU and Atucha fuels.

with both types of fuel channel, the most restrictive of the mass and ΔP curves must be used, with Atucha having the most restrictive ΔP requirements and Embalse the most restrictive mass requirements. Using both curves, and with a 1 m long bundle, approximately 50 rods could be used for both reactors, as is shown in Fig. 158.

6.3.5.2. Bundle geometry

For studying different bundle geometries, a symbolic algebraic language enables very simple and fast evaluations to be made, including assessment of rod ring rotation and different central rod numbers. Using this approach, four final geometries were studied having 48–52 fuel rods. Finally, a 52 element bundle geometry was chosen on the basis of good symmetry and compactness of the array. This geometry, shown in Fig. 159, has rings of 4, 10, 16 and 22 elements. The reduction in the number of end plugs and end plates, resulting from the employment of a double length bundle, gives a uranium credit that can be used to increase the bundle uranium mass.

The pellet diameter is very similar to that of the smaller CANFLEX elements, but with a slightly greater cladding thickness and gap clearance compared with the standard 37 element bundle.

6.3.5.3. Neutronic calculations

The lattice code WIMS D/4 was used to estimate the burnup of the CARA fuel bundle, normalizing to the core excess reactivity [186]. The beginning of life excess

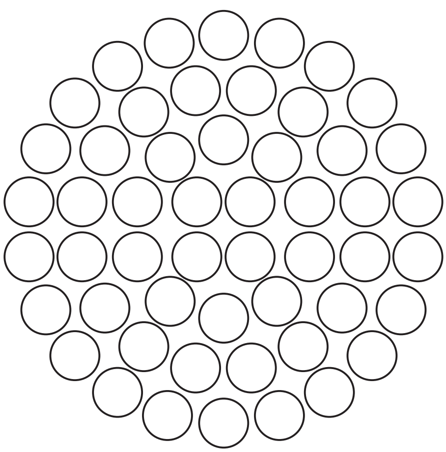


FIG. 159. Geometry of the CARA bundle with 52 fuel rods.

reactivity, power peaking factor and burnup level are shown in Table XXXI, for both natural uranium and SEU, and for both types of reactor in Argentina. Using the power evolution, burnup and peaking factor calculated with WIMS, together with the geometry and compositions, the complete thermomechanical behaviour could be calculated for the most restrictive CARA rods.

6.3.5.4. Thermomechanical behaviour of CARA fuel element

A hypothetical power history for 37 element CANDU fuel is shown in Fig. 160. High power is reached (~60 kW/m), and two shutdowns are included, a step by step increase in the power level being effected after the second shutdown. This hypothetical power history was then extrapolated to estimate the equivalent CARA fuel conditions in a CANDU reactor, using WIMS to adjust the linear element ratings and extrapolating the exit burnup to 15 MW·d/kg HE. With the CARA fuel bundle, the linear element rating is reduced to 72% of the original value with 37 element fuel.

These power histories were used as input for the BACO code for calculating the corresponding peak centre line fuel temperature (Fig. 161) of both the 37 element fuel bundle and the equivalent CARA fuel [187, 188]. The peak centre line temperature in the CARA fuel bundle is reduced by 500°C compared with the 37 element bundle.

Figure 162 shows the local power history of the seventh axial segment of a 5 m long Atucha 1 fuel element (from the top of the fuel string that contains ten axial segments). The seventh segment experiences the most demanding conditions during irradiation, encountering a maximum power level of 55 kW/m. This power history corresponds to the fourth module of a CARA assembly in Atucha 1. The burnup at end of life is 14.75 MW·d/kg HE and the power level is reduced to 73% of the original value in Atucha.

TABLE XXXI. WIMS RESULTS: NEUTRONIC CHARACTERISTICS OF A CARA FUEL ELEMENT FOR THE EMBALSE AND ATUCHA 1 REACTORS

Fuel type (natural U/SEU)	Embalse	Atucha 1	CARA (in Embalse)	CARA (in Atucha 1)
Natural uranium:				
Burnup (MW·d/kg HE)	7.50	6.10	7.53	6.37
Peaking factor (peak/av. ratings)	1.126	1.094	1.136	1.078
SEU (0.9%):				
Burnup (MW·d/kg HE)	14.54	13.47	14.58	14.52
Peaking factor (peak/av. ratings)			1.148	1.084

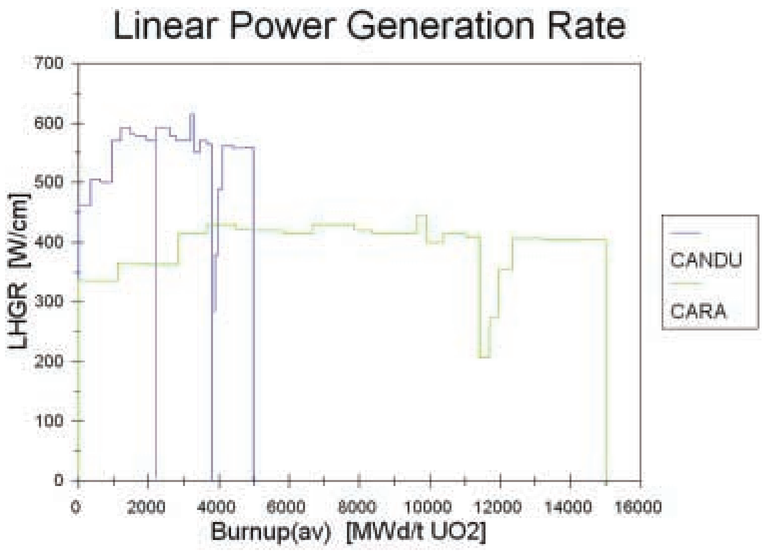


FIG. 160. Averaged power history of a 37 element fuel rod and the equivalent history of a CARA fuel element in the Embalse CANDU reactors.

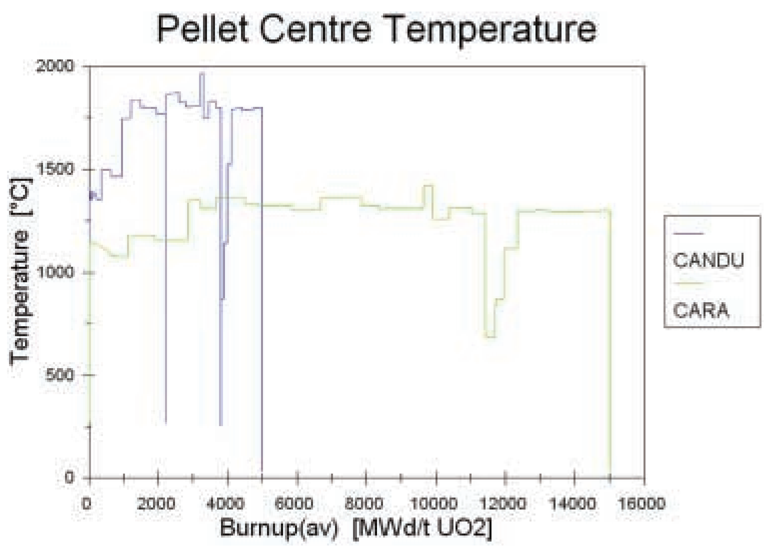


FIG. 161. Averaged temperature at the pellet centre of a 37 element fuel rod and the temperature of the equivalent CARA fuel element.

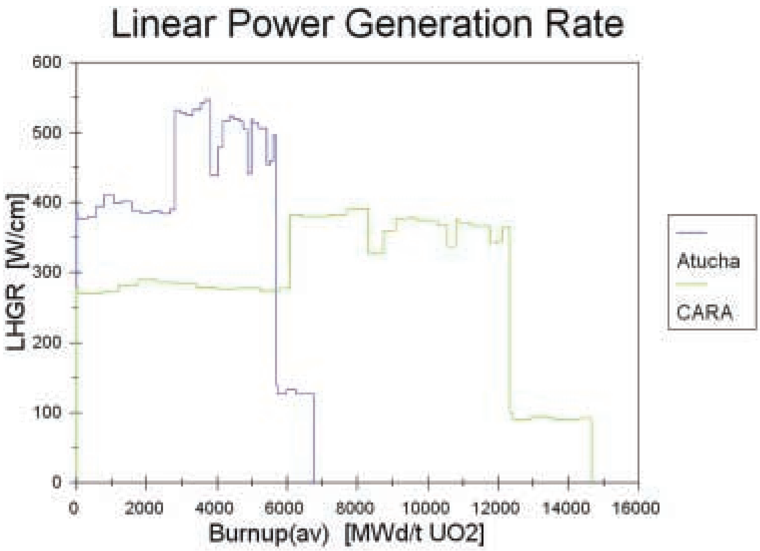


FIG. 162. Local power history of the seventh segment of a fuel rod at the Atucha 1 reactor and the equivalent history of a CARA fuel element at Atucha 1.

The maximum peak centre line temperature for the Atucha fuel is 1850°C (Fig. 163), and the corresponding temperature for the equivalent CARA module is 1340°C, a decrease of 510°C.

The BACO code calculations for the CARA fuel element show lower temperatures, reduced fission gas release, low thermal expansion, and finally good dimensional tolerance, with no restructuring and no central hole. This allows an improvement to be made in the dishing and shoulder of the pellet, and addition of a small plenum. The BACO code was used in the IAEA's Co-ordinated Research Project on Fuel Modelling at Extended Burnup—CRP FUMEX [189]. Its use is supported by experience gained with the Embalse and Atucha reactors, and by comparison with post-irradiation examination measurements and data published in the international literature, including Argentina's own experimental irradiations.

6.3.5.5. CARA development project

The CARA design is of interest to the nuclear power utility in Argentina (Nucleoeléctrica Argentina), and also to the fuel manufacturing company CONUAR. A new project is now being planned with the co-operation of three partners (CNEA,

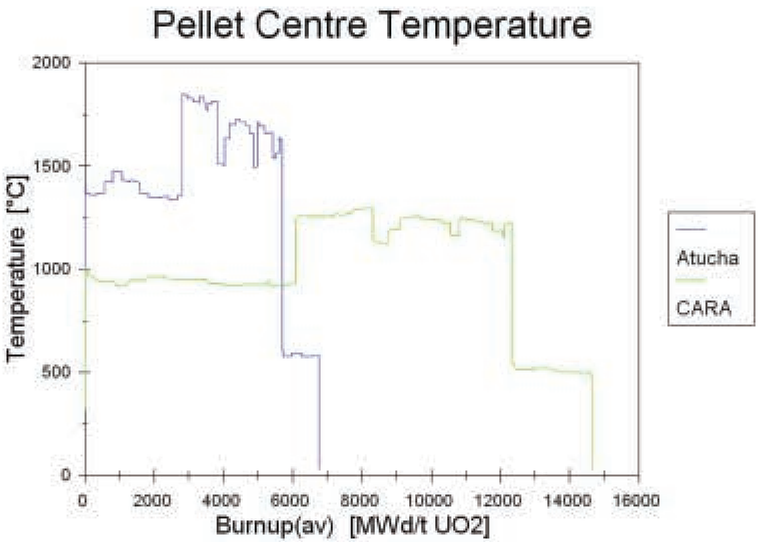


FIG. 163. Local temperature in the seventh segment of a fuel rod at Atucha 1 and the temperature of an equivalent module of a CARA fuel element.

Nucleoeléctrica Argentina, CONUAR) in order to complete the development programme in the shortest possible time and which will result in the commercial production of CARA fuel for both reactors.

The strong economic advantages offered by the use of the new fuel, together with the excellent experience gained with the introduction of SEU at Atucha 1 (Section 6.4.10), give strong incentives for the fastest possible commercialization of the CARA fuel.

The current CARA project, including an Atucha and Embalse demonstration irradiation and post-irradiation examination is an ambitious four year programme.

Currently, the design and independent verification analysis have included mechanical, neutronic, hydraulic and thermohydraulic computer code calculations (including subchannel models), using tools already validated in the SEU project.

In the case of the Atucha fuel channel assembly, four CARA fuel bundle options are available: five CARA bundle options are compatible with the CANDU 6 reactor, with minor changes made to the end plates to permit their use in both reactors. These changes to end plate design do not affect the cost reduction due to standardization to a single bundle type for both reactors because the end plate welding is the last step in bundle fabrication.

Two demonstration CARA fuel assemblies have been built by CONUAR and are shown in Figs 164 and 165. All CNEA's development centres are now involved in the CARA project, together with CONUAR.

6.3.5.6. Summary

A feasibility study on an advanced SEU fuel assembly, compatible with both CANDU 6 and Atucha reactors, has been successfully completed. The CARA fuel assembly comprises 52 elements, each of equal diameter. The project is well advanced and strong commitments have been given by the utility and the fuel manufacturing company. The purpose of the present project is to develop the CARA fuel assembly in the shortest time possible, culminating in its commercial production.

6.4. SEU AND RECYCLED URANIUM

6.4.1. Introduction

Its capability to use natural uranium fuel distinguishes the HWR from other commercial reactor types. Natural uranium fuel can enhance the security and

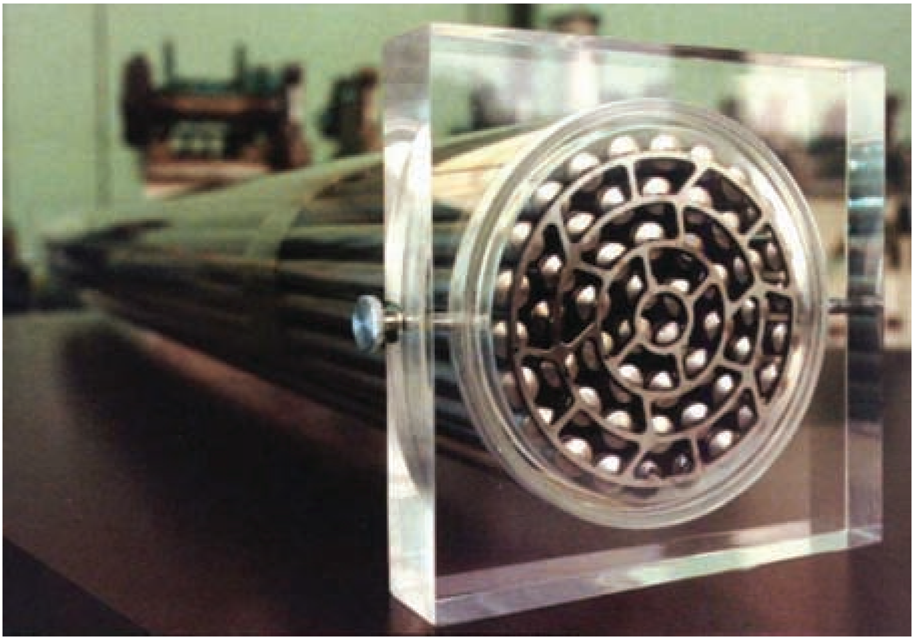


FIG. 164. End view of CARA bundle.

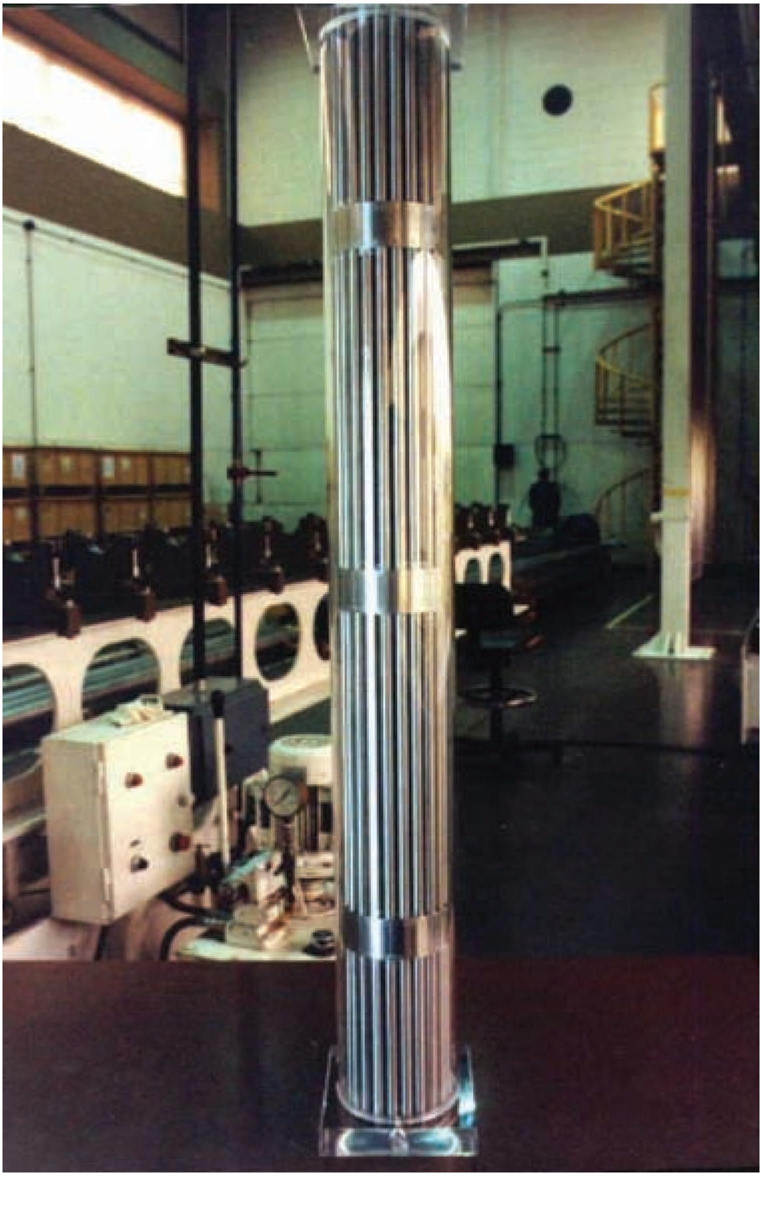


FIG. 165. The CARA fuel bundle.

independence of fuel supply of those countries for which these are important fuel cycle or energy drivers. Its use can also help localize the fuel supply, thereby eliminating reliance on foreign sources of enrichment. It must be understood, however, that excellent neutron economy in the HWR enables the use of natural uranium fuel, and is not a result of its use. Excellent neutron economy (as measured, for instance, by the amount of energy derived from the mined uranium) is even further improved through the use of SEU fuel in the HWR. Furthermore, enrichment is readily available on the open market, from a wide variety of sources.

The use of SEU (or recycled uranium from reprocessing spent PWR fuel) in the HWR offers many benefits, and in many of the countries operating HWRs, it is anticipated that enrichment will be introduced within the next ten years.

The inherent differences in the neutronics, and the low fabrication cost of HWR fuel, mean that the optimal enrichment for HWRs is much lower than for PWRs, between ~0.9% and 1.2%, corresponding to burnups of between ~14 MW·d/kg HE and ~21 MW·d/kg HE. This is in contrast to LWR fuel, where the much higher fabrication cost drives the optimal enrichment to as high a level as can be achieved. This pushes LWR fuel technology towards life limiting fuel performance limits. In the HWR, most of the benefits of SEU can be realized at an enrichment level of near 0.9%. This represents a small incremental step, technically, in the evolution of HWR fuel cycles.

While enrichments of around 0.9% can likely be successfully introduced using the existing 37 element fuel bundle, it is likely that SEU will be incorporated into an advanced bundle design, such as CANFLEX or CARA, which would provide greater confidence in maintaining the excellent fuel performance achieved with natural uranium fuel, as well as other operational benefits, such as improved thermohydraulic performance. A likely fuel cycle scenario in some countries would be the introduction of CANFLEX with natural uranium fuel in order to take advantage of higher thermohydraulic margins. Once experience and confidence have been gained with the new carrier, SEU (or recycled uranium) would then be introduced using CANFLEX.

6.4.2. Benefits of enrichment in the HWR

The use (in the HWR) of SEU with enrichment levels of between 0.9% and 1.2% may very well be a ‘compelling’ product, providing many benefits. This section summarizes some of these benefits.

The already high level of uranium utilization (e.g. the electrical energy derived from the mined uranium) is increased by 32% and 36% compared with natural uranium fuel, for enrichment levels of 0.9% and 1.2% respectively (Table XVII). As lower enrichment plant tails become economic through the use of advanced enrichment technologies (such as atomic vapour laser isotope separation), the improvements in uranium utilization with SEU will become even larger: 43% for 0.9%

SEU, and 56% for 1.2% SEU with 0.1% enrichment plant tails, relative to natural uranium. Lower enrichment plant tails push the optimal enrichment level higher.

The inverse of uranium utilization is uranium consumption: relative to a PWR, natural uranium requirements in an HWR are ~28% lower with natural uranium fuel, and 47% lower with 1.2% SEU (i.e. natural uranium requirements are nearly half). The reduction in mined uranium requirements also has environmental benefits at the front end of the cycle, which will become even more important in the coming decades as cheaper, higher grade uranium resources are depleted, requiring the mining of greater volumes of lower grade ores.

Enrichments of 0.9% SEU and 1.2% SEU would increase the fuel burnup by a factor of ~2 and ~3 respectively, relative to that of natural uranium fuel. This corresponds to a reduction by a factor of 2 or 3 in the quantity of spent fuel to be stored, and eventually disposed of. The lower volume of spent fuel is offset somewhat by the higher decay heat generation, but overall, there is a reduction in back end costs (both interim storage and disposal), with spent fuel disposal costs reduced by as much as 30% relative to those of natural uranium (see Section 6.4.7).

It is expected that SEU usage will also reduce front end fuel cycle costs, with even greater savings being made if the enriched material is recycled uranium (see Section 6.4.6). These costs will be sensitive to local conditions, such as the cost of conversion of UF_6 to UO_2, and the SEU fabrication cost, which will both be dependent on throughput. While fuel cycle costs are currently only a small fraction of the total unit energy cost of HWRs, it is anticipated that in the future there will be greater utility interest in exploiting any opportunity for reducing operation, maintenance and administration costs, in order to improve competitiveness in an increasingly deregulated electricity supply industry. Low enrichments will reduce criticality concerns and limitations in the fuel fabrication process, since enrichments of around 0.9% are below the threshold at which criticality considerations result in restrictions and complications in fuel fabrication, fuel handling, and fresh or spent fuel storage.

In the case of operating reactors that have surplus heat removal capacity, or in which this can be provided in a cost effective manner during a planned outage (such as during retubing), SEU can be used to uprate the reactor power, without increasing the limits on maximum bundle or channel power, by flattening the channel power distribution across the core. Power uprating can provide a large economic benefit to operating plants and enhance the case for plant life extension.

Alternatively, enrichment could be used to flatten both the channel and bundle power distributions in the core without increasing reactor power, and using CANFLEX, for example, as the carrier for the SEU, peak linear element ratings could be reduced to below 40 kW/m, resulting in virtually no fission gas being released during normal operation, and increased operational and safety margins.

An important application of SEU in HWRs is in the reduction of specific capital costs in new designs. This is a critical direction to follow if nuclear power is to compete with independent power producers which are bringing new electrical capacity on-line through the use of technologies that have low capital cost, but which have high fuel cost (such as combined cycle gas turbines). There are a number of ways in which SEU could be used in new HWR designs to reduce specific capital costs [190–192]. One avenue is the use of enrichment to increase the reactor power by flattening the channel power distribution across the core but without increasing maximum bundle or channel powers (which can also be done in operating reactors that have sufficient excess heat removal capability). This is a more cost effective way of uprating power than that of adding more fuel channels, and would be achieved by increasing the channel power in the peripheral channels (trading off some of the burnup potential of SEU for higher power in the peripheral channels, which increases the leakage from the core).

This power uprating through flattening of the channel power distribution across the core can be achieved using a single fuel type, or with two or more fuel types (enrichments) across the core [192]. Using a single fuel enrichment, the flattening is achieved through differential burnup across the core, reducing the burnup (and dwell time) of the fuel in the outer part of the core relative to the unflattened reference case, in order to increase the power in that region. In the SEU fuelled CANDU 9 reactor, for example, the use of 0.9% SEU to flatten the channel power distribution in the core, without increasing bundle or channel power limits, results in ~1100 MW(e) from a 480 channel, Darlington size core, nominally rated at 935 MW(e). Global differential enrichment, using two or more fuels having different enrichments, with the higher enrichment located in the outer part of the core, provides an even greater opportunity for flattening the channel power distribution. One example of this two region core is discussed in Section 6.5.4.3.

When enriched fuel is used in an advanced bundle, such as CANFLEX, maximum bundle and channel powers can be increased while maintaining current operating and safety margins. By employing natural uranium fuel, the CANFLEX bundle power can be increased by as much as 10% while maintaining the current LOCA safety margins with 37 element fuel [193]. Combining channel power flattening with higher bundle and channel power limits, enables reactor power to be increased by more than 20% for a given size core, which will result in a significant reduction in specific capital costs.

Another avenue for reducing specific capital costs through the use of SEU lies in the reduction of the heavy water inventory. This can be most easily accomplished by reducing the size of the radial reflector that surrounds the core (and/or eliminating the radial ‘notch’ in the calandria).

More substantial reductions in capital cost may be possible through more aggressive design changes, enabled through the use of enrichment. One option is to reduce significantly the lattice pitch, and to use light water coolant, thereby further reducing the heavy water inventory. Void reactivity could be significantly reduced without the use of neutron absorbers in the fuel, allowing for simplifications to be made to the PHTS, and for a reduction in the requirements of the safety system. A high degree of passive safety could be incorporated into such a design. With the appropriate choice of enrichment level and lattice pitch, the neutron economy would be even better than that obtained for natural uranium in the current CANDU design. Preliminary studies indicate a potential reduction in capital costs of more than 20% relative to the CANDU plants currently under construction.

Enrichment could be used to optimize new HWR designs in other ways: to increase the pressure tube thickness, thereby extending pressure tube lifetime, or to upgrade the PHTS conditions, thereby achieving higher thermodynamic efficiency.

The use of SEU also offers greater flexibility in fuel bundle design, providing the means to gain specific operating benefits, or to achieve reductions in capital cost. For example, reactivity coefficients can be tailored to meet specific objectives. The ‘low void reactivity fuel’ concept provides a means of optimizing the value of void reactivity and burnup [194, 195]. In this concept, a neutron absorber is employed in the central elements of the bundle, with enriched fuel (SEU or MOX) located in the outer rings. By independently varying the concentration of poison, and the level of enrichment, the desired value of void reactivity and burnup can be achieved. Hence, void reactivity can be considered as a design variable in new plants. Reduced, or even negative void reactivity might offer opportunities for capital cost reduction in new plants by reducing the number of coolant loops, eliminating the need for interlacing the feeders, or by allowing an increase in the maximum bundle power before safety limits are encountered. These benefits would have to evaluated against the higher fuel cost. AECL has conducted an extensive qualification of low void reactivity fuel designs, including thermohydraulic measurements of critical heat flux in freon, reactor physics measurements of void reactivity and fine flux distributions through the bundle in the ZED-2 zero power reactor, and irradiation of prototype bundles in the NRU reactor.

Reactor operational considerations are easily met by enrichments that will lie in the range of interest over the next decade (0.9–1.2% SEU), with no changes to the reactor (Section 6.4.4). The HWR’s on-power refuelling offers flexibility in fuel management that facilitates the use of SEU and other advanced fuel cycles. This flexibility extends from the equilibrium core where, for example, different fuel management strategies could be used to accommodate different levels of enrichment, to the transition from one fuel type (e.g. natural uranium) to another (e.g. SEU).

In summary, SEU offers compelling advantages to both operating and new HWRs. It is anticipated that SEU will be introduced into operating plants, and will be included in the reference design of new plants, within the next decade. It is likely that in this time period, a reactor optimized for the use of SEU would still be able to revert back to the use of natural uranium fuel if desired (although at some penalty).

6.4.3. Fuel design and performance experience

To facilitate the achievement of extended burnups of interest in the near term (two to three times burnups achieved with natural uranium), AECL and KAERI have developed the CANFLEX bundle (Section 6.3.2), ICN (Pitesti) in Romania has developed the SEU 43 bundle design (Section 6.3.4), and Argentina has developed the CARA bundle (Section 6.3.5). These designs all feature greater bundle subdivision (more elements in all cases, and two element sizes in the case of CANFLEX and Romania's SEU 43 bundle). This reduces the peak linear element rating, as well as peak and average fuel temperatures, hence lowering fission gas release. Other minor changes to the pellet or element design can be included to accommodate higher burnup, as summarized in Table XXX. These include:

- Optimization of the chamfer shape, pellet length:diameter ratio and pellet dish. The chamfer shape at the ends of the pellets influences the size of the interpellet and circumferential sheath ridges, as does the length:diameter ratio. The chamfer, as well as the dishes at the pellet ends, also provide space to accommodate fission gas release.
- Optimization of radial and axial clearances; provision of plenum voidage at the ends of the element to provide volume to accommodate fission gas release.
- Optimization of end cap shape to avoid stresses in that region.
- Optimization of cladding thickness for extended burnup.
- Provision of natural uranium fuel pellets at the pellet stack ends to reduce end flux peaking.
- Use of pore formers in the pellets to enhance dimensional stability.
- Use of alternative appendage attachment (such as resistance brazing) to reduce the extent of the heat affected zone.
- Provision of increased thickness of CANLUB, or alternate CANLUB coating (to enhance protection against stress corrosion cracking failures at extended burnup).
- Use of large grain pellets to reduce fission gas release into the free inventory.

The ongoing SEU fuel development programme at AECL and elsewhere involves the irradiation of elements that incorporate design improvements to enhance extended burnup fuel performance. Other design changes have been tested in the past which would achieve even higher burnups over the longer term and would significantly reduce gas release by lowering fuel temperatures, for example, graphite discs between pellets to provide a radial heat conduction path from the fuel to the sheath, and/or annular fuel [196–198].

AECL's experience with SEU and natural uranium (UO_2) fuel, irradiated to extended burnup in AECL research reactors and CANDU power reactors, is summarized in Tables XXXII and XXXIII, respectively.

TABLE XXXII. HIGH BURNUP NATURAL AND SEU UO_2 FUEL IRRADIATED IN RESEARCH REACTORS

Experiment	Bundle or element	Outer element linear power (kW/m)	Outer element burnup (MW·d/kg HE)
BDL-406:			
	AAH	39	17.5
	GC	44	18.4
	GE	44	17.5
	GF	44	37.5
	XY	36	20.4
	ZR	41	22.9
	AAM	41	16.7
	ZS	39	29.2
NPD-56	10 bundles	28–34	29.2–33.4
BDL-416	AAW	72	26.3
NRU-229	JC	66	29.2
BDL-400	PY	68	16.3
BDL-401	HR	55	17.9
BDL-413:			
	ZY	69	20.0
	ZX	72	21.7
DME-190:			
	4 elements	55–65	1 ≥ 18.8
		55–65	3 ≥ 20.8
DME-191:			
	8 elements	55–65	3 ≥ 20.8
		55–65	5 ≥ 20.8
DME-195	3 elements	55–65	3 ≥ 20.8

TABLE XXXIII. DATA FOR SOME OF THE HIGH BURNUP NATURAL UO_2 FUEL FROM CANDU POWER REACTORS

Bundle	Outer element maximum power (kW/m)	Outer element burnup (MW·d/kg HE)
F04857C	55	23.8
J24518C	20	19.2
J24546C	50	31.7
J24533C	50	32.1
J03311W	58	22.5
J98315C	57	24.2
J98324C	56	25.0

The experimental irradiations undertaken in research reactors associated with the initial development of CANDU fuel all involved enriched uranium (in order to achieve sufficient ratings in the NRX and NRU reactors). Some 66 bundles have been irradiated to high burnup (maximum 45 MW·d/kg HE) at high power, and information from this was supplemented by data from the irradiation of 173 single elements. Table XXXII lists the bundles and elements with burnups higher than 17 MW·d/kg HE that are most relevant to the SEU fuel cycle. This experience has been gained with 25 bundles and 15 elements. Ten of the bundles (BDL-406) are natural UO_2 fuelled Bruce production bundles irradiated as fillers in the NRU U-1 and U-2 loops when an insufficient number of experimental bundles were available. As the performance of the natural UO_2 is analogous to that of SEU the results can be used to augment the statistics for extended burnup operation. As expected, the data do not indicate any anomalies in performance and irradiation behaviour compared with natural uranium.

Over 3000 CANDU bundles have been irradiated to above average burnup in power reactors, with a few irradiated to the maximum of 30 MW·d/kg HE [199, 200]. About 150 bundles have experienced burnups above 16.5 MW·d/kg HE, mainly in Ontario Power Generation reactors. Examination of some of this fuel (Table XXXIII) has yielded important information on the performance of 37 element bundles with respect to extended burnup [201, 202].

The extensive experience gained with both research and power reactors demonstrates that the current 37 element bundle design will operate successfully up to about 19 MW·d/kg HE, at power and burnup levels representative of the 0.9% SEU/recycled uranium fuel cycles [203]. The lower linear element ratings achieved with CANFLEX fuel will further increase confidence in good fuel performance at extended burnup, and most importantly, provide additional operating benefits, such as improved thermohydraulic performance.

6.4.4. Fuel management with SEU in an HWR

On-power refuelling in the HWR offers flexibility in fuel management that facilitates the use of SEU and other advanced fuel cycles. This flexibility extends from the equilibrium core where, for example, different fuel management strategies could be used to accommodate different levels of enrichment, to the transition from one fuel type (such as natural uranium) to another (such as SEU). Fuel management strategies formulated to date have focused on the equilibrium core; fewer studies having been made of the transition to SEU (whether from a natural uranium reference core, or from a fresh SEU core).

6.4.4.1. Fuel management with 0.9% SEU

Fuel burnup for SEU with an enrichment of around 0.9% is nearly double that of natural uranium fuel, and spent fuel quantities would correspondingly be reduced by half. Fuel management is particularly simple, and several such studies have been reported [163, 204, 205]. A regular two or four bundle shift, bidirectional fuelling scheme results in excellent axial power profiles being achieved, both in the vicinity of the adjuster rods, and away from them. Figure 166 compares typical axial power profiles for 0.9% SEU fuel and natural uranium fuel, in a central channel in the vicinity of the adjuster rods (normalized to 6500 kW channel power). The axial power distribution for SEU peaks around bundle position 4, thereafter decreasing towards the outlet end of the channel. This power distribution, which is skewed towards the inlet end of the channel, will have higher critical channel power than the cosine shaped power distribution associated with natural uranium fuel. The inlet skewed axial power profile will also result in reduced pressure tube creep (which is a function of temperature), and fast neutron fluence (which is proportional to bundle power). The power profile is excellent from the perspective of fuel performance, as little, if any, power boosting would be expected at extended burnup. The axial power distribution also has a lower peak than for natural uranium fuel (for the same channel power).

The choice between a two or four bundle shift fuelling scheme would involve local station considerations. A two bundle shift involves twice as many channel visits as either a four bundle shift scheme with 0.9% SEU, or an eight bundle shift scheme with natural uranium. Although the SEU refuelling rate in terms of bundles per day is the same for both two and four bundle shifts, it is about half that of natural uranium fuel. Thus, the number of channel closures and fuel movements in the channels is increased with a two bundle shift fuelling scheme. However, a two bundle shift could be done in such a way that the total time spent refuelling would not exceed that of an eight bundle shift scheme with natural uranium fuel. A two bundle shift scheme reduces both the power ripple occurring during refuelling (the local increase in power as a result of the addition of reactivity during refuelling) and the peak bundle powers.

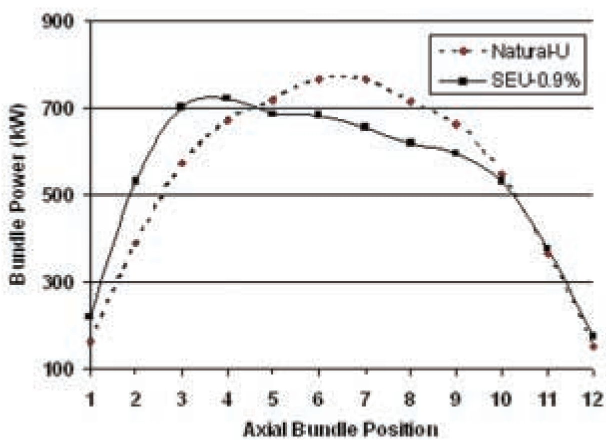


FIG. 166. Typical axial power profiles for central channel for natural uranium and 0.9% SEU.

One means of reducing channel power ripples during refuelling in both two and four bundle shift schemes, and hence increasing the margin to critical channel power, is to add a very small amount of burnable poison (such as gadolinium) to the fuel. The poison will burn out within the first few days of operation, and hence would have a negligible impact on burnup and uranium utilization, but would reduce the bundle and channel power ripples occurring during refuelling. This technique could be used with any level of enrichment (even with natural uranium), where a reduction in ripple is desired.

In the case of 0.9% SEU fuel, the worth of the reactivity devices in the control and safety systems is adequate to meet the requirements. In fact, the reactivity worth (with 0.9% SEU) of the adjuster rods exceeds the current requirements for xenon override time following a reactor shutdown.

Typical results of detailed fuel management simulations with 0.9% SEU are discussed in Section 6.4.5.3 in the context of recycled uranium.

6.4.4.2. Fuel management with 1.2% SEU

Using SEU fuel with an enrichment of around 1.2%, a bidirectional, two bundle shift refuelling scheme results in an excellent axial power profile in the absence of adjuster rods (e.g. throughout the core in a reactor in which the adjuster rods have been removed, or in the outer ~216 channels of a CANDU 6 reactor away from the adjuster rods, which are located in the centre of the core). The power increases gradually from the inlet end of the channel, peaking in bundle positions 4 and 5, and then decreases along the channel to the outlet end. With this fuel management scheme, only relatively fresh fuel (which is resilient to power ramps) will experience significant power boosting, and the asymmetrical power distribution peak at the inlet end of the channel will improve the critical channel power.

However, in the centre of the core, the adjuster rods depress the flux and power in the middle of the channel, resulting in an asymmetrical ‘double hump’ power distribution. During fuelling, some bundles will experience a power boost at extended burnup, and this is undesirable from the viewpoint of avoiding fuel failure due to stress corrosion cracking. The effect of the adjuster rods on the axial power distribution is illustrated in Fig. 167, which shows the power of a central channel (L9), normalized to 6500 kW, in a two bundle shift fuelling scheme using 1.2% SEU. Two situations are represented: one in which the adjusters are present in the core, and the other in which the adjusters have been removed from the core. The asymmetrical double hump in the axial power profile due to the presence of the adjuster rods is clearly evident. For the sake of comparison, the axial power distribution of the same channel with natural uranium fuel is also shown.

In the case of HWRs, the flexibility provided by on-line refuelling in shaping the axial power distribution, particularly in the presence of adjuster rods, is

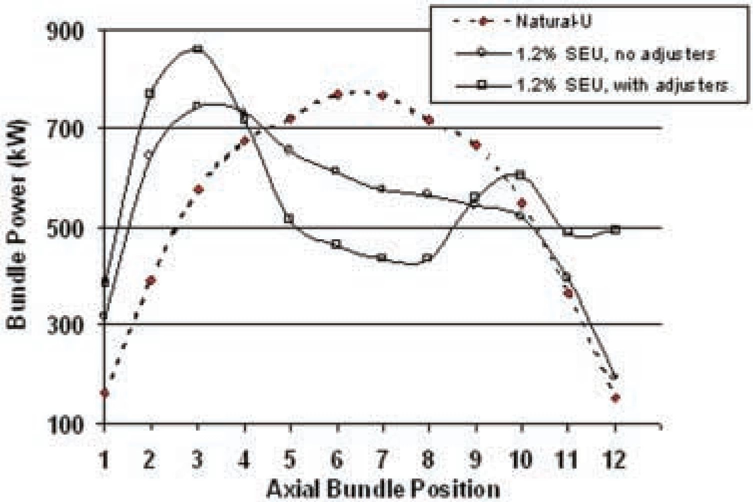


FIG. 167. Typical time average axial power profiles of the central channel for natural uranium and 1.2% SEU in core with and without adjuster rods.

demonstrated by the development of many refuelling strategies designed to accommodate higher enrichments in current reactors.

One strategy is termed 'two pass' refuelling [190]. In this scheme, a bundle passes through the reactor twice: one bundle that is discharged after its first pass through the channel is loaded with a fresh bundle and inserted into a different channel for its second pass. Therefore, each refuelling operation produces one bundle which is discharged and one bundle which is to be reinserted into the core.

Another strategy that has been devised to accommodate high enrichments in the form of SEU or MOX is the 'chequerboard' refuelling scheme, combined with 'multistage' shifting [206]. In the chequerboard refuelling scheme, channel pairs are refuelled using the same fuelling scheme (refuelled in opposite directions), while adjacent channel pairs are fuelled using a different fuelling scheme. For example, in a group of four channels, one pair of adjacent channels might be fuelled using a six bundle shift (each channel in the pair being refuelled from opposite ends), while an adjacent channel pair might be refuelled using a two bundle shift. Thus, in chequerboard fuelling, different bundle shift schemes are mixed systematically in a region or regions of a reactor with the specific intention of achieving the desired axial flux (hence power) distribution. The chequerboard region may cover either a part of, or the entire reactor core. Multistage shifting refers to unequal intervals between fuelling machine visits. For example, a six bundle shift might be carried out in three stages, with short time intervals elapsing between each stage. Reference [206] describes a chequerboard fuelling scheme for accommodating 1.7% SEU (30 MW·d/kg HE), in which the six bundle shift scheme in the chequerboard is carried out in three stages of two bundles each. The time interval between the stages is 30 full power days. Following three such refuellings in a channel, the fuel is irradiated for another 540 full power days. The two bundle shift in the surrounding channels is also carried out at irregular intervals: three shifts are carried out with 120 full power days between shifts; the fuel is then irradiated for another 360 full power days. The combination of chequerboard fuelling and multistage shifting provides a great deal of flexibility in shaping the axial power distribution of a wide range of enrichments and adjuster rod loadings.

Another strategy that provides a great deal of flexibility in accommodating the adjuster rods is 'axial shuffling' [207, 208]. Axial shuffling involves removing some or all of the bundles from the channel into the magazine of the fuelling machine, rearranging them in pairs, discharging those bundles that have reached their target burnup, and reinserting the other fuel bundle pairs back into the same channel in a new but predetermined pattern, along with fresh fuel. This method is most effective in new reactors designed for single ended fuelling. However, preliminary assessments indicate that axial shuffling is technical feasible in all HWRs, although changes to the fuel handling equipment may be required in some cases.

The ease of application, and the number of bundles which can be removed from the channel, depend on several factors. For instance, the fuelling machine magazine capacity must be sufficient to hold all the bundles discharged from the channel, the fresh bundles which are to be inserted into the channel, and other components, such as the channel closure and shield plugs. The coolant flow must be sufficient to discharge all the bundles.

References [207, 208] discuss both time average and time dependent refuelling simulations performed for a CANDU 6 core in which a simple two bundle shift, bidirectional refuelling scheme was used for the outermost 248 channels, and an axial shuffling scheme for the innermost 132 channels, which are located near the adjuster rods. The shuffling scheme is bidirectional.

If bundles are numbered from 1 to 12 starting at the 'refuelling' end of the channel, then bundles are first loaded into positions 1 and 2. In subsequent fuelling operations, the bundles in positions 1 and 2 are shifted as follows:

$1 \rightarrow 5 \rightarrow 3 \rightarrow 9 \rightarrow 7 \rightarrow 11 \rightarrow$ discharge,
$2 \rightarrow 6 \rightarrow 4 \rightarrow 10 \rightarrow 8 \rightarrow 12 \rightarrow$ discharge.

This scheme results in excellent axial power profiles, which in fact are very similar to those achieved in the outer channels, away from the adjuster rods, fuelled with a simple, bidirectional, two bundle shift fuelling scheme. No significant power boosting is predicted, except for relatively fresh fuel.

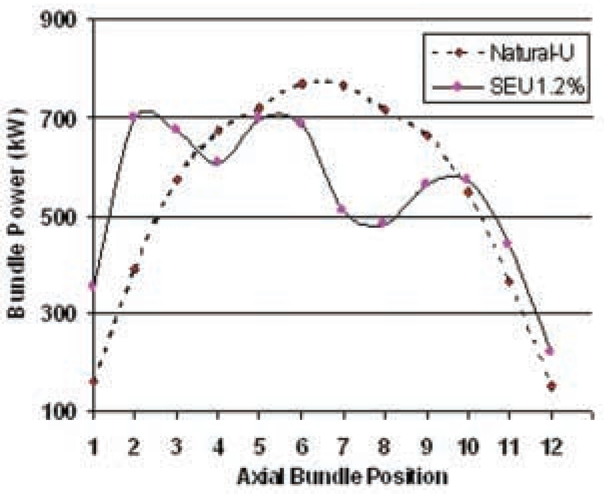


FIG. 168. Typical time average axial power profiles for the central channel using natural uranium and 1.2% SEU with axial shuffling.

Figure 168 compares the time average axial power profile along a central channel using the axial shuffling scheme mentioned above (with 1.2% SEU) with that of natural uranium fuel (both normalized to the same total channel power of 6500 kW). Figures 169 and 170 show the corresponding bundle power histories that would be experienced by bundles loaded initially either in axial position 1 or 2,

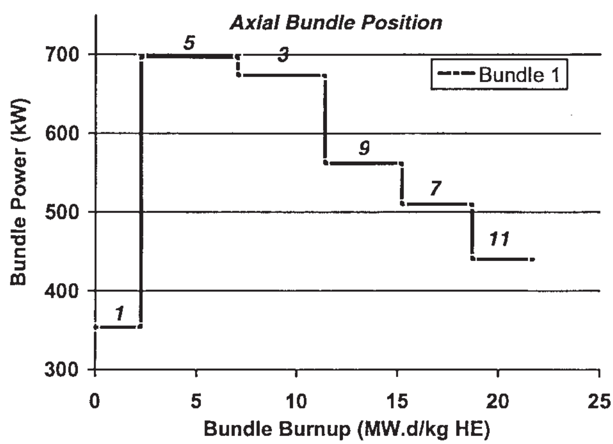


FIG. 169. Time average bundle power history (axial shuffling bundle 1).

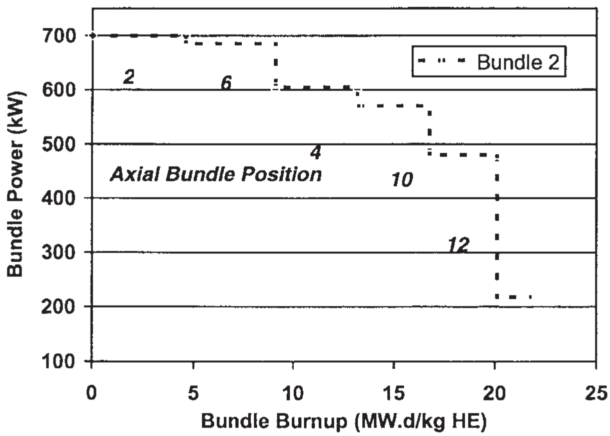


FIG. 170. Time average bundle power history (axial shuffling bundle 2).

respectively, for the same channel using axial shuffling. These are estimates based on the time average power distribution, and do not show the effects of refuelling. The flattening of the axial power profile compared with that due to natural uranium fuel, and the excellent power history are evident.

The channel power peaking factor is 1.11; similar to that of natural uranium. The channel power peaking factor is a measure of the refuelling ripple, and is defined here as being the largest channel overpower (ratio of channel power at a point in time to the time average power of that channel) taken over channels having an instantaneous power of at least 90% of the peak channel power at that point in time. The channel power peaking factor is an important parameter in the method currently used for calibrating the neutron overpower detectors which are used to determine operating margin. A short, time dependent refuelling simulation was performed. Peak channel and bundle powers were very low: 6480 kW and 780 kW respectively (lower than with natural uranium fuel). The linear element ratings for almost all fuel in the core were below 40 kW/m with the CANFLEX fuel bundle. Sizeable power boosting only occurred with the introduction of relatively fresh fuel, as expected.

In summary, in the case of 1.2% SEU, a simple, bidirectional, two bundle shift refuelling scheme results in excellent axial power histories and core characteristics, either in a core from which the adjuster rods have been removed, or away from the influence of the adjuster rods in the centre of the core. For operating HWRs having the adjuster rods present, other fuel management schemes have been devised that result in good axial power histories and core characteristics. Axial shuffling is one such fuel management strategy that results in a great deal of flexibility in shaping the axial power distribution.

6.4.4.3. Mixed natural uranium/SEU core

An interesting variation that can be considered is the use of a mixed natural uranium/SEU core. One such configuration would have SEU in the perimeter of the core, away from the adjuster rods, and natural uranium in the centre of the core. This is one option for increasing the power in the peripheral channels, without switching to a full core of SEU, and without unduly increasing the fuelling machine usage and sacrificing the burnup, as would be the case if radial flattening were pursued using natural uranium fuel only. This could also be an intermediate configuration in a transition from an all natural uranium core, to an all SEU core.

In one CANDU 6 study, a ring of 216 channels containing 1.2% SEU surrounded the central 164 channels, which contained natural uranium fuel [209]. Confining the SEU to the outer channels, away from the central adjuster rod region, allowed the use of a simple two bundle shift fuelling scheme for the SEU fuelled channels, and the usual eight bundle shift fuelling scheme for the central natural uranium fuelled channels. The study considered both the equilibrium core characteristics, and the transition from natural uranium to the mixed core, which was modelled in a 1000 d, time dependent refuelling simulation.

The mixed core in equilibrium had many advantages over the natural uranium core. The SEU was used to flatten the radial channel power distribution in the core, with the peak channel power in the mixed core 7% below that in the natural uranium reference core. While this study made no attempt to maximize the uprating potential, with feeders appropriately sized and with sufficient heat removal capacity, a 7% power uprating would be possible in a new reactor, or in a reactor that was appropriately refurbished. Powers in the outer channels of the mixed core were 20–35% greater than those in the corresponding channels of the natural uranium core. In the centre of the core, the channel powers in the mixed core were about 9% lower than in the natural uranium core. The quantity of spent fuel produced was reduced by 40%; core average burnup increased by ~70%. The refuelling rate in terms of channels/d remained about the same as with natural uranium, but the refuelling rate in bundles per day decreased from 18.5 to 11. The mixed core reduced the annual uranium requirement by 24%, with significant savings in fuel cycle costs. The axial bundle power profile with the SEU fuel peaks towards the inlet end of the channel, and decreases along the length of the channel. In a CANDU 6 reactor, with coolant flow and refuelling occurring in the same direction, this power shape will improve the critical channel power.

The transition from the equilibrium natural uranium core to the mixed core was particularly straightforward, since the adjuster rods did not distort the axial power profile in the SEU fuelled channels. In the SEU fuelled channels, the axial power profile evolved smoothly, with no significant power boosting at extended burnups. One issue that would have to be addressed during the transition would be normalization of the neutron overpower trip detectors, since the reference channel power distribution changes (becomes flatter) over time. On the basis of the power histories, fuel performance is expected to be good, both during the transition, and in equilibrium. Although the study was undertaken using the 37 element bundle, the use of CANFLEX would result in similar bundle and channel powers, and peak ratings that are 20% lower than for the 37 element bundle, providing even greater confidence in excellent fuel performance.

Since the first use of enriched fuel in an HWR will likely be the use of SEU at enrichment levels of around 0.9%, a mixed core with 0.9% SEU in the peripheral channels is a particularly attractive option for introducing SEU into a HWR, while uprating the reactor power. An uprating of up to 12% can be achieved in this manner [190].

6.4.4.4. Optimizing reactivity device locations for SEU in new reactors

As noted in Section 6.4.4.2, fuel management with 1.2% SEU is straightforward in a core without adjuster rods (a simple, bidirectional, two bundle shift). If adjuster rods are not needed for xenon override or for provision of reactivity shim, they can be removed.

Another option available to new reactors involves the relocation of reactivity devices, particularly adjuster rods, to facilitate the use of a variety of fuel types. One configuration that has been examined comprises four rows of adjuster rods (as opposed to the three rows used in current designs), with the outer rows moved closer to the channel ends [210]. The outer adjuster rods would be removed when using natural uranium fuel; with enriched fuel, the number, location and relative strengths of the adjusters would be chosen to optimize the axial power profile, while achieving the required reactivity worths. This configuration was studied for a wide range of fuels: natural uranium, SEU and MOX having a burnup of 21 MW·d/kg HE, and high burnup SEU with a burnup of 31 MW·d/kg HE. Figure 171 shows that the axial power profile for 1.2% SEU in a central channel in a core with repositioned adjuster rods is nearly identical to the axial power distribution without adjuster rods. This would reduce the perturbations in the power distribution resulting from insertion and removal of the adjuster rods. Consequently, load following capability should be improved, and the rise to full power after a shutdown achieved in a shorter time than with natural uranium fuel.

The capability to relocate the adjuster rods is of great advantage in shaping the axial power distributions in the equilibrium core. Maximum bundle and channel powers for both the time average core, and for a time dependent refuelling simulation of the equilibrium core, were comparable to, or lower than, the natural uranium

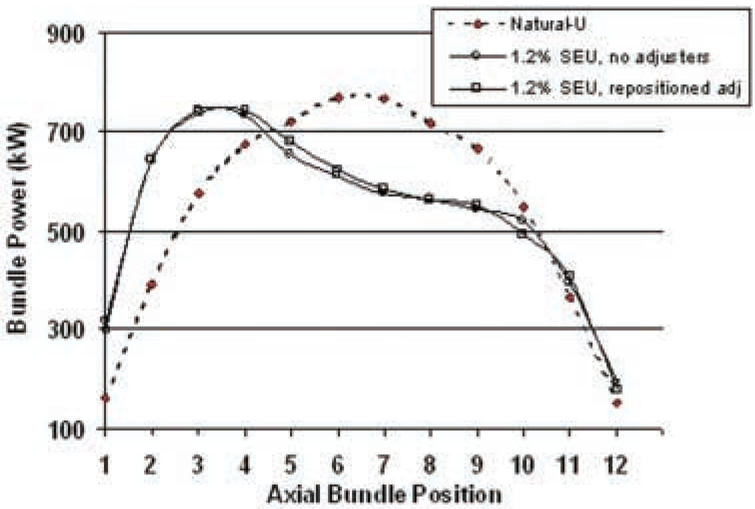


FIG. 171. Typical time average axial power profiles for the central channel using natural uranium and 1.2% SEU in a core without adjusters, and in a core with repositioned adjusters.

reference core with the current configuration of reactivity devices. Although these were 'scoping' studies, they demonstrated significant promise in respect of the relocation of reactivity devices in new designs to enable the adjuster rod configuration to be chosen in order to optimize the axial power profile and other core characteristics for a wide variety of fuel types.

Using this same configuration of reactivity device, an 1100 d, time dependent simulation of the transition from natural uranium to a core with 1.2% SEU was modelled [211]. While it was anticipated that having the capability to relocate the adjuster rods during the transition would help in shaping the axial power profile, this was found to be of little benefit for the strategy employed in the study. Simply replacing natural uranium by 1.2% SEU, two bundles at a time, was not a viable option during the early part of the transition, because the high reactivity SEU was added to low worth flux locations at the channel ends, while higher burnup natural uranium was shifted to the high worth flux locations at the centre of the channels. As a consequence, it was difficult to maintain criticality. It was concluded that a better strategy during the early part of the transition would be either to introduce the SEU more slowly, by alternating SEU fuelling with natural uranium, or to add two or four bundles of natural uranium, followed by two bundles of SEU.

In a new reactor, the use of a mixed core (two different enrichments in different radial regions) together with repositioned reactivity devices presents the greatest potential for power uprating [190].

6.4.4.5. *Transition to SEU: Indian perspective*

There are two ways of introducing SEU into HWRs. The first starts from a natural uranium fuelled core, and converts to SEU by gradually replacing natural uranium fuel bundles by SEU fuel bundles during refuelling. While on-line refuelling of the HWR provides this flexibility, as was noted in Section 6.4.4.4, the fuel management strategy over the transition period must be carefully considered. Alternatively, a new reactor (or a reactor that is returning to power after a maintenance or refurbishment outage) can be started fully loaded with SEU fuel. In this case, fuel management during transition to the equilibrium core is simpler, but the reactivity of the initial core will be high, and has to be suppressed.

In the Indian HWR, with the whole core loaded with 1.2% SEU fuel, k_{eff} is found to be 1.254. The easy way of suppressing this excess reactivity is by adding boron to the moderator. Apart from the fact that this negates the benefit provided by improved uranium utilization of the SEU in the initial core, this approach will give rise to severe power peaking and an increase in void reactivity during a LOCA. The power peaking would be worse than that in a natural uranium core, since the channel flow distribution of the Indian HWR has been designed to match the power distribution in the equilibrium core. As explained in Section 6.5.3 in the context of the TANDEM fuel cycle, as fuel burnup proceeds, the bundle powers become acceptable but the coolant outlet temperatures in the peripheral channels increase beyond the channels' rated values.

An alternative solution involves the use of thoria (ThO_2) bundles to suppress the excess reactivity of a fresh core of SEU. A distribution of thoria bundles that would suppress the initial reactivity (>299 mk) but which would not require derating of the reactor was determined for the Indian HWR. Details of the reactor are given in Table XXXIV, while details of the 19 element fuel cluster are given in Table XIV. The

TABLE XXXIV. PERFORMANCE CHARACTERISTICS OF 1.2% SEU FRESH CORE, USING THORIA BUNDLES FOR REACTIVITY SUPPRESSION

Parameter	Value
k_{eff}	1.040
Total power (MW(th))	655
Maximum channel power (MW)	2.70
Maximum permissible channel power (MW)	2.75
Maximum bundle power (kW)	375
Maximum permissible bundle power (kW)	420
Maximum channel outlet temperature (°C)	296.8
Maximum permissible coolant outlet temperature (°C)	297.0
U-233 produced from thoria (kg)	108

loading of thoria bundles is shown in Fig. 172. Every channel has 10 bundles (numbered 1 to 10). The numbers given in each channel position in Fig. 172 denote the positions of the thora bundles in that channel. For instance, channel T-8 carries the numbers 3 and 7, indicating that in this channel, positions 3 and 7 are occupied by thoria bundles, while the rest are occupied by 1.2% SEU bundles. The shaded

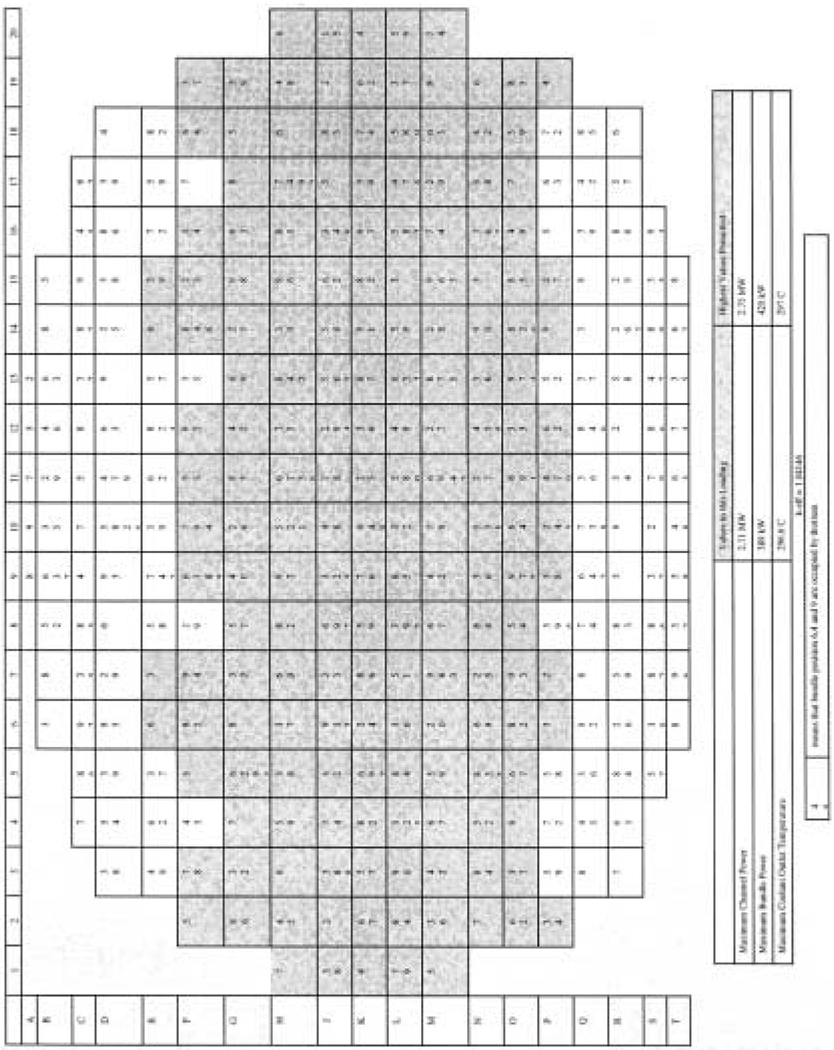


FIG. 172. Indian PHWR showing distribution of thoria bundles in fresh 1.2% SEU core used to suppress initial reactivity.

channels in Fig. 172 represent the high power region of the core. The core contains nearly 600 thoria bundles, i.e. nearly 20% of the core.

When the thoria bundles come out of the core, they contain nearly 100 kg of ^{233}U. Table XXXIV gives the performance characteristics of this thoria loaded core. The channel power, bundle power and channel outlet temperature are within the prescribed limits.

6.4.5. Recycled uranium

6.4.5.1. Introduction

Recycled uranium is one of the products of the conventional chemical reprocessing of spent oxide fuel. In this context, recycled uranium is simply a subset of SEU, as it is acquired on the open market, as is SEU or natural uranium, and its use is not linked to the reprocessing of the spent fuel of the utility acquiring the material. It is considered simply as an alternative source of enrichment (anticipated as being much cheaper) compared with SEU derived from the enrichment of fresh uranium, and can be substituted for SEU of equivalent enrichment. The enrichment level is around ~0.9%, the actual ^{235}U content and composition depending on the initial enrichment and burnup of the fuel from which it was obtained. In general, reactor physics data and fuel performance will be comparable to those recorded for 0.9% SEU.

The use in HWRs of recycled uranium derived from reprocessed spent PWR fuel can be considered as an illustration of HWR/PWR synergy on a global basis.

The cumulative quantity of recycled uranium derived from the reprocessing of European and Japanese spent fuel by 2000 amounted to around 25 000 Mg. This material, which is owned by the utilities and reprocessors, is an alternative fuel source to that of new natural uranium and is suitable for use in LWR and CANDU reactors. Each country and utility will determine its strategy for recycled uranium on the basis of local factors. Theoretically, this 25 000 Mg of recycled uranium would provide sufficient fuel for 500 CANDU 6 reactor-years of operation, since the initial core load of uranium for a CANDU 6 reactor is 85 Mg, and the annual refuelling requirement for a recycled uranium fuel burnup of 13 MW·d/kg HE is around 50 Mg.

Current reprocessing technology produces a recycled uranium product suitable for interim storage pending possible re-enrichment and recycle in LWR reactors. British Nuclear Fuels Limited (BNFL) uses thermal denitration to convert uranyl nitrate liquor to UO_3, the chemical form in which the recycled uranium is stored. Cogéma uses the ADU route to convert uranyl nitrate liquor to U_3O_8. Several processes could be used to convert the recycled uranium from the form used in storage, to a UO_2 powder capable of being sintered. For example, the UO_3 from BNFL's THORP reprocessing plant could be processed to UF_6 and then converted to ceramic grade UO_2 using the integrated dry route process. Alternatively, BNFL has a laboratory scale facility in operation which can convert the uranyl nitrate liquor directly to a ceramic grade UO_2 by a kiln denitration process known as the modified direct route. Cogéma has also used the ADU route to convert directly a batch of uranyl nitrate liquor to UO_2 (recycled uranium) powder that was supplied to AECL in a collaborative programme in the early 1990s. That powder had characteristics akin to those of standard natural uranium (UO_2) powder produced through the ADU route. The conversion route is a key step in determining the cost of the recycled uranium material.

The burnup associated with recycled uranium is about double that of natural uranium, and within the burnup envelope for operating reactors using natural uranium fuel. While no design modifications need be made to either the element or the pellet in order to accommodate recycled uranium fuel, those design enhancements discussed in Section 6.4.3 would apply equally well to recycled uranium fuel and would provide greater confidence in maintaining excellent fuel performance at extended burnup. With the CANFLEX bundle as the carrier, peak linear element ratings would be 20% lower than with the 37 element bundle, which would facilitate the achievement of extended burnup with recycled uranium, as well as providing enhanced thermohydraulic operating margins.

In considering fuel cycle costs attributable to recycled uranium, generally, it is worth noting that recycled uranium is owned by the utility that contracts for reprocessing of the spent oxide fuel. Most countries and/or utilities which adopt a reprocessing strategy do so for strategic energy self-reliance and/or waste management reasons. The uranium and plutonium recovered from reprocessing are often held as ‘low’ or ‘zero’ cost stocks by the utilities. Hence, there is the possibility that recycled uranium will be competitively available on the open market. The potential annual saving to a CANDU utility offered by utilization of recycled uranium is significant, but strongly dependent on the price paid for the recycled uranium powder (which includes the cost of conversion to UO_2) and for fuel fabrication. There would need to be a strong economic incentive for a CANDU utility to use recycled uranium in preference to SEU, because of the higher radiation fields. Preliminary ‘scoping’ studies on the front end of the fuel cycle (excluding back end storage and disposal costs) are discussed in Section 6.4.6.

The use of recycled uranium in CANDU reactors would reduce the quantity of spent fuel per unit of electricity generated by a factor of around two, compared with natural uranium fuel. A decrease in the spent fuel disposal costs of the order of 10% (for 10-year cooled) to 20% (50-year cooled) compared with natural uranium (10-year cooled) would be expected (Section 6.4.7).

6.4.5.2. Radiological considerations for recycled uranium

The radiation fields associated with recycled uranium are higher than those of natural uranium. The isotopic composition and activity of unenriched recycled uranium (UO_2) powders depend on the reactor type, initial enrichment and discharge burnup of the PWR fuel, the time between spent PWR fuel discharge and reprocessing, the route chosen to convert the uranyl nitrate liquor to UO_2, and the delay until fuel fabrication. Recycled uranium typically contains ~1 ppb ^{232}U, which has a half-life of 69.8 years. The daughters in the ^{232}U decay chain are removed during reprocessing but grow during storage. Conversion processes via UF_6 also remove daughter products. The first daughter in the chain is ^{228}Th, which has a half-life of 1.9 years. Since all the other daughters in the chain have much shorter half-lives, including the radiologically important ^{208}Tl and ^{212}Bi, they are all in secular equilibrium with ^{228}Th. Therefore, the ^{228}Th buildup governs the rate of buildup of gamma activity and is an indicator for the variation of the gamma activity over time, relative to the quasi-equilibrium level attained after about 10 years. Recycled uranium also contains ^{234}U, and this contributes to a higher specific alpha activity compared with natural uranium. However, the level is about the same as that in conventional enriched PWR fuel, since the source of the increased ^{234}U comes from the initial enrichment of natural uranium. Recycled uranium also contains trace fission product gamma and beta emitters, and transuranic alpha emitters.

An initial assessment of the health physics aspects of manufacturing and handling recycled uranium as a reactor fuel for CANDU was made in a joint programme between AECL and Cogéma, and more recently in a collaborative programme between BNFL, KAERI and AECL [163, 212]. BNFL has converted recycled uranium from reprocessed spent PWR fuel into 200 kg of UO_2 using the modified direct route. The recycled uranium (UO_2) powder met CANDU specifications, both in terms of its chemical impurity content and its physical characteristics. The powder was granulated and pressed into green pellets, which were sintered under the normal conditions for CANDU fuel. The finished pellets met all the physical and chemical specifications for CANDU fuel.

The conversion took place one year after reprocessing. Activity level measurements made on the finished CANFLEX recycled uranium bundle were ~1.3 times higher than on a natural uranium bundle, when measured at a distance of 30 cm. Consequently, because the total fuel quantity required can be reduced by around 50% by using recycled uranium, the overall dose uptake to the workforce during the fabrication and handling of recycled uranium bundles should be comparable to that presently seen for natural uranium fuel. By reducing the time intervals between spent PWR fuel discharge and reprocessing, and between reprocessing and conversion, fuel fabrication and insertion into the reactor, the dose uptake can be minimized.

During sintering, the release of ^{137}Cs and other volatile fission products from recycled uranium was below detectable levels. Also, AECL had earlier concluded that no significant radiation fields in a commercial fuel fabrication plant would build up as the result of release of ^{137}Cs during sintering, even after decades of production [163].

Studies have been undertaken on the impact on personnel dose incurred during fuel manufacturing operations of the increased specific activity of recycled uranium compared with natural uranium. These studies have shown that this impact can be readily minimized, without incurring a significant cost penalty, to the acceptable levels recognized in modern standards for fuel manufacturing operations. The increase in external activity is unlikely to require any change to current fuel manufacturing or fuel receipt and handling procedures at the reactor. Nonetheless, gamma dose rates for recycled uranium powder will increase linearly for several years after conversion to UO_2, and this will require attention being paid to the time delays between conversion, manufacturing and use in-reactor.

6.4.5.3. Reactor characteristics

The extra isotopes present in recycled uranium have minimal effect on the reactor physics characteristics in an HWR, compared with SEU of the same ^{235}U content. Spent PWR fuel typically contains 0.4 wt% ^{236}U (the range varying from 0.2%–0.7%), and has a strong resonance at 5.5 eV, originating from neutron capture by ^{235}U present in the original PWR fuel. As a result of the softer neutron spectrum in an HWR, the absorption worth of the ^{236}U is an order of magnitude lower in an HWR than in a PWR. Therefore, the main determinant in HWR reactor physics with regard to recycled uranium is the ^{235}U level.

The reactor core characteristics and the benefits of using recycled uranium in an HWR would be much the same as those for 0.9% SEU. AECL and KAERI have performed reactor physics simulations to evaluate the feasibility of CANFLEX recycled uranium fuel being used in a CANDU 6 reactor by taking the isotopic composition of typical recycled uranium (UO_2) produced by the modified direct route [205]. Core follow simulations were performed by AECL for 500 full power days on CANFLEX recycled uranium having 0.96 wt% ^{235}U, using a bidirectional two bundle shift refuelling scheme; KAERI performed 600 full power days core follow simulations on CANFLEX recycled uranium having 0.88 wt% ^{235}U, using a bidirectional, four bundle shift refuelling scheme. Standard computer codes and methods were used for the simulations and analysis. WIMS-AECL, with ENDF/B-V nuclear data library, were used to construct fuel tables for use with a core code, RFSP, which modelled the reactor core [170, 213]. The results of the CANFLEX recycled uranium core follow simulations show that the recycled uranium fuel would be satisfactory in an equilibrium CANDU 6, as summarized in Table XXXV.

An assessment was made using the results of the AECL refuelling simulation of the probability of stress corrosion cracking occurring through power boosting. The CANFLEX recycled uranium elements do not come close to approaching the stress corrosion cracking element power threshold, and none of the linear element powers were above 44 kW/m.

TABLE XXXV. SUMMARY OF CORE CHARACTERISTICS FOR RECYCLED URANIUM

Parameter	AECL 500 full power day simulations (two bundle shift refuelling)*	KAERI 600 full power day simulations (four bundle shift refuelling)	Licence limit
Recycled uranium enrichment (wt% U-235)	0.96	0.88	
Maximum channel power (MW)	7.021	7.088	7.3
Maximum bundle power (kW)	857	883	935
Average zone fills for individual zone controllers	0.3–0.7; did not reach the minimum or maximum limits	0.46–0.55; did not reach the minimum or maximum limits	
Average discharge burnup (MW·d/kg HE)	14	12.4	

* See Ref. [205].

Following the KAERI preliminary results, a CANDU fuel element performance analysis code, ELESTRES, predicted that the internal pressure of the outer CANFLEX recycled uranium elements during normal power operation was below 2.5 MPa, which is lower (by a factor of ~2) than that of the outer elements of the 37 element natural uranium fuel bundle [128]. The maximum fuel stack length of the outer and inner CANFLEX recycled uranium elements increased by 0.46% through thermal expansion, which is equivalent to a reduction of less than 0.2 mm in the axial gap between the fuel stack and the end cap. In addition, a preliminary safety assessment of a CANDU 6 demonstrated that, for all the shorter half-life isotopes, the gap (or 'free') inventory with CANFLEX recycled uranium fuel is 5–10 times smaller than that of 37 element natural uranium fuel, and the total inventory with recycled uranium fuel is very similar to that of 37 element natural uranium fuel.

In the case of the longer half-life isotopes such as ^{137}Cs, the gap inventory with CANFLEX recycled uranium is very similar to that with 37 element natural uranium fuel, but the total inventory with CANFLEX recycled uranium fuel is about double that of 37 element natural uranium fuel, owing to the higher burnup.

6.4.6. Front end fuelling costs with SEU/recycled uranium

Fuel cycle costs are well established for CANDU natural uranium fuel and for LWR fuel (although there exists a wide range in LWR fuel manufacturing costs). Typical CANDU natural uranium fuel cycle costs are about half those of a PWR [130].

The cost of natural uranium CANDU fuel comprises few components: the cost of uranium, the cost of converting U_3O_8 to ceramic grade UO_2, and the cost of fuel fabrication. The cost of SEU CANDU fuel comprises the cost of uranium, the cost of converting U_3O_8 to UF_6 for enrichment, the cost of enrichment, the cost of converting the enriched UF_6 to ceramic grade UO_2, and the cost of SEU fuel fabrication.

The cost of some of these steps is well known (e.g. enrichment). In the case of LWR fuel, the cost of converting enriched UF_6 to ceramic grade enriched UO_2 is included in the cost of fuel manufacturing, hence, there is some uncertainty in this cost when applied to CANDU fuel. There are several routes for obtaining ceramic grade, enriched UO_2 powder for HWR fuel, including the direct purchase of ceramic grade, enriched UO_2 powder or pellets directly from any one of a number of LWR fuel manufacturers around the world. The cost of SEU CANDU fuel fabrication is sensitive to several commercial factors: the volume of SEU fuel fabrication (alternatively, the larger volume of natural uranium fuel that it displaces in the fuel manufacturing facility); the actual level of SEU enrichment (which will determine the extent of criticality constraints in the fuel fabrication process); and how SEU and natural uranium fuel fabrication are integrated in the manufacturing plant (separate facility, separate lines, continuous or batch processes). It should be noted with regard to this last point, that the Canadian fuel fabricators currently handle several different fuel types in their facilities (37 element fuel for CANDU 6 reactors, 37 element 'long' and 'standard length' fuel bundles for Bruce and Darlington reactors, 28 element fuel for the Pickering station, as well as natural and depleted uranium fuels).

As discussed in Section 6.4.5, an important potential source of enrichment for CANDU is recycled uranium from the reprocessing of spent PWR fuel. The fuel cycle cost for recycled uranium will include the cost of the material itself, the cost of conversion to ceramic grade UO_2 (from either the U_3O_8 or UO_3 form in which it is stored, or directly from the uranyl nitrate solution), and the fuel fabrication cost. The conversion route selected will be a key determinant of the recycled uranium cost.

Given the uncertainty in some of the cost components for SEU CANDU fuel, a parametric survey over indicative cost ranges can provide insights. The following assumptions were made:

- *Natural uranium and SEU CANDU fuel*. Cost of uranium: US \$25/kg U and US \$80/kg U (in the form of U_3O_8, representing current spot market, and the longer term trend).
- *Recycled uranium CANDU fuel*. Cost of 'uranium' (in the form of U_3O_8, UO_3, or uranyl nitrate): 0%, 50%, or 100% of the cost of 'natural' uranium (which is either US \$25/kg U or US \$80/kg U).
- *Natural uranium CANDU fuel*. Cost of fuel fabrication (including the cost of conversion from U_3O_8 to UO_2): US \$50/kg U.

- *SEU CANDU fuel*. Cost of conversion from U_3O_8 to UF_6: US $3/kg U.
- *SEU CANDU fuel*. Cost of enrichment: US $80/kg SWU and US $110/kg SWU (current spot market and United States Enrichment Corporation prices, respectively).
- *SEU and recycled uranium CANDU fuel*. Cost of fuel fabrication (including the cost of conversion to ceramic grade UO_2, from either UF_6 (for SEU) or U_3O_8 (for UO_3), or directly from uranyl nitrate (for recycled uranium): US $75/kg U and US $100/kg U (e.g. 50% and 100% higher than the cost of conversion plus fabrication for natural uranium CANDU fuel).

Spent fuel storage and disposal costs were not included, nor the costs of transportation. SEU fuel disposal costs are discussed in the following section. The possible effect of the actual value of enrichment on fuel fabrication costs was also not considered (e.g. the fuel fabrication cost was assumed to be independent of the value of enrichment). In the case of SEU, an enrichment tails of 0.25% was used.

Five CANDU fuel types plus the natural U reference were considered (with nominal burnups determined from a reactor physics lattice code, and not from a full-core simulation: actual burnups depend on the detail of the reactor (size, presence of adjuster rods):

- Natural uranium reference (7.5 MW·d/kg HE),
- 0.9% SEU (14 MW·d/kg HE),
- 1.2% SEU (21 MW·d/kg HE),
- 1.5% SEU (28 MW·d/kg HE),
- 1.7% SEU (32 MW·d/kg HE),
- Recycled uranium (13 MW·d/kg HE).

Figure 173 is indicative, and results from using a very simple fuel cycle model (accounting only for the front end). Some of the broad conclusions drawn are as follows:

- There is generally little difference in the fuel cycle costs of CANDU SEU fuel with enrichment levels between 1.2% and 1.7%. A more detailed assessment would likely favour lower enrichments in this range (e.g. reduced criticality limitations); 1.2% SEU is close to being optimal for CANDU.
- SEU is the most economic route, relative to natural uranium fuel, for high uranium, low enrichment, and low SEU conversion plus fabrication costs. Fuel cycle costs are 15% lower with 0.9% SEU, and 20% lower with 1.2% SEU.
- The 0.9% SEU fuel cycle costs are particularly sensitive to the costs of conversion plus fabrication, and there will be a strong incentive to minimize this cost differential with natural uranium fuel.

- The effect of enrichment (SWU) costs on fuel cycle costs increases with increasing enrichment, and is quite minimal for 0.9% SEU.
- There is a real opportunity to achieve significant fuel cycle cost reductions with recycled uranium, depending, of course, on the cost of material, conversion and fuel fabrication. Fuel cycle cost savings which represent a factor of two improvement compared with natural uranium fuel seem to be attainable.

This survey indicates the potential for making significant fuel cycle cost savings, particularly with recycled uranium fuel. Until the use of enrichment is commercially established for HWR fuel, uncertainties will remain in the cost parameters.

6.4.7. Disposal of spent SEU fuel

The disposal cost is an important consideration in assessing the economic feasibility of any advanced fuel cycle. The use of SEU in HWRs will reduce the volume of spent fuel, with benefits for both storage and transportation of the spent

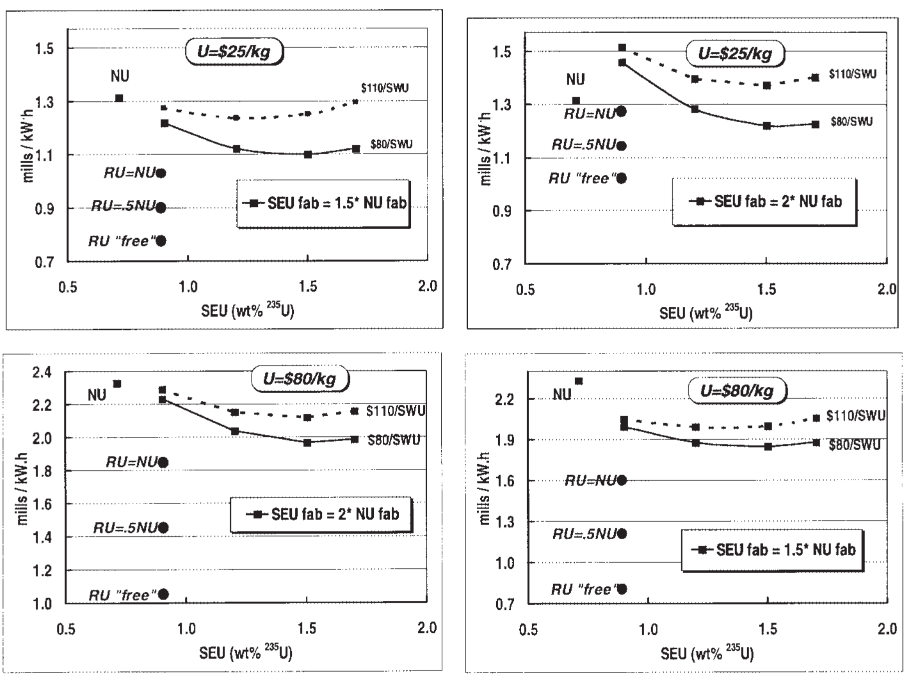


FIG. 173. SEU/recycled uranium fuel cycle costs.

fuel to the final repository. Preliminary cost analyses have been conducted to determine the impact of enrichment on disposal costs [214]. Specific spent CANDU fuels examined were:

- Natural uranium reference (8 MW·d/kg HE),
- 0.9% SEU (14 MW·d/kg HE) (this will also be representative of recycled uranium fuel),
- 1.2% SEU (21 MW·d/kg HE),
- 1.5% SEU (28 MW·d/kg HE),
- 1.7% SEU (32 MW·d/kg HE),

The disposal costs of CANDU DUPIC fuel were also assessed, and this case is discussed in Section 6.5.2.5.

A natural uranium burnup of 8 MW·d/kg HE is typical of several Canadian CANDU units, but is somewhat higher than the nominal burnup of a CANDU 6 reactor.

In deriving the spent fuel disposal costs, the Canadian concept for deep geological disposal in plutonic rock was used as the reference base for design and costing. This concept envisages in-floor borehole emplacement of titanium shell disposal containers having a 72 fuel bundle capacity in a 1000 m deep repository [146]. The interrelated key variations that were applied in the conceptual designs for the spent advanced CANDU fuel repositories were:

- Post-irradiation age of the spent fuel before disposal (storage time),
- Number of spent fuel bundles in a disposal container (Mg HE/container),
- Number and spacing of containers across the width of a disposal room and their spacing along the length of the room.

Following the costing of the conceptual disposal designs by Canada, the NEA undertook a study on the costs of high level waste and spent fuel disposal in geological repositories, bringing estimates to a common basis [116]. Costs were normalized to the amount of electricity generated (million US $/TW·h in terms of 1991 US $). Currency differences and inflation rates were taken into account. Figure 174 shows a comparison of the estimated cost of spent fuel disposal prepared by several countries, including Canada, and plotted against the electricity generated from the corresponding fuel. The costs included those attributable to a waste/spent fuel packaging plant and disposal repository design, construction, operation, decommissioning and closure. Excluded from consideration were the costs related to site screening, site selection and evaluation, spent fuel storage and transportation, R&D and financing, because these requirements varied considerably between countries. The exclusion of these costs allowed the comparison to be made on a more

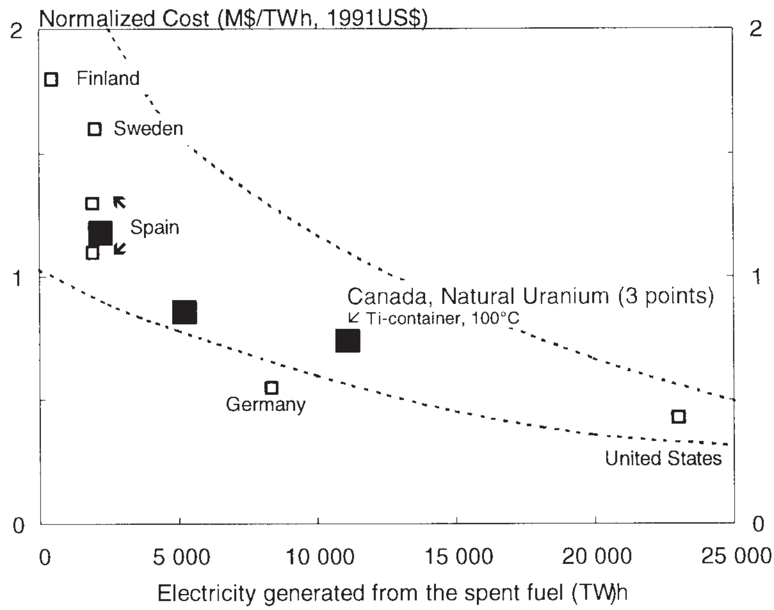


FIG. 174. Unit cost of spent fuel disposal as a function of total electricity generated (after NEA 1993 [116]*). CANDU natural uranium scaled costs shown for three repository sizes.*

common technical basis. It should be noted, however, that in cases where a longer storage time elapses before disposal, additional (small) storage costs will be incurred which are not included in this study. The costs are not discounted.

The trend displayed in Fig. 174 shows that disposal costs per unit of electrical energy decrease with increasing repository capacity (in terms of total electricity produced by the fuel). However, the significant differences in unit disposal costs, as shown by the wide bounds, reflect the large variation in the details of the technical design requirements in each of the countries (e.g. the spent fuel emplacement method, nature of backfilling, type and post-irradiation age of spent fuel, temperature limits, disposal container design).

Figure 174 also shows the variation of the unit disposal cost (million US $/TW·h in terms of 1991 US $) for spent CANDU natural uranium fuel calculated for three repository capacities (i.e. for three different values of total electricity produced from corresponding quantities of spent fuel). The unit disposal costs for each capacity are composed of three components:

- Unit operating costs (constant with the quantity of electricity or corresponding spent fuel produced),

- Unit construction and decommissioning variable costs (constant with the quantity of electricity or corresponding waste produced),
- Unit construction and decommissioning fixed costs (inversely proportional to the quantity of electricity or corresponding waste produced).

If the titanium container were to be replaced with a more expensive copper container, the unit disposal cost would increase by less than 10%.

Figure 175 and Table XXXVI show that the unit disposal costs for 1.2% SEU with 10-years' cooling before disposal, are lower than for natural uranium fuel. Less spent fuel needs to be packaged into fewer spent fuel disposal containers, even though the repository size is increased. However, in the 1.7% SEU case, the unit disposal cost is greater than that in the reference CANDU natural uranium case because the cost saving resulting from having fewer containers is exceeded by the cost of providing increased container spacing and satisfying the size requirements for the repository. All these cases are based on 10-years' post-irradiation storage.

If the storage period for spent SEU CANDU fuel is increased from 10 years to 50 years to allow time for more thermal decay to occur, the unit disposal costs are significantly reduced (Fig. 176). This is also clearly shown in Fig. 177 for the full

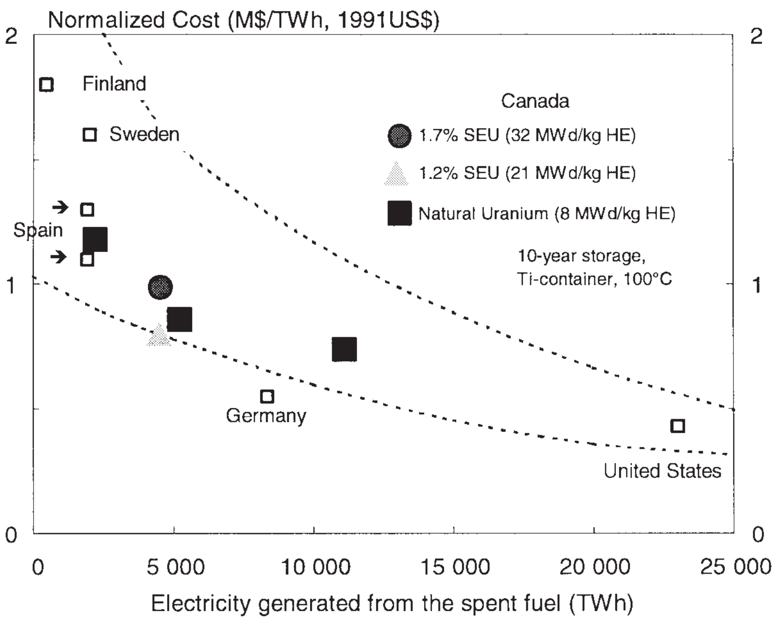


FIG. 175. Canadian CANDU unit cost of spent fuel disposal: Effect of change in fuel enrichment and burnup (SEU 10 year post-irradiation storage).

TABLE XXXVI. UNIT DISPOSAL COSTS FOR ALL CASES

Case study	Container		Fuel age (a)	Unit disposal cost case (million US $/TW·h in terms of 1991 US $)		
	Material	Temperature (°C)		2000 TW·h	4321 TW·h	10 000 TW·h
CANDU (natural uranium):						
	Ti	100	10	1.20	0.95	0.77
	Ti	90	10	1.36	1.01	0.84
	Cu	90	10	1.45	1.10	0.93
CANDU (SEU):						
0.9 wt%	Ti	100	10	1.19	0.84	0.67
	Ti	100	50	1.11	0.76	0.59
1.2 wt%	Ti	100	10	1.16	0.81	0.64
	Ti	100	50	1.03	0.68	0.51
1.5 wt%	Ti	100	10	1.19	0.84	0.67
	Ti	100	50	0.99	0.64	0.47
1.7 wt%	Ti	100	10	1.36	1.01	0.84
	Ti	100	50	0.97	0.63	0.46

range of potential spent CANDU SEU fuels for an electrical generation case of 4321 TW·h (generated by four million CANDU natural uranium fuel bundles).

Disposal cost savings of up to 15%, relative to natural uranium, can be achieved for 1.2% SEU fuel after 10-years' post-irradiation storage. The cost savings can be further improved, to a maximum of about 30%, if post-irradiation storage is extended to 50 years. The number of disposal containers, and thus packaging costs, do not change over the increased storage period. Only the repository costs decrease (i.e. repository size decreases) with the increase in storage time. It should be noted that post-irradiation storage costs are not included in any of the analyses. Increasing the cooling time for natural uranium fuel does not mean that the borehole spacing can be reduced in the reference repository design, since this spacing is already limited by geotechnical considerations.

In the case of 0.9% SEU, Table XXXVI shows that disposal costs are reduced by about 10% (10 year stored) to 20% (50 year stored) compared with natural uranium.

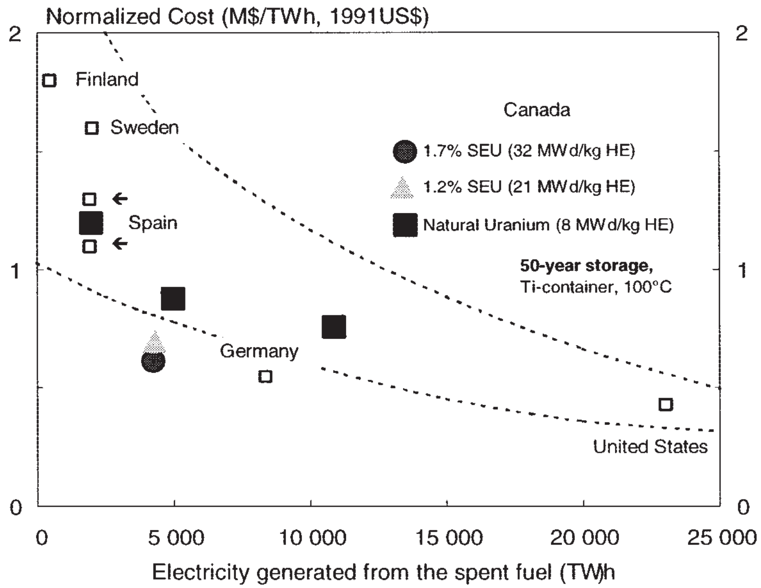


FIG. 176. Canadian CANDU unit cost of spent fuel disposal: Effect of change in fuel enrichment and burnup (SEU 50 year post-irradiation storage).

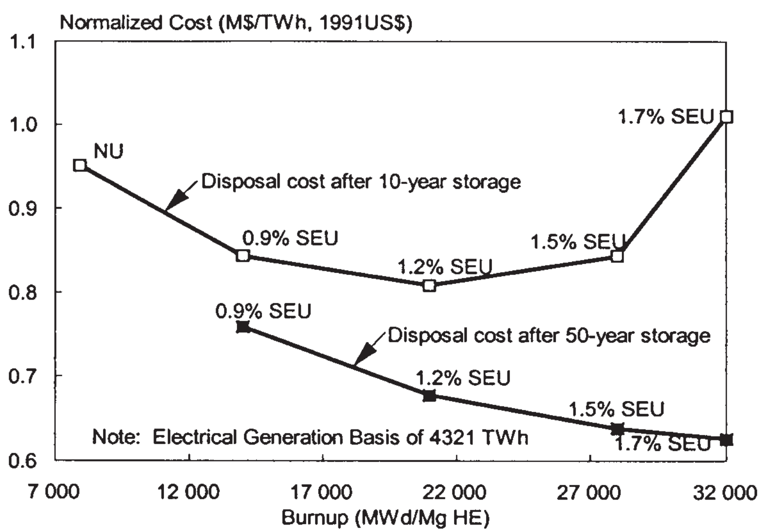


FIG. 177. Canadian CANDU unit cost of spent fuel disposal: Comparison of spent CANDU natural uranium and SEU fuels as a function of burnup (enrichment) and post-irradiation storage (based on a total electrical generation of 4321 TW·h and on the NEA (1993) publication [116]*).*

6.4.8. Non-proliferation and safeguards considerations with SEU/recycled uranium

There are no significant non-proliferation or safeguards aspects that need be taken into consideration with the use of SEU or recycled uranium in an HWR, in contrast to natural uranium (considering recycled uranium as being available on the open market, rather than linked to a utility's decision to reprocess).

6.4.9. Time-frame for deployment of, and national perspectives on, SEU/recycled uranium

The use of SEU/recycled uranium represents near term fuel cycle options for HWRs; ones that will certainly become available within 5–10 years. There are no feasibility issues associated with their use, and no R&D is required. In certain countries, the use of recycled uranium or SEU may follow the full-core implementation of CANFLEX with natural uranium.

6.4.10. Converting the Atucha 1 pressure vessel reactor to SEU fuel

6.4.10.1. Introduction

Argentina has two reactors currently in operation: the 648 MW(e) Embalse CANDU 6 reactor using the reference 0.5 m long, 37 element fuel bundles loaded into 380 horizontal pressure tube channels, each containing 12 fuel bundles; and the 360 MW(e) Atucha 1 pressure vessel reactor, equipped with 252 vertical fuel channels, each one accommodating one 6.5 m long, 37 element fuel assembly. Both reactors use on-power refuelling with natural uranium fuel in the form of UO_2 pellets in 37 element fuel bundles. The construction of a third reactor, Atucha 2, is about 80% complete. This is a Siemens design, pressure vessel reactor (capacity 745 MW(e)) with 451 vertical fuel channels (similar to Atucha 1).

The natural uranium fuel exit burnup in Atucha 1 (6 MW·d/kg HE) is lower than the burnup at Embalse (7.1 MW·d/kg HE), and the manufacturing cost is much higher for Atucha 1 fuel than for Embalse, owing mainly to the length and complexity of Atucha 1 fuel assemblies (see Fig. 178). As a result, the fuelling cost at Atucha 1 has a high impact on the total operating cost, and various activities have been in progress for several years in Argentina to lower the cost of Atucha 1 fuel.

One development aimed at lowering the fuel cost at Atucha 1 is the conversion to using 0.85% SEU, which is being done without major modification being made to the plant hardware and without perturbation to plant availability [215].

Meanwhile, developments are under way that could, in the future, lead to a possible introduction of SEU fuel in the Embalse CANDU plant. These developments

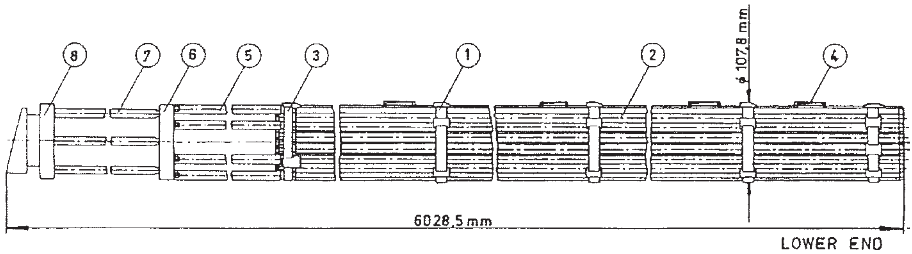


FIG. 178. General view of the Atucha 1 fuel assembly.

include, in particular, the CARA project aimed at developing a common fuel assembly for both operating reactors in Argentina, which is discussed in Section 6.3.5.

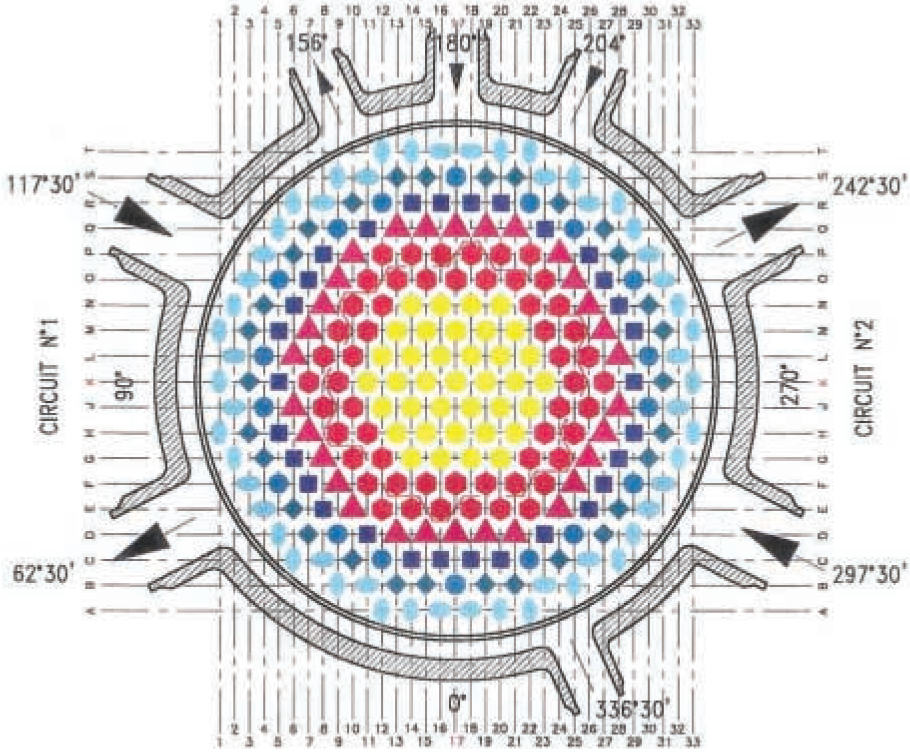
6.4.10.2. Main features of the Atucha 1 core

The Atucha 1 core contains 252 vertical fuel channels arranged on a 27.2 cm triangular pitch lattice, inside the moderator tank. This tank lies within the reactor pressure vessel, while the annulus between them forms the coolant downcomer similar to that of a PWR. Essentially, each channel is an approximately 2 mm thick Zircaloy 4 tube separating the inner D_2O coolant (at 280°C average temperature) from the outer D_2O moderator (at 190°C average temperature). Both fluid systems are connected outside the core and kept at the same pressure (~11.65 MPa); the channel tubes ensure that the cooler moderator is separated from the coolant in the core region.

In order to establish a fairly uniform exit coolant temperature among the 252 fuel channels at the upper plenum, these channels are grouped into eight different 'hydraulic zones', each zone having a different coolant flow selected to match approximately the radial power profile. The central hydraulic zone, corresponding to the highest rated fuel channels, has no flow restrictors, while the seven outer zones have seven different types of throttle located at the entrance to each channel (Fig. 179).

Up to the beginning of 1995, only fresh natural uranium fuel assemblies were loaded and shuffled following a three position in-core path. Such a scheme defined three 'burnup regions', which, from the core centre to the periphery, were as follows:

- *Burnup region 1*: The central burnup region with intermediate burnup fuel assemblies.



TIPO DE ESTRAN_ GULAMIENTO	DIAMETRO EQUIVALENTE mm.	N°DE E.C.	CAUDAL min. kg/seg.(1)	CAUDAL min. kg/seg.(2)	PLANO M20
1	18,6	30	8,54	4,14	405–1
2	21,0	24	11,86	5,43	405–1
3	25,0	30	15,40	7,30	405–1
4	28,4	18	19,14	9,04	405–1
5	34,2	30	23,74	10,96	405–2
6	39,0	30	28,67	13,43	405–3
7	44,0	54	29,41	13,84	405–4
8		37	32,90	16,08	

FIG. 179. View of hydraulic zones in the Atucha 1 core.

- *Burnup region 2*: The intermediate burnup region — entrance region (fresh fuel assembly loading) with low burnup fuel assemblies.
- *Burnup region 3*: Outer burnup region — exit region (fuel assembly unloading) with high burnup fuel assemblies.

Every fuel assembly followed the path:

Fresh fuel assembly → burnup region 2 (core load) → burnup region 1 → burnup region 3 (core unload) → fuel pool.

6.4.10.3. Organizational aspects

The project to convert Atucha 1 to 0.85% SEU is led by the utility NASA, through its engineering division, with the participation of Atucha 1 staff.

Other sectors of NASA, as well as CNEA, provide technical support and execute engineering tasks in specific areas. The manufacture of Atucha 1 SEU fuel assemblies is carried out (as with all other Argentine reactor fuel) by the local manufacturer CONUAR.

The project was formally divided into phases, the first of which comprised:

Phase 1: Initial SEU fuel assembly irradiation and examinations. The SEU fuel assembly load was limited to a maximum of 12 fuel assemblies (out of a total of 252 fuel assemblies in the core). There was minimum perturbation of core parameters.

On the basis of both engineering studies and early evaluation of phase 1 results, the subsequent phases were grouped into two, as follows:

Phase 2: Massive SEU irradiation. The SEU load was limited to a maximum of 60 fuel assemblies and later extended to 99 out of a total of 252 fuel assemblies in the core.

Phase 3: Approach to the full SEU core.

At the onset of the project in 1993, more phases were foreseen, reflecting a limited knowledge of how SEU fuel would work in this HWR. Main concerns centred on:

- Atucha 1 fuel assembly behaviour under extended burnup (changing exit burnup from 6–11 MW·d/kg HE);
- Reactor control system behaviour under much higher positive reactivity insertion during fuel assembly loading;

- Adequate selection of core channels for fresh SEU fuel assembly loading, taking into account fairly narrow margins for channel power and local powers, even for fresh natural uranium fuel assemblies;
- Adequate selection of locations for irradiated SEU fuel assemblies, taking into account channel power and local power limits, as well as criteria to prevent pellet–cladding interaction;
- Overall core behaviour, core parameter changes and core response for postulated accidents and abnormal transients, with the presence of SEU in the core.

During 1993 and 1994, the engineering tasks associated with the first phase were carried out, and the first SEU fuel assembly was loaded in January 1995.

The main results of phase 1 were as follows [216]:

- All SEU fuel assemblies were loaded at six selected channels where exit coolant temperature thermocouple measurements were taken but which were located at the outermost channels of a high coolant flow hydraulic zone in order to have the maximum possible channel power margin.
- The in-core SEU fuel assembly movement scheme was the same as that used for natural uranium fuel assemblies, though with much higher residence times in each burnup region and higher transition burnups.
- Excellent behaviour was observed for Atucha 1 fuel assemblies containing SEU fuel. Most of the 18 SEU fuel assemblies irradiated in this phase reached exit burnups of ~10 MW·d/kg HE, which approached the target exit burnup for the final core (11 MW·d/kg HE).
- A special study was undertaken to provide practical and more flexible pellet–cladding interaction criteria and a new specification for in-core fuel handling involving SEU fuel assemblies was devised and applied [216].
- A comprehensive experimental verification of predicted/calculated parameters was performed. This included verification of regulation rod bank movements and coolant exit temperature at fresh SEU fuel assembly channels during SEU fuel assembly loading.

The most significant aspect of phase 2, owing to its complexity, was the determination of an appropriate in-core refuelling scheme involving a rapid replacement of fresh natural uranium fuel assemblies by SEU fuel assemblies. This was accomplished through the use of two schemes for SEU fuel assembly movement and a single scheme for natural uranium fuel assembly movement within the core.

The phase 2 irradiation programme was initiated just after the planned annual outage of the plant in mid-1996. Originally, it was planned to have a maximum of

60 SEU fuel assemblies loaded in the core by the end 1997, but subsequently it was decided to extend phase 2 to incorporate a total load of 99 SEU fuel assemblies.

The results from the phase 2 irradiation programme have been excellent. No SEU fuel assembly failure attributable to the use of SEU fuel has been observed: the fuel assembly failure rate remained very low, as was the case before the initiation of the SEU irradiation programme. The core operation parameters were hardly modified, as was expected, and the safety requirements were met as mentioned below.

The target average exit burnup for the equilibrium full SEU core, namely 11 MW·d/kg HE, had already been achieved for SEU fuel assemblies unloaded during phase 2, while a progressive increase in burnup for natural uranium fuel assemblies was also observed.

The end of phase 2 was reached in August 1998, and phase 3 was initiated immediately after. By the end of 1999, approximately 75% of the core load was SEU [217].

6.4.10.4. Equilibrium SEU core

The main aspects of this future core have already been calculated and analysed. In-core fuel management would be based on a three burnup region scheme, quite similar to that employed for the former, full natural uranium core which used two schemes (Fig. 180). Radial power flattening associated with the use of SEU fuel would bring larger power margins for the two innermost hydraulic zones (those having the highest rated fuel assemblies), and provide more flexibility for fuel assembly shuffling and general reactor operation. The predicted average exit burnup is 11.1 MW·d/kg HE, representing a reduction of ~46% in the number of irradiated fuel assemblies compared with a full natural uranium core. Significant savings are also expected in terms of the utilization of the fuel loading machine, the spent fuel pool capacity requirements, and particularly in the current and levelized fuelling costs, where savings of between 35% and 40% are expected.

6.4.10.5. Safety aspects

Activities in this area encompass the following for each phase of the project, according to the maximum load of SEU fuel for that phase:

- Determination of new reactor physics core parameters;
- Determination of operation systems changes under normal conditions;
- Determination of changes in radioactive inventories and releases, operational exposure and doses to the public through different paths and conditions using conservative assumptions;
- Simulation and analysis of the set of accidents and abnormal transients postulated for the previous full natural uranium core;

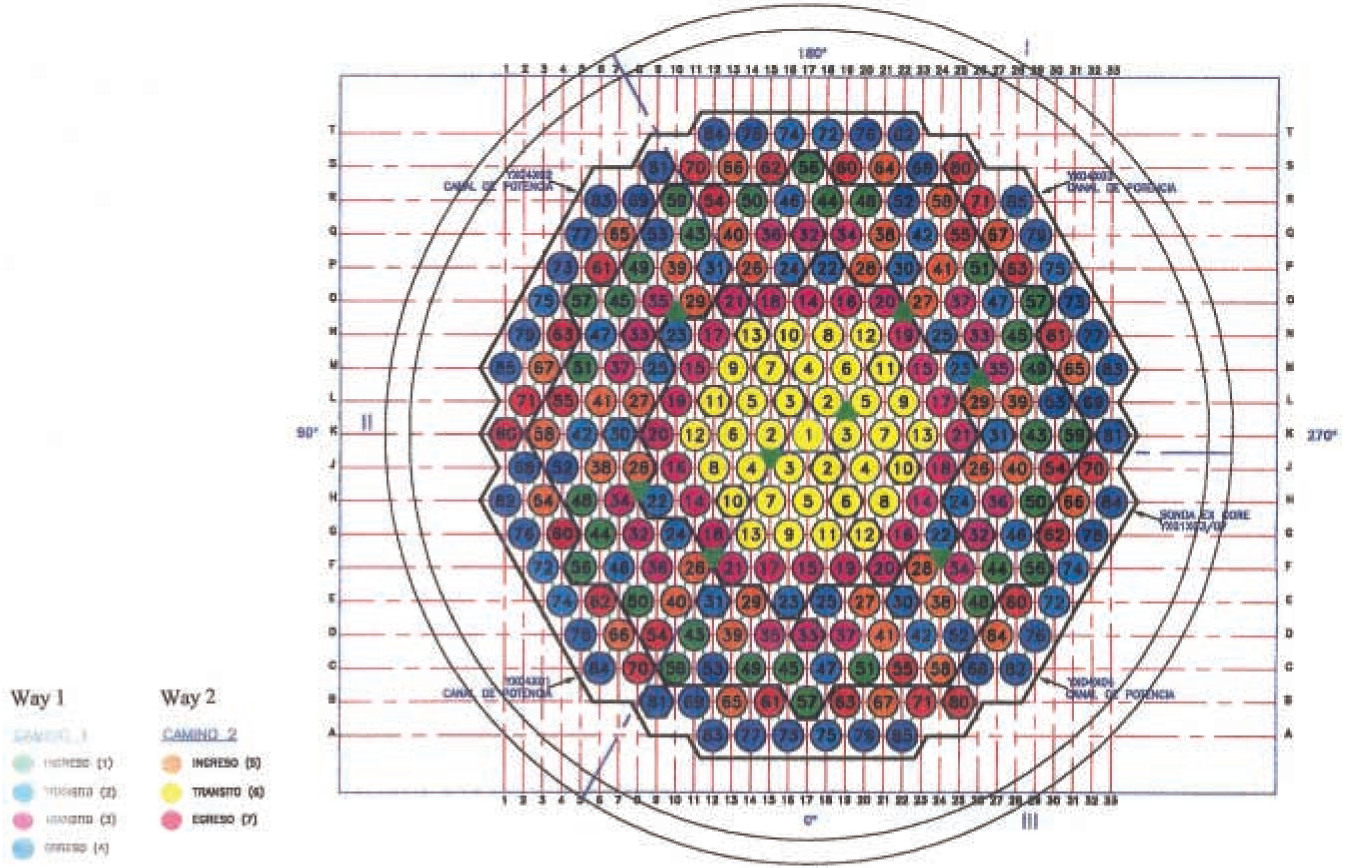


FIG. 180. *Scheme of in-core SEU fuel assembly management for the equilibrium Atucha 1 core.*

- Identification, simulation and analysis of new postulated accidents and abnormal transients associated with the use of SEU fuel assemblies.

The most relevant fact is the small impact on all reactor physics safety parameters of the replacement of natural uranium fuel by SEU. Postulated accidents are quite insensitive to such small changes (Table XXXVII) [216].

6.4.10.6. Longer term prospects

The future equilibrium 0.85% SEU core could, in principle, be the last stage in the fuel strategy for Atucha 1, since the most significant economic savings could be achieved just from the point of view of the fuel assemblies' operating costs. However, another aspect could then become relevant, namely, the necessity for a new fuel pool for irradiated fuel assembly storage up to the end of plant lifetime, since the current capacity using this type of SEU fuel assembly is insufficient. One interesting alternative to the expense entailed in constructing a new fuel pool in the plant would be to undertake a future transition from 0.85% SEU to around 0.95–1.0% SEU, just after achievement of the full 0.85% SEU load in the core. In that case, the additional increase in exit burnup will reduce the total accumulated, irradiated fuel assembly inventory towards the end of plant lifetime, thereby avoiding the need for a new fuel pool and bringing at the same time an additional saving (in the form of the direct fuel assembly consumption cost contribution) to the total operating cost.

6.4.10.7. Economic implications of enrichment in the Atucha 2 initial core

Regarding the initial core of a new reactor, an economic assessment has been performed of the impact on Atucha 2 of starting either with an initial full core of SEU, or with natural uranium, followed by a transition to SEU [218]. The startup core,

TABLE XXXVII. CHANGE IN REACTOR PHYSICS PARAMETERS FOR SEU CORE COMPARED WITH NATURAL URANIUM CORE IN ATUCHA 1

Parameter	Change (%)
Effective delayed neutron fraction	– 1.3
Average prompt neutron half-life	– 1.3
Fuel temperature reactivity coefficient	– 3.6
Coolant void reactivity	– 0.2
Boron reactivity	– 1.0

TABLE XXXVIII. LEVELIZED FUELLING COST SAVINGS AT ATUCHA 2: EFFECT OF INITIAL CORE

Case	Savings in fuelling cost (%)
Reference (natural uranium core full time)	
SEU core (full time)	27
Natural uranium fuel, equilibrium core SEU fuel	25

which is normally fully or mainly composed of fresh fuel assemblies, is far in composition from the later equilibrium core, which has a distribution of fuel burnup from fresh to discharge. The HWR startup core requires the use of neutron absorbers to compensate for the initial excess reactivity. These usually take the form of a liquid poison introduced into the moderator and/or a partial load of depleted uranium fuel assemblies. This situation is unique to the startup core, since the equilibrium core has little excess reactivity. This is very different from the case of LWRs, where there is no on-power refuelling and operation is under batch refuelling cycles.

The assessment showed very little difference in the fuelling cost (per kW·h) levelized over the plant lifetime between starting with an initial core of SEU, or starting with natural uranium followed by a transition to SEU. Using quite conservative assumptions and considering an equilibrium natural uranium core as the reference case, Table XXXVIII shows the savings in fuelling cost for the two cases at Atucha 2 using 0.85% enrichment SEU fuel.

6.5. HWR/PWR SYNERGISTIC FUEL CYCLES

6.5.1. Overview

The basis for the synergism between HWRs and PWRs arises from the fundamental characteristics of the two reactor types; spent PWR fuel has a higher residual fissile content than spent HWR fuel, while more energy can be derived from recycling that fissile material in the HWR than in the PWR. It should be noted that while the discussion pertains to LWRs generally, most of the analysis has focused on PWRs, and therefore this term is used here.

PWR fissile requirements are higher than those of the HWR, owing to the excellent neutron economy of the latter. The higher parasitic loads in the PWR lattice need to be compensated for by extra fissile material, in both fresh and spent fuels. As a result, spent PWR fuel has a high fissile content, about 0.9% ^{235}U and about 0.6%

fissile plutonium, depending on the initial enrichment and exit burnup. In the HWR, the fissile content in the fresh fuel is low, and the initial ^{235}U and the self-generated plutonium are burned to low levels as a result of the high thermal flux. HWR fresh natural uranium contains 0.7% ^{235}U, while the spent fuel contains ~0.2% ^{235}U and 0.2–0.3% fissile plutonium. Even using SEU in an HWR, with an initial ^{235}U enrichment of around 1.2%, there is still only about 0.4% fissile material in the spent fuel. Spent PWR fuel has about 1.5% fissile content, compared with about 0.4% fissile material in spent HWR fuel. Spent PWR fuel, therefore, can be viewed as a 'mine' of fissile material for HWRs.

The excellent neutron economy of the HWR means that up to double the thermal energy can be extracted from the fissile material in the spent PWR fuel by recycling it in an HWR rather than in a PWR. This applies whether the recycled material is uranium, plutonium, or both materials (see Table XVII).

Many technologies can be envisaged as being capable of exploiting this synergism. While separation and concentration of plutonium are required before it can be reused in a PWR, this is not required for recycle in an HWR. Hence, separation of plutonium, only to be followed by its dilution in fresh uranium, is clearly not optimal. This opens the door to new recycling technologies, which have the potential for being simpler and cheaper than conventional reprocessing, while having a much higher degree of proliferation resistance. The DUPIC cycle (Section 6.5.2) is a dry process for converting spent PWR fuel into fresh CANDU fuel, and one which offers a very high degree of proliferation resistance. The TANDEM cycle (Section 6.5.3) involves some degree of chemical 'decontamination' of the spent PWR fuel, followed by co-precipitation of the uranium and plutonium together. It has a higher degree of proliferation resistance than conventional reprocessing, but not as great as that offered by the DUPIC cycle.

If a country already has reprocessing facilities, or has access to them, then all of the products from reprocessing could be effectively utilized in HWRs. The uranium (recycled uranium) constitutes the largest component from reprocessing, and this can be recycled 'as is' in HWRs. The direct use of recycled uranium in HWRs has several benefits compared with its re-enrichment for recycle in PWRs (see Section 6.4.5) [163]. The second product of reprocessing is plutonium. This could be mixed with natural or depleted uranium, and recycled as MOX fuel in HWRs (Section 6.5.4). Finally, the true waste from reprocessing consists of minor actinides and fission products. The destruction of the actinide waste in an inert matrix carrier in CANDU is discussed in Section 6.5.5. These recycle options make use of the products of existing reprocessing facilities.

Finally, conventional reprocessing and DUPIC define the two extremes of the spectrum of PWR spent fuel recycling opportunities with the HWR. Most of the possibilities between these extremes have yet to be explored. Depending on local and international constraints and values, the optimal recycling process might lie between them. For example, a higher degree of proliferation resistance can be achieved in the TANDEM cycle by leaving in the highly radioactive fission products (and removing the rare earths that impact fuel burnup). This would also be a much cheaper process than conventional reprocessing, with its very high decontamination factor. Regarding the DUPIC process, fuel cycle economics could be improved by selectively removing the neutron absorbing rare earth fission products, thereby increasing burnup of the DUPIC fuel. Several techniques could be envisaged as achieving this. The simple, small HWR fuel bundle would facilitate fabrication.

The HWR is an 'evolutionary' reactor, offering a 'tailor made' fuel cycle. Fuel cycle flexibility and high neutron economy open the door to unique recycling opportunities having the potential for significant cost and non-proliferation benefits. The HWR could form an indispensable part of any LWR system employing recycling, on either a national or regional basis. New recycling technologies may be developed to take purposeful advantage of the unique niche that the HWR can fill with regard to spent PWR fuel recycle using processes that are simpler and cheaper than reprocessing, and which are designed from the start with a high degree of proliferation resistance. Moreover, this recycle could be done in existing HWRs, or in new reactors optimized for this purpose. The development of advanced recycle technologies could provide an opportunity for international collaboration.

6.5.2. The DUPIC fuel cycle

6.5.2.1. Introduction

The acronym DUPIC refers to a group of options for recycling spent PWR fuel into CANDU fuel using only dry thermal/mechanical processes; no wet chemical processing being employed. Furthermore, there is no selective element removal in the DUPIC process. This, along with the high radiation fields associated with the fuel, offers a very high level of proliferation resistance. AECL, KAERI and the US State Department have collaborated since the early 1990s on an assessment of these dry recycle options [219, 220]. The IAEA has also participated in the safeguards aspects of this programme.

Several DUPIC variants have been identified. One option is simply to cut the PWR fuel elements into CANDU lengths (~50 cm), straighten them, and then weld new end caps to the ends. Optionally, the elements could be double clad. The smaller diameter PWR elements would enable the use of a 48 or 61 element fuel bundle, which would significantly reduce the linear element ratings compared with those of a 37 element bundle. The low linear element ratings would enhance fuel performance and help accommodate the variation in fissile content between elements.

Another option is the vibration packing of ground PWR pellets into fresh CANDU sheaths. Yet another option is the so-called ‘OREOX’ (oxidation and reduction of oxide fuels) process whereby a series of oxidation/reduction cycles are employed to convert the used PWR pellets into a ceramic grade powder, after the cladding has been removed. The powder would be pressed and sintered to form ‘new’ CANDU pellets, which would be loaded into standard sheaths and assembled into fresh bundles. All options were judged to be technically feasible, and the last option is the focus of a collaborative programme between AECL, KAERI and the US State Department, which is further assessing the technical feasibility.

The DUPIC fuel cycle provides a country having both PWRs and CANDU reactors with greater energy self-reliance. The PWR spent fuel storage bays become essentially a domestic mine of fissile material that can be used to fuel CANDU reactors.

In the nominal DUPIC fuel cycle, spent PWR fuel, having a nominal burnup of 35 MW·d/kg HE, would be processed into CANDU fuel. An additional burnup of >15 MW·d/kg HE would then be obtained through irradiation in CANDU, followed by interim storage of the spent DUPIC fuel and then final disposal.

The DUPIC fuel cycle offers several benefits to a country that has both PWRs and CANDU reactors, including:

- A very high degree of proliferation resistance throughout the entire fuel cycle,
- Fuel conversion and fabrication costs that are expected to be lower than for conventional fuel reprocessing and MOX fabrication,
- A significant improvement in uranium utilization,
- Reduced quantities of spent fuel per unit of electrical energy produced,
- A simple disposal concept combined with a reduction in the cost of geological disposal.

6.5.2.2. Fuel design, fabrication and performance

The heart of the reference DUPIC fuel cycle is the OREOX process. During this dry process, uranium from spent PWR fuel is sequentially oxidized and reduced to a fine powder, which forms the starting material for the fabrication of DUPIC fuel pellets. The powder is conditioned to improve its sintering properties, pressed into pellets, sintered to a high density, ground to final size, and seal welded within Zircaloy sheaths. As the fuel remains highly radioactive, all processing must be done in hot cells.

The entire fabrication process involves the following steps:

- Decladding the spent PWR fuel;
- Exposing the fuel to thermal cycles of oxidation and reduction, in order to break the fuel into a fine powder;

- Milling (or any subsequent powder conditioning) to improve the sintering properties of the powder;
- Fabricating CANDU quality fuel pellets from the powder;
- Loading the pellets into sheaths;
- Assembling the CANDU bundles;
- Disposing of irradiated PWR assembly hardware;
- Trapping and disposing of volatile fission products released during the decladding, OREOX and sintering processes.

The CANFLEX bundle is the reference for the DUPIC cycle, having 20% lower peak linear element ratings than a 37 element bundle, and improved thermohydraulic performance.

The fuel from different PWR assemblies would be blended to ensure that each bundle has equivalent (within specified tolerance) neutronic properties.

The OREOX dry reconstitution process employed in the DUPIC fuel cycle is potentially a much simpler process than conventional wet chemical reprocessing. This process does not involve chemical separation of the spent fuel. The final fuel product (DUPIC bundles) is fabricated in the same facility in which the spent fuel is processed. This process contrasts with conventional wet reprocessing, in which the fuel is separated into uranium, plutonium and fission product/actinide waste, in a process that requires very high decontamination factors. The PWR MOX fuel made from recovered plutonium is fabricated remotely in a different processing facility. Moreover, the CANDU fuel bundle is very simple in design and fabrication (with only seven distinct components), and is smaller (~0.5 m long) than a PWR assembly (~4.1 m long), which facilitates remote fabrication. Thus, even though fabrication must be done in a dedicated shielded facility, DUPIC fuel processing and fabrication is expected to be less expensive than conventional reprocessing and PWR MOX fuel fabrication. While preliminary assessments indicate that this is the case, more technical work is required to specify the process details before more definitive costs can be confirmed. A comparison of indicative DUPIC and reprocessing/PWR MOX fabrication costs is shown in Table XXXIX [116, 130, 221–223].

TABLE XXXIX. INDICATIVE DUPIC AND REPROCESSING/PWR MOX COSTS

Process stage	DUPIC costs (US $/kg)	Reprocessing/PWR MOX costs (US $/kg)
Spent fuel separation		1000
Fuel fabrication	600	800
Total cost of fuel	600	1800

(a) Fabrication progress to date

Following a series of hot cell experiments conducted at AECL's Chalk River Laboratories to demonstrate fabrication of CANDU quality pellets using actual spent PWR fuel, a campaign was initiated to fabricate DUPIC fuel elements for irradiation testing in the NRU research reactor at Chalk River Laboratories. A total of three DUPIC elements were fabricated in the hot cells at AECL's Whiteshell Laboratories Shielded Facilities [224]. Approximately 3 kg of spent PWR fuel was processed into fuel pellets using the OREOX process. The pellets were then formed into stacks, loaded into three fuel elements and welded. The DUPIC fuel elements were designed to be mounted on a special 37 element geometry bundle used for experimental irradiations in the NRU research reactor at Chalk River Laboratories.

Measurements of the chemical content of the fuel before and after the DUPIC fuel fabrication process indicated that volatile caesium was released during the process. Krypton, iodine and xenon were also thought to have been released. All other fission products and transuranic elements were retained in the fuel. This retention is an important feature for the proliferation resistance of the fuel cycle. The irradiation of these elements in NRU started in the spring of 1999.

6.5.2.3. Reactor characteristics

One of the key criteria in the final selection of the DUPIC variants for further assessment, was compatibility of the fuel and fuel cycle with existing reactors, the objective being that no major reactor changes be required to utilize DUPIC fuel. Hence, the OREOX option utilizes an existing element and bundle design (CANFLEX), and a pellet design that meets current CANDU specifications. Obviously, changes will have to be made to storage and handling, and to the loading of fresh DUPIC fuel into the refuelling machines, but aside from these, other changes to the reactor are expected to be minimal.

Detailed reactor physics assessments have been performed for the DUPIC fuel, including lattice studies, detailed time dependent fuel management simulations and LOCA analysis [225]. These studies confirm that DUPIC fuel can be accommodated within existing CANDU reactors. Some of the key results of these studies are summarized below.

In this analysis, the Republic of Korea's 17 × 17 optimized fuel assembly was used as the reference PWR fuel, from whence the composition of the DUPIC fuel was derived. The reference PWR fuel has an initial enrichment of 3.5 wt% ^{235}U, and a burnup of 35 MW·d/kg HE, which are representative of the 950 MW(e) PWR plant, Yonggwang 1. At discharge, the fuel contains 0.92 wt% ^{235}U, 0.56 wt% ^{239}Pu and 0.08 wt% ^{241}Pu, representing a total fissile content of 1.56 wt%. It was assumed, for the purposes of the physics calculations, that during fuel processing (OREOX and sintering), all of the ruthenium and technetium were removed, as well as 98% of the krypton, iodine, xenon and caesium. None of the high neutron absorbing rare earth isotopes were assumed to be removed during processing.

Since DUPIC fuel has a faster dynamic response than UO_2, because of the lower delayed neutron fraction and lower prompt neutron lifetime of plutonium isotopes, the same positive reactivity insertion will result in a faster response than for UO_2. To compensate for this in a postulated large LOCA, void reactivity for the DUPIC fuel was reduced by adding a small amount of neutron absorber to the centre element (26 g of natural dysprosium). This is an embodiment of the low void reactivity fuel concept, and reduces void reactivity sufficiently that the power pulse with the DUPIC fuel is lower than with natural uranium. The downside, of course, is reduced burnup: without the neutron poison the burnup of the DUPIC fuel is ~18.8 MW·d/kg HE; with the burnable poison, the burnup drops to 15 MW·d/kg HE, which is still double that of natural uranium fuel.

The reactor physics core characteristics were calculated using the standard three dimensional CANDU core physics code, RFSP [213]. A 600 d, time dependent refuelling simulation was performed to assess various fuel management options, and to arrive at accurate estimates of core characteristics. A two bundle shift, bidirectional fuelling scheme was found to give excellent results. Maximum bundle and channel powers were well below licensing limits; maximum channel powers were comparable to natural uranium with an eight bundle shift fuelling scheme, while maximum bundle powers were 10% lower. The channel power peaking factor (defined as the ratio of instantaneous channel power to time average channel power for that channel) is similar to that of natural uranium. The refuelling rate in channels per day (4.05) is double that of natural uranium fuel (2.0), while in bundles per day it is half (8 as opposed to 16 bundles per full power day). With a two bundle shift fuelling scheme, there were large margins to the stress corrosion cracking failure curves, giving good confidence in fuel performance. With a two bundle shift, the axial power distribution has the inlet skewed, flattened distribution characteristic of enriched fuel, peaking at bundle position 4, and decreasing along the length of the channel to the coolant outlet end. This is expected to result in an improved critical channel power relative to natural uranium, but this has not been quantified.

A 20% inlet header break LOCA was simulated using the CERBERUS three dimensional, space time, neutron kinetics module within RFSP, with the thermohydraulic response provided by a three dimensional system thermohydraulics code. The peak power pulse with the DUPIC fuel was 7% lower than for natural uranium. Hence, the performance is comparable to that of natural uranium fuel.

In summary, detailed reactor physics assessments have shown that the reference DUPIC fuel can be fuelled in the current CANDU 6 reactor without requiring significant modification of the reactor system, and with good performance predicted.

In addition to the reactor physics analysis performed by AECL and KAERI as part of the DUPIC programme, significant, independent analysis has been performed at the Ecole Polytechnique de Montréal [226, 227]. This analysis was performed using a single chain of codes and the associated nuclear cross-section library (ENDF/B-V), and a single set of nuclides for both the PWR (to determine the spent fuel composition) and CANDU DUPIC parts of the cycle, ensuring self-consistency for the entire cycle. The neutron transport code DRAGON was used to model the PWR assembly (a 2-D, 17 × 17 depletion calculation) and the cell properties of the CANDU cluster [228]. The CANDU DUPIC core characteristics were calculated using the DONJON code, which solves the 3-D diffusion equations using a finite element model [229]. A 17 × 17 French standard 900 MW(e) PWR fuel assembly was used as the reference, with an initial enrichment of 3.2% ^{235}U and a discharge burnup of 32.5 MW·d/kg HE. This corresponds to the Daya Bay Nuclear Power Plant, using a three batch size and an out–in loading pattern.

Typical time average axial power distributions are shown in Fig. 181 for the CANDU DUPIC core for two and four bundle shift refuelling schemes, and for an eight bundle shift fuelling scheme with natural uranium, for comparison. Instantaneous calculations (simulating a ‘snapshot’ of the power distribution at a point in time) show that maximum bundle and channel powers are lower than those of a natural uranium fuelled core.

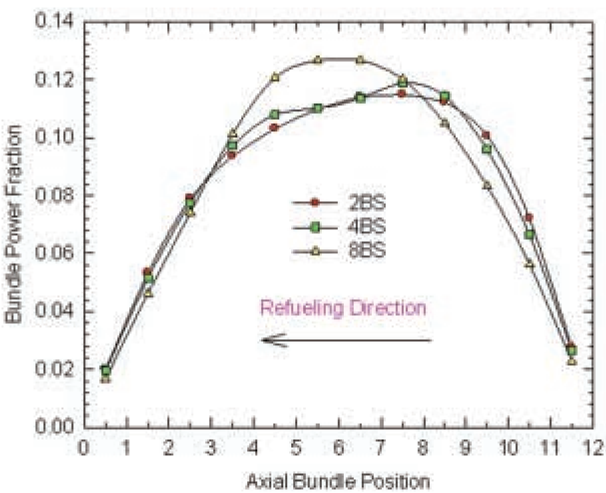


FIG. 181. Typical axial bundle power profiles for DUPIC fuel (channel L3; DUPIC, two and four bundle shifts; natural uranium eight bundle shift) [226].

The effect of heterogeneity in the fresh DUPIC fuel composition has been investigated by varying both the initial PWR enrichment and the discharge burnup by 10%, and by considering the pin to pin and axial variation in properties within an assembly [226]. Simply mixing all the fuel powder from one spent PWR assembly into a single powder batch after OREOX processing eliminates the heterogeneity arising from pin to pin and axial variations. The CANDU lattice reactivity of DUPIC fuel made from single assembly mixing is very close to that of the nominal DUPIC fuel with the same assembly average discharge burnup. In the case of assemblies having the same enrichment but different discharge burnups, then mixing the powder from two assemblies in order that the volume average discharge burnup is close to the nominal value eliminates the assembly to assembly heterogeneity. In the case of assemblies having different enrichments, blending with virgin enriched or natural uranium powder can be considered a means of ensuring that the DUPIC fuel possesses homogenous neutronic properties [230].

The effect of PWR fuel management strategy (PWR initial enrichment, batch size and discharge burnup) on the CANDU DUPIC burnup has been investigated and the main conclusions are summarized in Table XL [227]. For a given PWR initial enrichment, as the PWR batch size decreases, the PWR burnup increases, the CANDU DUPIC burnup decreases, and the total PWR plus CANDU DUPIC burnup remains relatively constant. For a given PWR batch size (for example, 1/3 batch size, or 1/4 batch size), the CANDU DUPIC burnup remains constant with respect to PWR initial enrichment, even though the PWR discharge burnup can increase significantly with initial enrichment. In all cases, the CANDU DUPIC instantaneous peak powers are comparable to, or lower than, those of natural uranium fuel (using a two bundle shift fuelling scheme with DUPIC fuel, and an eight bundle shift fuelling scheme with natural uranium fuel). These results have important implications in terms of optimizing the total PWR/CANDU DUPIC fuel cycle. For instance, the observation that the total PWR plus CANDU DUPIC burnup is constant for a given PWR enrichment suggests that total fuel cycle costs might be reduced in a PWR/CANDU DUPIC system by going to the largest batch size in the PWR (even single batch refuelling in the PWR), thereby maximizing the time between refuellings in the PWR, since the burnup lost in the PWR is gained in the CANDU DUPIC part of the cycle.

6.5.2.4. Non-proliferation and safeguards considerations

A major attraction of the DUPIC fuel cycle is that it offers a high degree of proliferation resistance. Several features of the DUPIC process significantly enhance its proliferation resistance relative to fuel cycles employing separated plutonium:

- The proliferation barriers that are present in spent fuel are also present in DUPIC fuel.

TABLE XL. CHARACTERISTICS OF DIFFERENT DUPIC CORE PERFORMANCES WITH VARIOUS SPENT PWR FUEL TYPES [227]

Initial enrichment in PWR (% U-235)	Batch size in PWR	DUPIC fuel types	Equilibrium discharge burnup (MW·d/kg HE)			Instantaneous calculation peak powers (kW)		Channel power peaking factor
			PWR	CANDU	Total	Channel	Bundle	
Natural U			0.0	7.45	7.45	6799	866	1.074
3.2	1/2.5	1	30.0	17.71	47.71	6821	857	1.074
3.2	1/3	2	32.5	15.10	47.60	6754	837	1.062
3.2	1/4	3	34.6	13.00	47.60	6741	806	1.056
3.5	1/2.5	4	33.9	17.75	51.65	6814	853	1.072
3.5	1/3	5	36.5	15.08	51.58	6759	837	1.062
3.5	1/4	6	38.9	12.72	51.62	6718	794	1.054
3.8	1/2.5	7	37.3	18.24	55.54	6813	852	1.072
3.8	1/3	8	40.3	15.19	55.49	6754	831	1.061
3.8	1/4	9	42.9	12.66	55.56	6703	793	1.053
4.2	1/2.5	10	42.0	18.63	60.63	6812	849	1.072
4.2	1/3	11	45.4	15.20	60.60	6743	820	1.060
4.2	1/4	12	48.3	12.42	60.42	6690	799	1.050
4.5	1/2.5	13	45.3	19.07	64.37	6817	849	1.072
4.5	1/3	14	48.9	15.46	64.36	6741	818	1.059
4.5	1/4	15	52.0	12.51	64.51	6684	802	1.049

- There is no purposeful separation of isotopes, nor can the processes be easily tampered with to effect such a separation.
- The fuel processing does not involve any wet chemistry; only dry thermo-mechanical processes are employed.
- With no selective separation, the plutonium concentration remains dilute throughout the entire fabrication process, making it much more difficult to remove a significant quantity.
- All stages of the fabrication process, as well as the final DUPIC fuel bundles themselves, are highly radioactive, making physical access to the material, and its removal, extremely difficult.
- The high level of radioactivity results in an easily detectable signature, making removal of material easy to detect.
- All processing and handling must be done in a shielded facility, thereby making physical entry into the facility and removal of material extremely difficult. These measures will also result in highly automated processes and the inherent capability to track movements and maintain fissile material inventory control.

- The processing facility is entirely self-contained. Spent PWR fuel is an input to the facility, and finished DUPIC fuel bundles are the product; there is no transport of intermediate products.
- Transportation of the spent PWR fuel into the DUPIC processing facility and of DUPIC fuel to the CANDU reactor involve the movement of highly radioactive materials.

Some of the very characteristics that enhance the proliferation resistance of DUPIC fuel also complicate the safeguarding of the process. For instance, plutonium 'accounting' is inherently easier to conduct with separated plutonium than it is with plutonium mixed with highly radioactive fission products. Unique safeguards measures are being developed and will be built into the design of a DUPIC manufacturing facility [231, 232].

6.5.2.5. Spent DUPIC fuel disposal

Approximately 40–60% more energy can be derived from PWR fuel by burning it again as DUPIC fuel in CANDU reactors. In an equilibrium system of CANDU reactors and PWRs employing the DUPIC cycle, the quantity of spent fuel arisings per unit of electricity generated is a factor of three lower than that of a dual open system with direct disposal. The DUPIC system results in a 30% reduction in spent fuel arisings compared with an all PWR system.

This reduction in the total quantity of spent fuel generated is an attractive feature because it can lead to a corresponding reduction in handling, storage and transportation activities prior to final disposal. This reduction in spent fuel quantities does not, in itself, have a significant impact on the cost of spent fuel disposal since, for a particular geological disposal design, the main determinant of cost is the total decay heat generated by the waste at the time of disposal (which is approximately determined by the amount of electricity generated), rather than the quantity of spent fuel created.

Nonetheless, there are two characteristics of the DUPIC fuel cycle that are expected to result in significant cost savings in fuel disposal: the simple CANDU fuel design and, more importantly, the decay heat characteristics of the spent DUPIC fuel. These are described in the following two sections.

(a) Simple disposal concept

The Canadian concept of spent CANDU fuel disposal (see Section 6.1.9) is relatively simple. The small size and compactness of the CANDU fuel bundles, which are loaded into corrosion resistant disposal containers, result in a small, easy to handle container (~2.3 m tall, ~0.65 m in diameter with a low mass of ~3 Mg). One option is to place these containers into boreholes drilled into the floor of disposal rooms, 500–1000 m underground in stable granitic rock formations.

PWR disposal containers are much larger (typically ~4.5 m tall and ~0.8 m in diameter) and heavier (~14 Mg). Thus, the height of the disposal rooms must be greater to accommodate the taller containers, and the containers themselves are more difficult to handle. In the Canadian context of geological disposal, the simple CANDU DUPIC fuel design provides a cost advantage over the disposal of PWR fuel, when compared either on the basis of $/kg HE, or ¢/kW·h.

(b) Decay heat of spent DUPIC fuel

A perhaps unexpected benefit of the DUPIC fuel cycle is that the additional energy obtained from DUPIC fuel compared with once through PWR fuel is achieved with little additional penalty in terms of spent fuel heat load [214]. A plot of heat output as a function of time for spent CANDU natural uranium, CANDU DUPIC and PWR fuels is shown in Fig. 182. The particular assumptions used in this calculation give a maximum burnup of 21 MW·d/kg HE for the CANDU DUPIC fuel, fabricated from spent PWR fuel having a burnup of 35 MW·d/kg HE, i.e. the CANDU DUPIC fuel has an effective burnup of 56 MW·d/kg HE. This figure is somewhat higher than

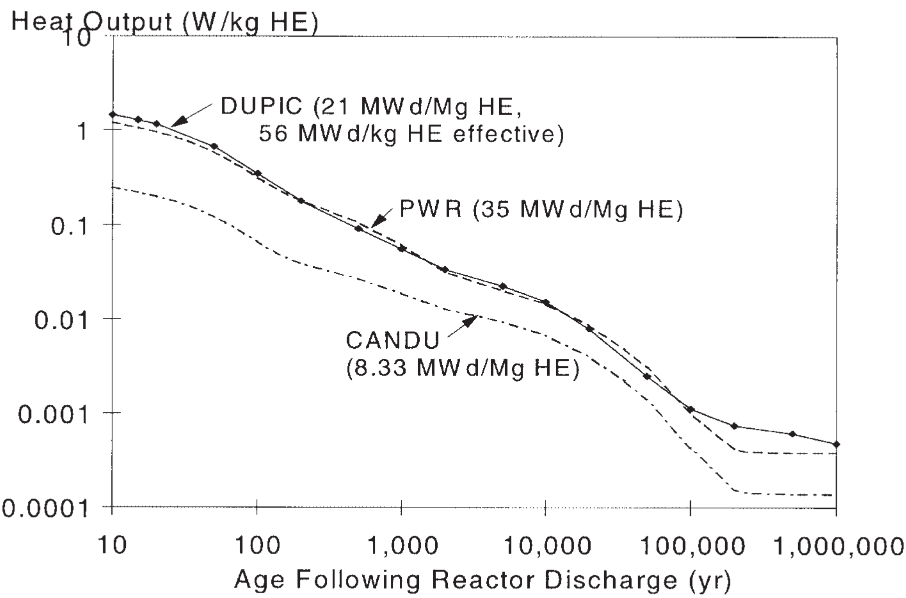


FIG. 182. Decay heat from discharged fuel.

that assumed in the core physics assessment, but this is not expected to change the main conclusions.

At 50 years following discharge, the decay heat output from the spent DUPIC fuel is 15% greater than that of the spent PWR fuel, even though an additional 60% more energy has been produced. As explained below, this is mainly a result of the removal of radioactive isotopes from the spent PWR fuel during the DUPIC OREOX and fabrication processes, particularly the heat generating caesium isotopes. In addition, some of the heat producing transuranic actinides (e.g. ^{241}Am) are transmuted or consumed in the softer CANDU irradiation spectrum.

In spent CANDU and PWR fuels, the fission product decay heat generally dominates the total decay heat for a period of up to 100 years. Soon after discharge from the PWR, the main decay heat contributors are ^{134}Cs, ^{137}Cs and ^{137}Ba^{m}. The ^{137}Ba^{m} is considered part of the ^{137}Cs decay heat source term because of its very short lived metastable decay state.

After 10 years of cooling, about 50% of the fission product decay heat in the spent PWR fuel (i.e. ~40% of the total decay heat (see Table XLI), essentially from Cs (and ^{137}Ba^{m})), is removed during the OREOX process. With the removal of Cs, the period of fission product decay heat dominance in the subsequent spent CANDU DUPIC fuel is decreased to <10 years (Fig. 182).

An important actinide decay-heat source is ^{241}Am (i.e. from about 50–1000 years after discharge). Much of the initial ^{241}Am inventory that is present in the fabricated CANDU DUPIC fuel is removed during irradiation in the CANDU reactor. The ^{241}Am that builds up as a result of ^{241}Pu decay in the spent PWR fuel is readily converted in the CANDU reactor to ^{242}Am and ^{242}Am^{m} via neutron capture. The ^{242}Am^{m} either undergoes neutron capture to form ^{243}Am, or it undergoes fission. Most ^{242}Am undergoes fission or beta decay and electron capture. The neutron cross-sections and decay rates for these reactions are quite large, and therefore the concentration levels of ^{242}Am and ^{242}Am^{m} remaining in the CANDU DUPIC fuel are quite small on a continuing basis.

TABLE XLI. DECAY HEAT FROM SPENT PWR FUEL

Parameter	Decay heat (W/kg HE)				
	Discharge	0.1 a	1 a	10 a	100 a
All isotopes	2.460×10^{3}	5.071×10^{1}	1.102×10^{1}	1.211×10^{0}	3.105×10^{-1}
Actinides	1.319×10^{2}	1.448×10^{0}	5.106×10^{-1}	2.340×10^{-1}	2.087×10^{-1}
Fission products:	2.328×10^{3}	4.926×10^{1}	1.051×10^{1}	9.768×10^{-1}	1.018×10^{-1}
Elemental Cs	1.574×10^{2}	1.819×10^{0}	1.307×10^{0}	1.585×10^{-1}	1.264×10^{-2}
Elemental Ba	9.155×10^{1}	1.207×10^{0}	4.156×10^{-1}	3.376×10^{-1}	4.219×10^{-2}

The levels of ^{241}Pu and ^{241}Am adjust to balance generation and depletion reactions during the CANDU stage irradiation. The concentration of ^{241}Pu at discharge is greater in PWR fuel than in CANDU DUPIC fuel. Conversely, the concentration of ^{241}Am at discharge is greater in CANDU DUPIC fuel than in PWR fuel. Eventually, the decay heat from ^{241}Am becomes greater in the spent PWR fuel than in the spent CANDU DUPIC fuel, owing to ^{241}Am buildup as a result of ^{241}Pu decay.

The total decay heat from ^{239}Pu and ^{240}Pu in both spent fuels is similar over the first 1000 years following discharge. The ^{240}Pu component is greater in the spent CANDU DUPIC fuel, reflecting the larger cumulative fuel irradiation than that of the spent PWR fuel. Since ^{239}Pu decays more slowly than ^{240}Pu, eventually the decay heat component from ^{239}Pu becomes the dominant actinide decay heat source for several hundred thousand years.

In summary, the decay heat characteristics of the higher burnup spent CANDU DUPIC fuel are similar to those of spent PWR fuel owing to the:

- Extraction of Cs in the spent PWR fuel by the OREOX process,
- Transmutation of ^{241}Am in the spent PWR fuel in the CANDU reactor,
- Consumption of ^{241}Pu in the spent PWR fuel in the CANDU reactor,
- Consumption of ^{239}Pu relative to ^{240}Pu in the CANDU reactor.

(c) Spent CANDU DUPIC fuel disposal costs

Owing to spent fuel decay heat being the major determinant of the density of spent fuel packing in a geological repository, the extra energy derived from the spent PWR fuel by burning it as CANDU DUPIC fuel is obtained with only a small increase in fuel disposal cost. As a result, the overall unit disposal cost of DUPIC fuel (in ¢/kW·h) is significantly lower than the disposal cost of either spent PWR fuel or CANDU natural uranium fuel.

In the conceptual design analysis of disposal of spent CANDU DUPIC fuel, it was found, because of its high heat output, that the amount of fuel in a disposal container must be:

- Reduced from 72 fuel bundles with natural uranium fuel, to 60 bundles;
- Stored for 50 years following discharge from the CANDU reactor, to achieve the 90°C temperature design limit.

Cost results are shown in Fig. 183 [214]. The costing methodology was the same as that used for assessing the disposal costs of SEU fuel (see Section 6.4.7). Unit total costs of spent CANDU DUPIC fuel disposal are less than those of spent CANDU natural uranium and CANDU SEU fuels. Generally, the unit total costs of

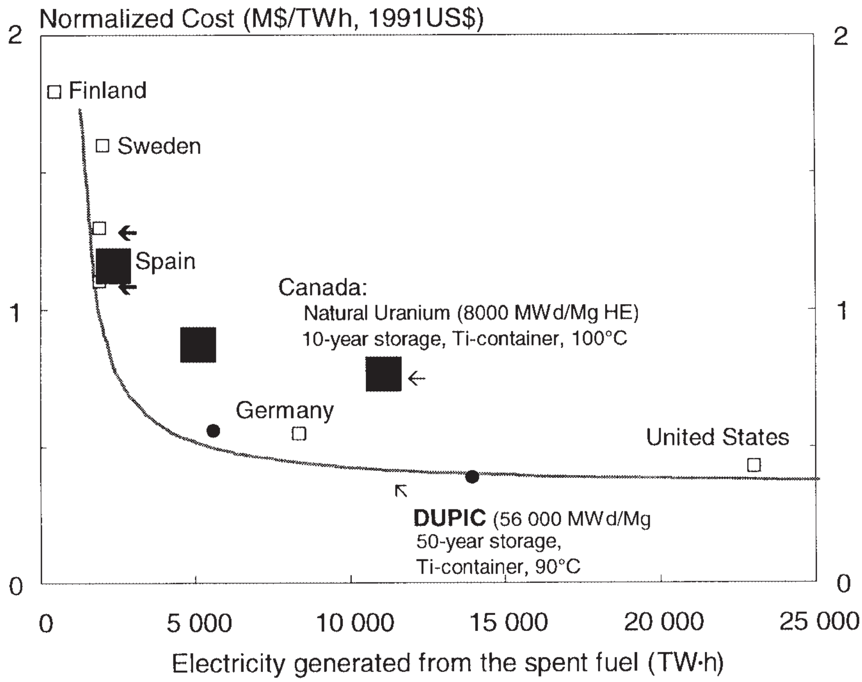


FIG. 183. Canadian CANDU unit cost of spent fuel disposal: Effect of recycling spent PWR fuel in CANDU reactor (DUPIC fuel cycle) [214].

spent CANDU DUPIC fuel disposal are estimated to be considerably less than those of spent PWR fuel in other countries.

(d) Caesium waste management

Radioactive ^{134}Cs and ^{137}Cs (with its short lived decay product $^{137}Ba^{m}$), which contribute 40% of the total decay heat from spent PWR fuel 10 years after discharge from the reactor, are removed in the DUPIC plant and trapped in ceramic based filters. Caesium-137, the dominant component, has a relatively short half-life (30 years) and can be stored until its decay heat decreases significantly (after approximately 50–100 years) before going to disposal. Preliminary unpublished analysis indicates that the co-disposal of caesium waste with spent DUPIC fuel would increase the disposal costs by ~10%. The other off-gases collected from the OREOX process generate no heat and contain very long lived radionuclides (e.g. ^{129}I, ^{99}Tc and ^{14}C) that require deep geological disposal and which should not significantly add to disposal costs (e.g. judged as amounting to a few per cent, although no analysis has yet been performed). On the basis of these additional costs, the overall unit disposal cost of DUPIC fuel remains significantly lower than that of the other options.

6.5.2.6. DUPIC fuel cycle economics

In the Republic of Korea, DUPIC fuel cycle economics have been assessed sensitivity studies and the results compared with once through PWR and CANDU fuel cycles, combinations of once through PWR and CANDU fuel cycles, and reprocessed PWR (MOX) fuel cycles [223]. The levelized cost analyses have considered front end costs (uranium purchase, conversion, enrichment, fuel fabrication and transportation) and back end costs (interim dry storage, transportation and ultimate disposal). The main conclusion drawn from these fuel cycle economic assessments is that the DUPIC fuel cycle is a competitive option compared with once through cycles with direct disposal, and with PWR reprocessing and PWR MOX recycle. These cost analyses are necessarily preliminary. More work is required to define the technical aspects of the DUPIC cycle, to reduce the uncertainties associated with some of the cost components, and to determine the impact of changes to the reactor system, such as remotely handled DUPIC fuel storage and loading. Nonetheless, the results achieved to date are encouraging, and suggest an overall attractive economic case for the DUPIC fuel cycle.

6.5.3. The TANDEM fuel cycle

6.5.3.1. Introduction

In the TANDEM fuel cycle, the uranium and plutonium from PWR spent fuel are co-precipitated without separation. Only the fission products and higher actinide isotopes are removed. AECL and KAERI jointly investigated this fuel cycle in the early 1980s. This fuel takes advantage of the fact that the fissile component in PWR spent fuel (about 1.5%) can be used directly in HWRs without readjustment of the enrichment. Fuel burnup in this cycle would be at least 25 MW·d/kg HE.

Removal of all parasitic absorbers from the spent PWR fuel in the TANDEM cycle results in the highest possible burnup in an HWR, and hence, the highest possible overall HWR/PWR system uranium utilization. In equilibrium, natural uranium requirements can be reduced by as much as 40%, compared with an all PWR system (Table XVII). This would apply to an equilibrium HWR/PWR system, where the fresh fuel requirements of the HWRs were met by the spent fuel discharge from the PWRs. These savings are insensitive to whether the spent PWR fuel is diluted with natural or depleted uranium when recycled into the HWR. In such a system, enrichment requirements are reduced by 40–50% of what they would be for an all

PWR system. Finally, in equilibrium, the TANDEM cycle could reduce the quantity of spent fuel by a factor of between 2.5 and 5, compared with the HWR/PWR system. This would be a significant benefit, both for interim storage and for final disposal.

6.5.3.2. Fuel design, fabrication and performance

The original TANDEM fuel cycle concept involved a ‘chemical decontamination’ process to separate fission products and unwanted actinides from the uranium/plutonium mixture [233]. This process would have a higher degree of proliferation resistance because separated plutonium is not produced in the process. It is also potentially cheaper than conventional reprocessing, since partitioning of the uranium and plutonium is not required, nor is purification of separated uranium and plutonium. Fewer steps should result in lower maintenance costs as well. Some of the important features of the process, as originally conceived in the AECL/KAERI studies, are as follows:

- Since the fuel is co-processed, the plutonium is not separated from the uranium at any stage in the process. The process is one of co-decontamination where fission products and higher actinides are removed from the fuel material. This affords significant simplifications over the more conventional approach, while increasing the proliferation resistance.
- Co-decontamination was achieved with three stages of PUREX solvent extraction (employing 30% tri-butyl phosphate in kerosene), resulting in a minimum decontamination factor of 10^7.
- High level waste management could use technology developed for conventional reprocessing.
- Several technical options are available for co-conversion of the reprocessing liquor into a form suitable for the fabrication of MOX fuel. The process selected was based on denitration of the liquor to a powder. All by-products from this process would be recycled, and no unfamiliar waste forms would be produced.
- Fuel fabrication is fairly conventional in the sense that glovebox technology would be used. The short, simple CANDU fuel bundles would facilitate fuel fabrication. Co-conversion avoids the troublesome step of blending pure uranium and plutonium powders, and removes concerns relating to the effect of ‘hot spots’ occurring in the fuel during irradiation.

Little work has been done on this process since the TANDEM cycle was jointly examined by AECL and KAERI in the early 1980s, the focus now being on the DUPIC cycle. Variants of the TANDEM cycle remain as candidates for future consideration as advanced recycling technologies, particularly simpler (and potentially cheaper) processes that selectively remove only the rare earth neutron absorbing fission products (to maximize the burnup), while leaving in the other fission products to increase proliferation resistance.

6.5.3.3. Reactor characteristics

The unadjusted TANDEM process yields a product having an enrichment of around 1.5% total fissile. While this could be used as is in CANDU, the fissile loading could also be reduced by dilution with natural or depleted uranium. Such dilution might be required to allow the varying fissile concentration of discharged PWR fuel to be adjusted to a more uniform range, in order that all the fuel had the same neutronic properties. Dilution would also allow fuel bundles of non-uniform radial enrichment to be fabricated, if necessary.

The CANFLEX bundle would be used for this application, since the lower linear element ratings would facilitate the achievement of higher burnups. If there were a need to accommodate the fast dynamic response of the MOX fuel owing to the presence of plutonium, one option would be to add, at the expense of neutron economy, a small amount of neutron absorber to the centre of the bundle in order to reduce void reactivity.

In general, the same fuel management approaches that have been devised for SEU can also be applied to MOX fuel in CANDU. With relatively low enrichments and corresponding burnups below ~15 MW·d/kg HE, a simple two or four bundle shift fuelling scheme is appropriate. With higher enrichments and burnups, the presence of the adjuster rods in the central region of the core would result in an undesirable asymmetric double hump in the axial power profiles of those channels near the adjuster rods, if a simple fuel management scheme were employed.

This could be accommodated for by either removing the adjuster rods, or by using one of the more sophisticated fuel management strategies discussed in Section 6.4.4.2.

6.5.3.4. National perspectives: India

Some work is being undertaken on strategies for recycling safeguarded plutonium from one safeguarded reactor in another safeguarded reactor. In this context, plutonium from the Rajasthan HWRs is being used for the fabrication of MOX fuel for use in the Tarapur BWRs. Studies are also investigating the possibility of using the plutonium in the fuel discharged from India's Tarapur BWRs as feed for the HWRs. At present, India has two BWRs, each of 160 MW(e) capacity. The discharged fuel from the BWRs could be subject to a co-processing treatment, as in the TANDEM cycle described above, in which the fission products would be removed but all the actinides would be precipitated together. There would thus be no separation of plutonium and uranium, which would enhance the proliferation resistance of the process. This mixture of heavy metals would then be fabricated into 19 element bundles and then introduced into the HWR. The salient characteristics of the BWR lattice are given in Table XLII. The isotopic composition of the fuel, which is discharged at a burnup of 21 MW·d/kg HE, is given in Table XLIII. Fuel of this composition, when fabricated into 19 element bundles and loaded into the HWR, can give a discharge burnup of 37.2 MW·d/kg HE. This means that 1 GW(e) of BWR can support the fuelling requirements of 1.7 GW(e) installed capacity of HWR. This ignores losses during fabrication, coprocessing, etc., but if all these factors are also taken into account, a ratio of ~1.5 can still be expected.

The void reactivity of this lattice is about 6.7 mk (positive). This is lower than the value (~10.8 mk) for natural uranium fuel. However, the delayed neutron fraction is 4.12 mk, as opposed to 6.5 mk in the case of natural uranium. The prompt neutron lifetime is 0.219 ms. An approximate LOCA analysis was carried out to evaluate the effect of changes in the kinetics parameters. It was assumed that the PHTS voiding process, at least in so far as the reactivity effect was concerned, would be complete in 1 s, and therefore the total void coefficient could be considered to have been added to the reactivity in 1 s. Blowdown was supposed to last for 10 s, during which time there was improved heat transfer. Thereafter, the ECCS would come in. However, this had been neglected in this calculation and therefore beyond 10 s the heat transfer was assumed to be negligible. Shutdown in this LOCA calculation was initiated by the overpower trip, by moderator dump. Among Indian HWRs, there are four units which were built in the early days, and in which shutdown is effected by moderator dump. Later units have two, independent, fast acting shutdown systems. The TANDEM cycle is being considered for one of the earlier reactors, hence the use of moderator dump for effecting reactor shutdown in the studies presented here. The peak power in

TABLE XLII. BWR LATTICE CHARACTERISTICS

Parameter	Value
Number of fuel assemblies	284
Mass of UO_2 in one assembly (kg)	160
Mass of heavy metal in one assembly (kg)	141
Total uranium inventory (Mg)	40
Specific power (kW/kg)	17.5
Fuel pellet radius (mm)	6.18
Inner radius of cladding (mm)	6.34
Outer radius of cladding (mm)	7.15
Pin to pin pitch (mm)	17.9
Average U-235 enrichment (%)	2.4
Discharge burnup (MW·d/kg HE)	21

TABLE XLIII. ISOTOPIC COMPOSITION OF BWR DISCHARGED FUEL

Isotope	Concentration in discharged fuel (%)
U-235	1.02
U-236	0.28
U-238	95.15
Pu-239	0.92
Pu-240	0.21
Pu-241	0.13
Pu-242	0.026
Am-241	0.006

the TANDEM fuelled HWR in a LOCA reaches 2.18 times nominal power and peak fuel temperature reaches 861°C (peak volumetric average fuel temperature in the high burnup zone). This may be compared with peak power of 1.6 times nominal power and a peak fuel temperature of 856°C in the case of the natural uranium equilibrium core, and peak power of 2.15 times nominal power and a peak fuel temperature of 864°C for the fresh natural uranium core.

The Indian HWR has a set of eight adjuster rods to provide xenon override. A set of four regulating rods is used for reactor regulation. In the natural uranium core, the adjuster rods have a reactivity worth of 8.2 mk, and the regulating rods have a worth of 4.8 mk. In the TANDEM fuelled core, the adjuster rod worth is 5.2 mk and the regulating rod worth is 3.2 mk. This reduced adjuster rod worth is, however, compensated for by a change in xenon worth. Whereas the 8.2 mk worth in the natural uranium core would provide a xenon override time of 30 min, the 5.2 mk worth in the TANDEM fuelled core gives an override time of about 60 min. Consequently, the decrease in adjuster rod worth to 5.2 mk is of no concern. The reduction in regulating rod worth may have some implications for operability, but with some backup worth provided by boron in the moderator, this can be managed.

The reactivity of the initial core will be very high, of the order of 200 mk. It is possible to suppress this either by loading a number of thoria bundles, or by using boron in the moderator. The boron level would have to be very high. However, the major disadvantage of boron is that power peaking in the initial core will necessitate the derating of the reactor. As core burnup proceeds, the flux will tend to flatten. By about 90 effective full power days (EFPD), both bundle power and coolant outlet temperature will have reached acceptable values. Refuelling, however, will be required only after about 900 EFPD and continued burnup will continue to flatten the power distribution, until the low flow channels located towards the periphery of the core produce more power than can be removed by the design flow for the power distribution of the natural uranium equilibrium core.

Studies have shown that while the maximum bundle power in the core continues to decrease until the core reactivity falls to zero (at about 925 EFPD), the maximum coolant outlet temperature reaches its lowest value of 295°C at 116 EFPD, and thereafter starts increasing again. At 925 EFPD, it has reached 311.9°C. The three power related constraints are bundle power, channel power and coolant outlet temperature. Figure 184 shows, as a function of EFPD, three quantities which are here termed bundle ratio (*B*), channel ratio (*C*) and temperature ratio (*T*). The bundle ratio is defined as the ratio of the highest bundle power in the core to the maximum permitted bundle power; similarly for the channel power ratio and the coolant outlet temperature ratio. For full power to be allowed, all three ratios should be less than unity. When any ratio exceeds unity, the reactor power needs to be derated by that factor in order to stay within design constraints. It is clear from Fig. 184 that whereas *B* and *C* start from values exceeding unity at zero EFPD, and decrease consistently right up to the first refuelling at 925 EFPD, *T* goes through a minimum and then rises again, reaching quite high values towards the end. The reason for this is that the TANDEM fuel is being retrofitted into a core originally designed for natural uranium, in which the coolant flows have already been fixed through appropriate orificing to suit the radial channel power distribution attributable to natural uranium fuel. Over the extremely long transition from the natural uranium core to the TANDEM core,

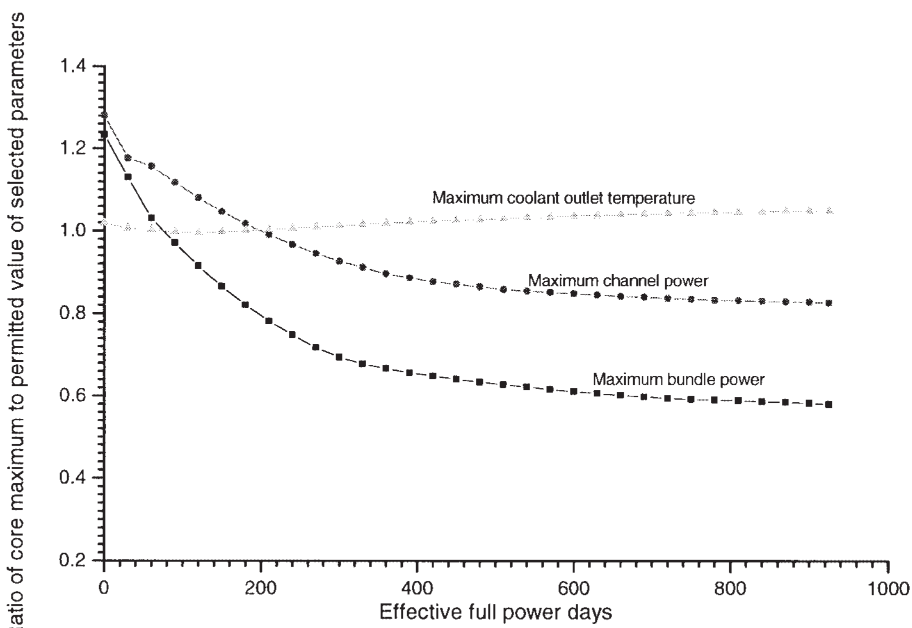


FIG. 184. Variation of power constraints with burnup.

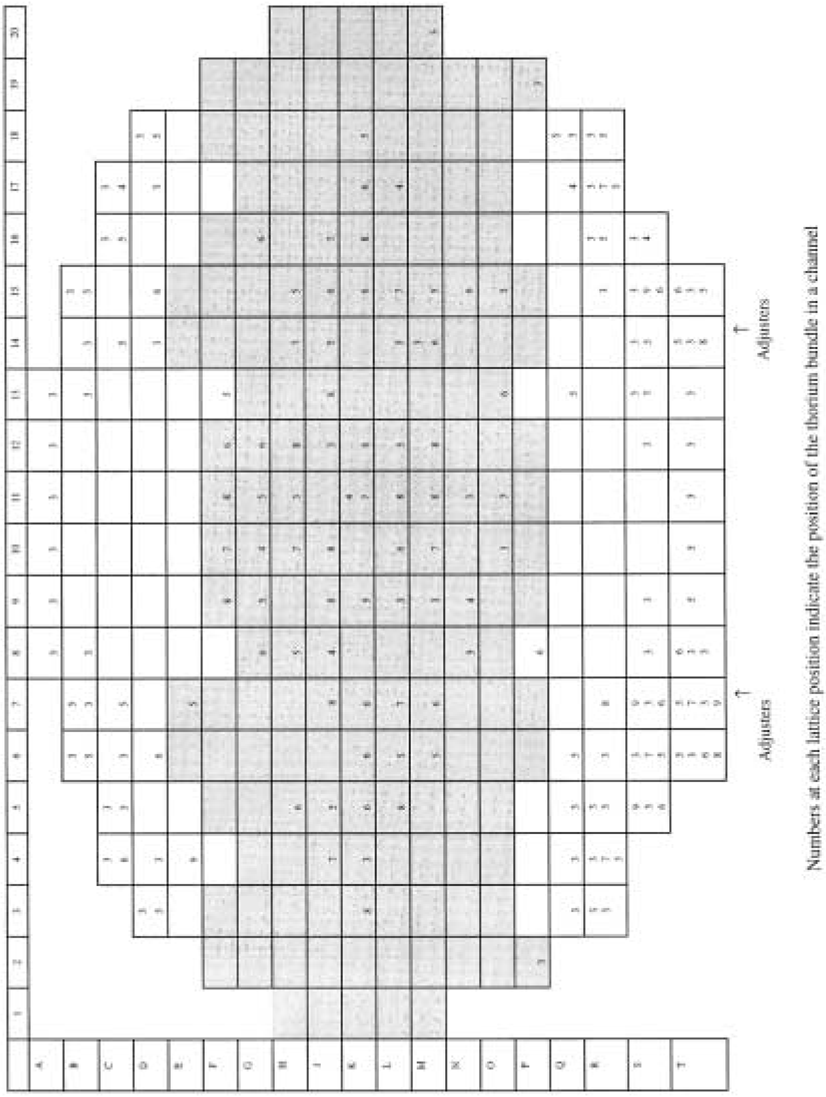


FIG. 185. Thoria bundle distribution in the TANDEM loaded core.

burnup induced power flattening reaches such proportions that the low flow channels at the core periphery are unable to remove the power.

One way to address this problem in the transition core is to use thoria fuel bundles to flatten the channel power distribution. Figure 185 shows the initial thoria loading which gives full power in the beginning. Figure 186 shows the variation of *T*

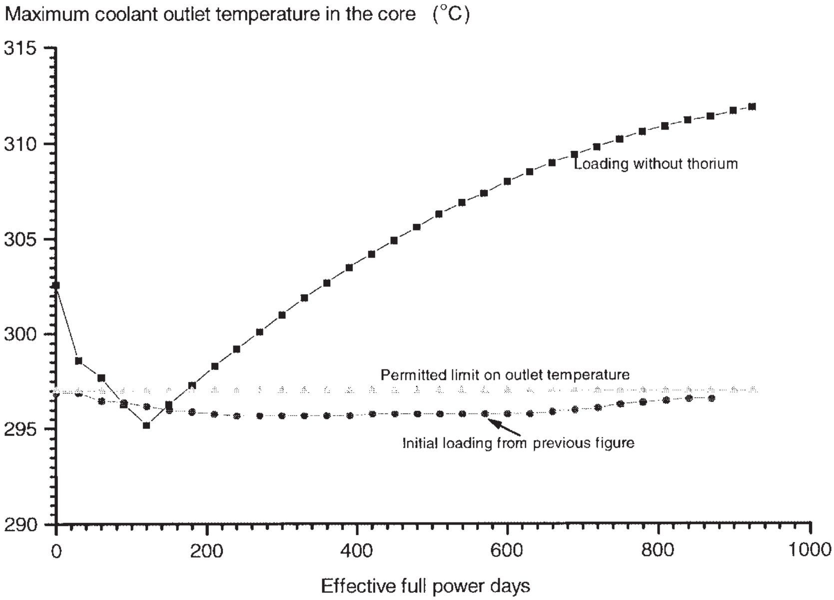


FIG. 186. Variation of maximum channel outlet temperature with burnup.

with EFPD for the core without thoria bundles, and for the core represented in Fig. 185. While in the core without thoria bundles, *T* crosses unity again at about 180 EFPD; in the core represented in Fig. 185, this happens only at 667 EFPD. Work is continuing on making further improvements, but it is clear that the use of natural ThO_2 bundles to flatten the power distribution can facilitate the transition from a natural uranium fuelled core to a TANDEM one.

6.5.4. HWR MOX with plutonium from spent PWR fuel

6.5.4.1. Introduction

The PWR/HWR recycling options discussed above are non-conventional and are based on the non-separation of the uranium and the plutonium from the fission product and minor actinide waste.

If a country has access to conventional reprocessing, then a PWR/HWR two reactor type system opens up the possibility of recycling the MOX fuel from reprocessed PWR fuel back into the HWRs. This would have potential benefits compared with recycling in PWRs. A full MOX core could be used in existing HWRs. Although MOX fuel fabrication will be much more expensive than using natural uranium, the simplicity of the HWR fuel bundle and its short length will result in cheaper MOX fuel fabrication costs compared with those of PWR MOX. In the case of uranium based fuels, CANDU fuel fabrication costs (expressed in $/kg U, excluding the costs of uranium and enrichment) are nominally lower by a factor of 5–10 than those of PWR fuel. A high burnup HWR MOX fuel, therefore, has the potential to have lower fuel cycle costs than PWR MOX. Significantly more energy could be extracted from the plutonium as MOX fuel in HWRs, compared with recycle in a PWR (Table XVII). This has important advantages in improving natural uranium utilization, reducing enrichment requirements and reducing the amount of spent fuel for ultimate disposal.

Plutonium is currently mixed with depleted uranium to form MOX fuel, which is recycled in 1/3 core PWRs in Europe. Considerable global experience has been gained in the fabrication and irradiation of MOX fuel.

6.5.4.2. AECL experience in CANDU MOX fuel fabrication

Research and development on MOX fuel containing plutonium commenced at AECL more than thirty years ago, and it still remains a strategic part of AECL's advanced fuel cycle programme [234]. Research activities, including development of MOX fuel fabrication, measurement of physical properties and production of fuel samples for experimental irradiation, have been undertaken by AECL at its Chalk River Laboratories since 1960.

In 1970, a decision was taken to remodel AECL's original plutonium laboratory and install new facilities to focus on development and demonstration of CANDU MOX fuel fabrication technology. Installation of the new facilities was complete by 1975. The facility, referred to as the Recycle Fuel Fabrication Laboratory (RFFL), is designed to produce experimental quantities of alpha active fuel such as MOX for reactor physics tests or demonstration irradiations.

(a) Fabrication campaigns at the RFFL

Following an extensive commissioning campaign using natural UO_2, a number of MOX fuel fabrication campaigns were conducted at the RFFL between 1979 and 1988, producing various quantities of fuel with different compositions. After a standby period of about five years, a project to rehabilitate the RFFL and bring it back into production was initiated in 1993. Rehabilitation and recommissioning of the RFFL were achieved in August 1996, and MOX operations were resumed in the facility with the production of 37 $(U,Pu)O_2$ bundles destined for reactor physics tests. The bundles simulated mid-burnup CANDU fuel, and were utilized in the ZED-2 zero energy reactor to validate the computer codes used to generate predictions of void reactivity in CANDU cores. This fabrication campaign was successfully completed in March 1997. The project, including both the rehabilitation and the fabrication campaigns, was funded by the CANDU Owners Group.

The various types and quantities of MOX fuel produced at the RFFL are listed in Table XLIV. The fuel elements and bundles were used for test irradiations in the NRU experimental reactor, and for physics tests in the zero power ZED-2 reactor. To date, about 5000 individual fuel elements, equivalent to over 160 bundles and containing close to 3 Mg of MOX, have been fabricated at the RFFL.

(b) Fabrication processes at the RFFL

Subject to restrictions imposed by the presence of plutonium (essentially all operations are done inside ventilated and filtered gloveboxes and fume hoods), the fabrication processes employed at the RFFL follow conventional natural UO_2 practice. Weighed amounts of the starting powders (UO_2 or ThO_2 and PuO_2) are mixed initially in the high energy mixer to produce a homogeneous ‘mastermix’, and then in a turbula blender for final blending. The blended MOX powder is prepressed into compacts, which in turn, are fed into a granulator. The resulting free flowing granules are then suitable for final pressing into green pellets using an automatic hydraulic press. The green pellets are loaded into a batch furnace, where sintering takes place under a dilute hydrogen cover gas. The sintered pellets are then centreless ground to a specified diameter and surface finish. Acceptable pellets are loaded into sheaths which are subsequently end closure welded using a TIG welding system. The sealed elements are helium leak tested, scanned for surface alpha contamination, and dimensionally inspected prior to bundle assembly.

The batch type fabrication process was designed to have a throughput of one 15 kg batch of MOX fuel per day. During the recently concluded fabrication campaign, production throughput averaged 0.6 batches per day (each batch weighing 11 kg

TABLE XLIV. FUEL FABRICATION CAMPAIGNS CONDUCTED AT THE RFFL

Experiment	Date	Fuel type	Number, type and weight of MOX elements/bundles
BDL-419	1979–1980	$(U,Pu)O_2$ @ 0.5% Pu	15 × (36 element) bundles (320 kg)
BDL-422	1981–1983	$(Th,Pu)O_2$ @ 1.75% Pu	6 × (36 element) bundles (120 kg)
BDL-430	1982	Natural ThO_2	1 × (36 element bundle) (20 kg)
WR1-1012	1982	$(Th,U)O_2$ @ 1.8% U-235	2 × (21 element bundles) (20 kg)
	1982	$(Th,Pu)O_2$ @ 2.3% Pu	2 × (21 element bundles) (20 kg)
WR1-1010	1982–1985	$(Th,Pu)O_2$ @ 2.3% Pu	1332 elements (650 kg)
BDL-432	1986–1988	$(Th,U)O_2$ @ 1.4% U-233	1350 elements (700 kg)
ZED2-96	1996–1997	$(U,Pu)O_2$ @ 0.3% Pu	37 × (37 element bundles) (810 kg)

MOX), with a peak throughput of 1.2 batches per day. Overall, 77 batches of MOX fuel totalling over 800 kg (contained in over 1370 finished fuel elements) were fabricated over a period of 26 weeks.

Standard inspection techniques were applied during the recently concluded campaign to obtain fabrication data such as immersion density of sintered pellets (95–98% of the theoretical density), microstructural grain size of sintered pellets (8–10 μm), and impurity content of the finished pellets. One inspection technique of interest is alpha autoradiography, which is used in combination with image analysis to determine plutonium particle size and distribution. It was determined that the average plutonium particle size was 20–30 μm, with a maximum of about 50 μm. Further work in correlating the information obtained from autoradiography with X ray wavelength dispersive spectrometry is continuing, in order to quantitatively determine the local plutonium concentration and provide an accurate plutonium distribution profile.

6.5.4.3. AECL irradiation experience with MOX fuel

Irradiation testing and post-irradiation examination of MOX fuel have progressed from multielement to multibundle demonstration testing of the 37 element design, and both continue to be used at AECL. Reference [235] summarizes the post-irradiation examination of several CANDU MOX bundles that had been irradiated in AECL's research reactors. This examination has confirmed the excellent performance of $(U,Pu)O_2$ bundles experiencing:

- Declining power histories from beginning-of-life powers of up to 50 kW/m to burnups of up to 50 MW·d/kg HE;
- Declining power histories from beginning-of-life powers of more than 50 kW/m to burnups of about 21 MW·d/kg HE.

In both cases, dimensional changes and fission gas release were comparable to that expected for UO_2. In general, MOX fuel performance has been excellent, and very similar to that of UO_2.

6.5.4.4. MOX fuel experience in the Fugen HWR [236]

The Fugen reactor is a prototype advanced thermal reactor, which is operated by PNC in Japan. The electrical power output is 165 MW(e). The reactor has operated successfully since 1979. Fugen is a heavy water moderated, boiling light water cooled, vertical pressure tube reactor, which uses a mixture of MOX and UO_2 fuel bundles. The core consists of between 34% and 72% MOX assemblies, although it was originally designed to use 100% MOX fuel. To date, the Fugen reactor has irradiated 658 MOX assemblies. The standard MOX fuel assembly consists of 28 elements, with a plutonium fissile enrichment of 2.0% and a design burnup of 20 MW·d/kg HE. Advanced, high burnup 37 element designs were irradiated, having target burnups of 35 MW·d/kg HE (from a plutonium fissile enrichment of 2.5%), and 40 MW·d/kg HE (from a plutonium fissile enrichment of 3.5%). These latter bundles contained four gadolinium doped UO_2 fuel elements for reactivity control. The MOX fuel performance is excellent, and provides further confirmation of its suitability for HWRs.

6.5.4.5. Indian experience with MOX fuel fabrication and irradiation

India has considered the possibility of recycling plutonium in both LWRs and HWRs (see Section 6.5.3.4). The only LWRs in India are BWRs. The first MOX pins fabricated in India were made at BARC, and were made to the specifications of the pins used in the BWR fuel assemblies. The composition of the MOX was 4.0% plutonium in natural uranium. Three clusters, each of six pins, were fabricated and irradiated in the pressurized water in-pile loop of the research reactor CIRUS, where the clusters were irradiated under power reactor conditions, reaching a burnup of ~16 MW·d/kg HE. The maximum linear element rating was 40 kW/m for the first cluster, and 50 kW/m for the two remaining clusters. All three completed their test irradiation trials without failure and their performance was deemed satisfactory. The discharged pins are now in the storage bay. Post-irradiation examination is planned, but has not yet been carried out.

Since the proposal to use MOX in the HWR is also under consideration, a cluster of MOX pins of HWR specification has also been fabricated. At present, this resides within the pressurized water in-pile loop, and it has performed satisfactorily so far.

MOX fuel fabrication has now been moved to a fabrication plant built at Tarapur. Eight MOX assemblies for the BWR were fabricated there and these are currently undergoing irradiation in the Tarapur BWR plant. Two of the assemblies have already achieved over 10 MW·d/kg HE and have performed well. More such assemblies are being manufactured at the plant.

6.5.4.6. Reactor core characteristics

All of the fuel management strategies that were discussed in Section 6.4.4.2 with respect to 1.2% SEU fuel can also be employed with MOX fuel in an HWR. In particular, chequerboard fuelling with multistage shifting and axial shuffling can accommodate MOX fuel in existing reactors. Similarly, in new reactors, adjuster rods and reactivity devices can be repositioned to facilitate the use of a simple fuel management strategy, such as a bidirectional, two bundle shift. Reference [237] describes the characteristics of a MOX core having an average discharge burnup of 21 MW·d/kg HE, and where reactivity devices have been repositioned. The axial power profiles are similar to those of 1.2% SEU with repositioned reactivity devices (see Fig. 171).

One difference between the performance of MOX fuel and that of SEU is the faster reactor dynamics obtained with MOX fuel. As already discussed with regard to DUPIC fuel, reactor dynamics are faster with plutonium based fuels because of the lower delayed neutron fraction and prompt neutron lifetime. This may require compensation in order to deal with the LOCA power pulse. One way of addressing this issue is to reduce void reactivity; as with DUPIC fuel, a small amount of neutron absorber can be added to the centre of the bundle to reduce the power pulse to a level equivalent to that obtained with natural uranium fuel.

Another difference between SEU and MOX fuel relates to the economics of optimal burnup. With SEU, the optimal burnup in an HWR is small, corresponding to a ^{235}U enrichment in the range 0.9–1.2%. With MOX fuel, the source of fissile material (plutonium) is often viewed as being available ‘free issue’. The high cost of MOX fuel fabrication provides the incentive to extract as much energy from the bundle as possible, and thereby push the burnup as high as possible.

Achieving a MOX burnup of 40 MW·d/kg HE may require a reduction in fuel operating temperatures (although a 37 element bundle was used in Fugen to achieve 40 MW·d/kg HE). Various means are available to achieve this, if required: graphite discs between pellets or an existing bundle design (37 element or CANFLEX), or even greater subdivision.

As part of the study of a CANDU ‘highly advanced core’ concept, AECL designed a 61 element bundle for use with MOX fuel (Fig. 187). Average burnup was 40 MW·d/kg HE, void reactivity was negative and the maximum linear element rating was less than 56 kW/m in a 640 channel core that produced 1250 MW(e). The plutonium isotopic composition corresponded to that of spent PWR fuel. The plutonium was mixed with depleted uranium in the outer three rings of the bundle, with average enrichments of 4.5%, 7.2% and 4.9% in rings 3, 4 and 5 respectively (the plutonium was 67% fissile). In the central element and the next ring of six elements, 15% dysprosium was mixed with depleted uranium, which resulted in negative void reactivity. Reactivity devices in the core were repositioned to optimize the axial power distribution. A typical time average axial power distribution along a channel in the centre of the core in the vicinity of the adjuster rods is shown for a simple, single bundle, bidirectional fuelling scheme (Fig. 188). All channel powers shown are arbitrarily normalized to 6500 kW. Again, this illustrates the flexibility of the HWR in accommodating a wide range of fuel types.

Another option is to optimize the HWR for the use of enriched fuels, including plutonium. Preliminary ‘scoping’ studies indicate that a burnup of 40 MW·d/kg HE could be achieved with MOX fuel having 3.5% fissile plutonium, using light water coolant and a reduced lattice pitch. Void reactivity would be negative without the use of neutron poisons in the fuel, and neutron economy would be good.

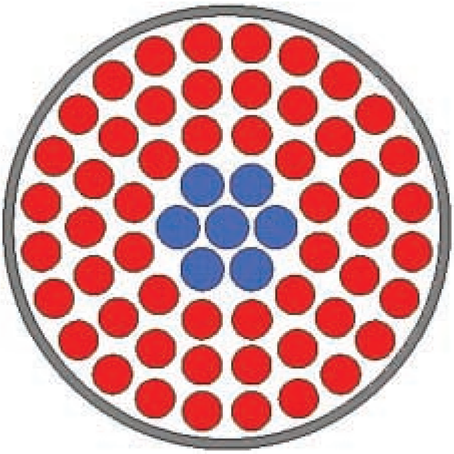


FIG. 187. *The 'highly advanced core' 61 element MOX fuel design.*

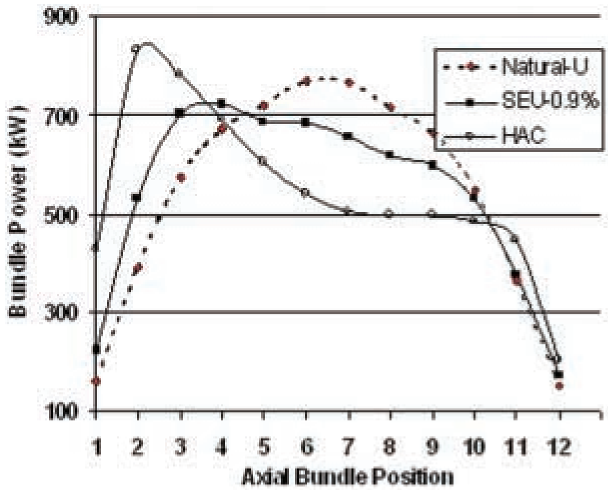


FIG. 188. Typical time average axial power profiles for the central channel for natural uranium, 0.9% SEU and high burnup MOX bundle.

6.5.5. Actinide burning in an inert matrix

6.5.5.1. Introduction

The other product of reprocessing is the minor actinide and fission product waste stream. There is worldwide interest (but particularly in Europe) in investigating the feasibility of annihilating certain long lived, carcinogenic actinides produced during normal reactor operation, and concentrated by reprocessing of the fuel.[5] The intent is to make an inert matrix fuel incorporating these actinides (^{237}Np, ^{241}Am, ^{244}Cm), together with some of the plutonium that is separated during reprocessing. The plutonium provides a major fissile component and the neptunium, americium and curium are transmuted or fissioned to shorter lived, less carcinogenic nuclides. Without ^{238}U in the fuel matrix, further production of these actinides does not occur. The CANDU reactor is particularly suited for this application because no major modifications to reactor design are required and because the neutron economy of CANDU reactors means the annihilation of plutonium/actinide waste can be more complete. This application would utilize the third product of reprocessing (the others being recycled uranium and plutonium), and is yet another manifestation of the synergism between CANDU reactors and PWRs.

Studies have also shown the effectiveness of burning some of the fission products in the high thermal flux of the moderator in a CANDU reactor [239].

6.5.5.2. Fuel design, fabrication and performance

(a) Programmes and candidate materials [240]

The major European effort comprises a multinational programme entitled Experimental Feasibility of Targets for Transmutation (EFTTRA), which is directed towards actinide waste annihilation [241]. The French fast reactor programmes, Concept to Amplify Plutonium Reduction in Advanced LMFRs (CAPRA) and SPIN, address the annihilation of plutonium and the recycling and burning of minor actinides, and include fast reactor irradiations of candidate materials [242]. Japan has a programme entitled Plutonium Rock-like Fuel In-Reactor Technology (PROFIT), which is directed towards the annihilation of military plutonium (particularly Russian military plutonium) [243]. The idea is to use a combination of materials to produce a multiphase inert matrix material that will resemble ordinary

[5] In the case of deep geological disposal of spent fuel in an underground repository having a reducing environment, Canadian assessments have shown that the radiological risk from the actinides is negligible [238].

rock, even after irradiation, and which will thus facilitate underground burial as a waste.

The US programme on inert matrix material development, which is smaller than either EFTTRA or PROFIT, focuses on the use of zirconia (ZrO_2) [244]. Zirconia is one of the few materials that has already been shown to behave acceptably as a fuel material (with the addition of ^{235}U). Its major disadvantage is its poor thermal conductivity (slightly lower than UO_2), which means that fuel temperatures will be as high as, or higher than, those in present-day fuels. Other researchers are searching for an inert matrix material that, besides having the ability to function acceptably as fuel host in-reactor, has a higher thermal conductivity than UO_2, which means so that fuel temperatures will be lower, fission gas release from the fuel will be lower, and safety margins will be increased.

Many papers have been written on candidate materials and the properties that make them suitable as inert matrix candidates. Reference [245] presents information on the fabrication, thermal properties and stability with regard to alpha radiation on five materials: $MgAl_2O_4$ (spinel), $ZrSiO_4$ (zircon), CeO_2 (ceria), SiC and Si_3N_4. A number of other papers have been published which deal with various aspects of, and choices for, inert matrix fuels [241, 243, 244, 246, 247].

(b) AECL's programme

Candidate inert matrix materials that have been assessed by AECL include $ZrSiO_4$, $MgAl_2O_4$, CeO_2, $CePO_4$, ZrO_2 doped with calcium, cerium, erbium or yttrium, and SiC. Ceria has the same crystal structure as UO_2, and therefore was thought likely to behave well in-reactor. Zircon is a stable mineral found naturally in the earth's crust, and therefore at least should be a stable waste form. It is being evaluated as a candidate for hosting military plutonium for direct burial in the ground. Cerium phosphate is a candidate supplied to AECL from the Netherlands (via the Transuranium Institute in Germany). Silicon carbide stood out early as a promising candidate material because of its high melting temperature and very high thermal conductivity, and because of its known resistance to attack by many corrosive agents, including oxygen, even at high temperatures. As no other laboratory in the world was examining SiC for this purpose, AECL focused its efforts on this application, although the TANDEM accelerator studies described in the next paragraph were used for investigating many candidate materials. The overall programme was aimed at examining all issues that would be important in the selection of SiC as an inert matrix fuel candidate.

Owing to the large number of candidate materials, the TANDEM accelerator was used to simulate in-reactor conditions and thereby serve as a first screening for candidate materials that might behave well as fuels. The critical unknown issue for candidate materials concerns their behaviour in-reactor in the face of severe damage inflicted by fission fragments blasting through the matrix at energies greater than 70 MeV, in particular whether they would become amorphous or undergo an unacceptable volume change. Producing fuel from each candidate material for in-reactor tests is expensive, particularly in the case of plutonium, and therefore the TANDEM accelerator at AECL was used to bombard the candidate materials with a beam of 72 MeV iodine ions. This beam substitutes for fission fragments, and the bombarded areas were examined for damage, especially swelling. To increase the number of candidate materials available for testing, and to foster international collaboration, AECL tested candidate materials from the countries of the European Union and the USA, as well as domestic materials.

Besides these accelerator simulation tests, the programme on SiC consisted of assessing:

- Reactor physics aspects of SiC fuels in CANDU reactors;
- Fabrication issues associated with SiC — high sintering temperatures, plutonium handling, and characterization of microstructures that form with the addition of plutonium and several sintering aids;
- Compatibility of SiC with water under reactor coolant conditions and with groundwater, the latter with a view to eventual storage;
- Compatibility of SiC with zirconium based sheath materials under accident temperatures;
- Waste disposal aspects.

(c) TANDEM accelerator tests

Since preliminary results were reported in 1995, AECL has exhaustively tested many candidate materials: $ZrSiO_4$, $MgAl_2O_4$, CeO_2, $CePO_4$, ZrO_2 doped with calcium, cerium, erbium or yttrium, as well as SiC, over temperatures ranging from ambient to 1200°C, and over a wide range of doses (10^{14}–10^{17} ions/cm^2) [248].[6] After accelerator bombardment of each sample, the surface relief of the 3 mm diameter beam spot was measured. Significant height of the spot above the original surface was taken as an indication that in-reactor swelling would occur. In general, results were not strongly dependent on dose—if a candidate material showed swelling at high dose, it also showed swelling at low dose. The results were also reasonably independent of temperature. The materials that showed the best results (i.e. least swelling) were SiC and ZrO_2 with any of its dopants. These candidate materials did not show *any* swelling, i.e. laser profilometry could not detect any surface relief at the

[6] 10^{17} ions/cm^2 is approximately equivalent to a CANDU fuel burnup of 20 MW·h/kg U.

implantation spots. Ceria turned out to be one of the poorest candidate materials on the basis of these tests. All other candidate materials showed some surface relief, implying that swelling could be expected when used in-reactor as a fuel.

(d) Manufacture

Pressureless sintering of SiC normally requires temperatures of 1900°C or higher: AECL has been working with Queen's University, Ontario, to develop methods of fabrication at lower temperatures, ones which are more representative of UO_2 sintering. In addition, fabrication tests were performed with cerium as an additive; cerium being generally considered, worldwide, to be a reasonable chemical substitute for plutonium.

Good progress has been made in establishing methods of fabricating SiC containing a wide range of cerium concentrations [249]. The methods are similar to those in current use for UO_2 fuel production. High densities have been achieved and the required sintering temperatures have been reduced from over 2000°C to approximately 1700°C by judicious use of sintering additives. In fact, ceria itself has proven to be a most effective sintering additive. High density material was fabricated without using cold isostatic pressing, a step not used in UO_2 fuel production. Cerium is usually incorporated into the SiC as an $AlCeO_3$ phase, often as micron sized particles, but also as larger particles (6–8 μm) under some fabrication conditions. Above 1830°C sintering temperature, with Al_2O_3 and Y_2O_3 sintering aids, X ray diffraction did not reveal the evolution of any phases except SiC. This suggests that all additives are taken into solid solution, which is not understood.

Measurements were taken of the specific heat capacity and thermal conductivity of SiC specimens fabricated in this programme, and although the thermal conductivity is lower than for pure, fully dense SiC, it remains sufficiently high that the central temperature in an operating fuel at 55 kW/m would only be about 100°C higher than the coolant temperature.

In summary, SiC containing cerium (plutonium substitute) and a few weight per cent of sintering additives was fabricated using equipment that is compatible with conventional UO_2 or MOX fabrication technology. The thermal conductivity of such SiC based inert matrix fuels is very high — a clear benefit from the perspective of in-reactor fuel performance and safety.

(e) Compatibility with water

Previous results have shown that SiC has negligible interaction with water under typical CANDU coolant conditions, i.e. 300°C and pH of 10.3 [248]. To determine the robustness of SiC, tests were performed under stagnant acidic conditions (pH of 3 and 300°C), for a period of 32 d, even though it is not expected that CANDU coolant would ever exist under such conditions.

Eleven samples, weighing between 75 mg and 400 mg were tested. Some interactions with the water occurred; a grey, visible oxide layer formed and some samples broke into two or more pieces. Weight changes were small, however; samples decreasing in weight by 1–1.5 mg.

Additional tests were performed to examine the stability of SiC as a waste under attack from groundwater. The SiC samples used were ceria doped, also with an alumina sintering aid, prepared at Queen's University. The 20 mL water sample was at a pH of 2.9 and a temperature of 96°C, and the test duration was 119 h. It should be noted that groundwater in a fuel disposal vault is not expected to have such a low pH, but this serves as a check of behaviour under 'extreme' conditions. As chemical attack at these low temperatures was so small, measurement of weight loss was not used; rather, the chemistry of the water was examined for dissolved components, and the surface of the sample examined by X ray photoelectron spectroscopy.

After the test, the water analysis indicated the following concentrations: 0.53 mg/L Al, 11.8 mg/L Ce and 11.3 mg/L Si. Since the water volume was 20 mL, total dissolved amounts were 0.011 mg Al, 0.24 mg Ce and 0.23 mg Si. A similar test with UO_2 has not been performed under these conditions, therefore direct comparison is not possible at present. However, these amounts are very low, especially considering the high acidity level of the water, and indicate that dissolution rates are not likely to be a problem for SiC waste. The X ray photoelectron spectroscopic analysis of the surface (2–3 monolayers) showed one change — the concentration of cerium was lower in the test sample than in an unused sample.

(f) High temperature chemical reactivity between SiC and Zircaloy 4

Tests were performed by holding polished discs of SiC and Zircaloy 4 under light pressure and in an argon atmosphere in a molybdenum cell at temperatures of 1000°C, 1500°C and 1700°C.

The temperatures were maintained for 1 h at 1000°C and 1500°C, and for 15 min at 1700°C. Two types of SiC specimen, fabricated at Queen's University, were used; both contained alumina as a sintering aid, and one contained titania, the other ceria. Extra carbon had been added to all specimens during sintering to ensure that no free silicon was present in the final products. After cooling and sectioning, the specimens were examined by optical and scanning electron microscopy to study the extent of interaction.

Although no significant interaction occurred at 1000°C, at 1500°C there was clear evidence of a diffusion based reaction to form ZrC and free silicon. Diffusion of free silicon into the Zircaloy disc led to the formation of a molten zirconium–silicon rich eutectic phase. This reaction was more pronounced in the 1700°C test specimens, where the amount of molten eutectic phase was sufficient to cause partial dissolution of the molybdenum cell side walls.

The test results indicate that formation of a molten zirconium rich phase could conceivably occur during a hypothetical reactor accident at temperatures significantly lower than the melting point of unoxidized Zircaloy cladding (1760°C). This possibility is of concern because it would allow further accelerated attack on the SiC carrier material, thereby allowing any fission products that had exsolved from the plutonium phase to be released. However, UO_2 interaction with Zircaloy begins at about 1200°C, and it dissolves into molten Zircaloy at 1760°C [250]. The rate and extent of UO_2–Zircaloy interaction depends on the amount of oxygen present. However, it appears that the SiC interaction with the sheath is, at least, no worse than the UO_2 interaction. Tests using SiC pellets clad with Zircaloy 4 sheathing, for more direct comparison, are being evaluated.

(g) Waste management considerations

Considerations to be borne in mind in using SiC based fuel as a waste form can be summarized:

- The SiC matrix itself, with sintering aids used to date, will not generate long lived activation products in-reactor. Care needs to be taken to minimize the impurity levels of nitrogen and chlorine to eliminate concerns about their activation products ^{14}C and ^{36}Cl.
- Silicon carbide itself is extremely stable and resistant to corrosion. Its performance as a waste form will depend on its capability to encapsulate and retain high burnup plutonium containing particles. It is likely that the plutonium rich phase will remain encapsulated, and hence be protected from leaching. This needs to be verified by post-irradiation examination and leach testing of irradiated inert matrix fuel materials.

(h) Conclusion

The practical realization of an inert matrix fuel for burning plutonium or actinide waste is ten to twenty years off, at least partly because there is no urgency. Silicon carbide and doped ZrO_2 are the two candidate materials that most clearly pass the accelerator simulation test, but tests conducted in-reactor are the final determinants. Compatibility of SiC with water does not seem to be an issue, either as a fuel for compatibility with coolant or as a waste form for compatibility with groundwater. The compatibility of SiC with Zircaloy does not seem to be worse than that of SiC with UO_2–Zircaloy interactions. A small amount of melt phase above 1700°C may be acceptable in safety evaluations. A concern for the actinide burning application may be due to SiC being a carbide rather than an oxide, and americium carbide, which could form when the actinide waste is incorporated in the SiC, is as yet unstudied. Mounting a programme to investigate it will be expensive because americium is highly alpha active. Americium oxide, on the contrary, has already been studied. Silicon carbide appears to be an excellent candidate material to use for plutonium burning, and for actinide burning if the americium carbide issue can be resolved.

6.5.5.3. Reactor characteristics [251]

Detailed full-core fuel management studies were conducted using the RFSP finite reactor code to assess the power distributions and reactivity effects of fuelling with an inert-matrix fuel containing a mixture of plutonium and minor actinides in an existing CANDU 6 reactor [213]. The lattice code WIMS-AECL was used to determine the optimum amounts of plutonium and actinides in the bundle using the CANFLEX fuel bundle design [170].

Figure 189 shows the RFSP model of a CANDU 6 reactor, divided into three fuelling regions: inner, outer and periphery. The inner and the outer regions are further subdivided into smaller zones having slightly different fuelling rates in order to optimize the reactor global power shape.

Figure 190 shows the configuration of the fuel elements in a CANFLEX fuel bundle. Plutonium and minor actinides are restricted to the 35 elements in the two outer fuel rings. The eight inner elements contain only SiC and a small amount of gadolinium.

Gadolinium is used to suppress the excessively high reactivity of the fresh fuel. Coolant void reactivity is negative. There is a high degree of flexibility in choosing the amounts and relative fractions of plutonium and minor actinides in the outer elements, and in choosing the amount of gadolinium in the inner elements in order to achieve specific goals, such as reactor power distribution, coolant void reactivity, and plutonium and actinide destruction efficiencies. The combination of on-power fuelling with a simple and flexible fuel bundle design offers many possibilities for actinide burning in existing CANDU reactors.

For the burning of plutonium and minor actinides obtained from the reprocessing of spent LWR fuel, the chosen composition consists of 356 g of plutonium plus 66 g of minor actinides in the two outer fuel rings and 20 g of gadolinium in the two inner rings. The central element contains 10 g of gadolinium and the remaining 10 g of gadolinium is distributed uniformly over the seven elements in the next ring. Each of the two outer fuel rings contains 178 g of plutonium and 33 g of minor actinides mixed with SiC. The plutonium and minor actinide content per element in the outermost fuel ring is two thirds that of the third fuel ring. The ratio of minor actinides to plutonium is about 50% greater than that in the unadjusted spent PWR fuel. The composition corresponds approximately to spent

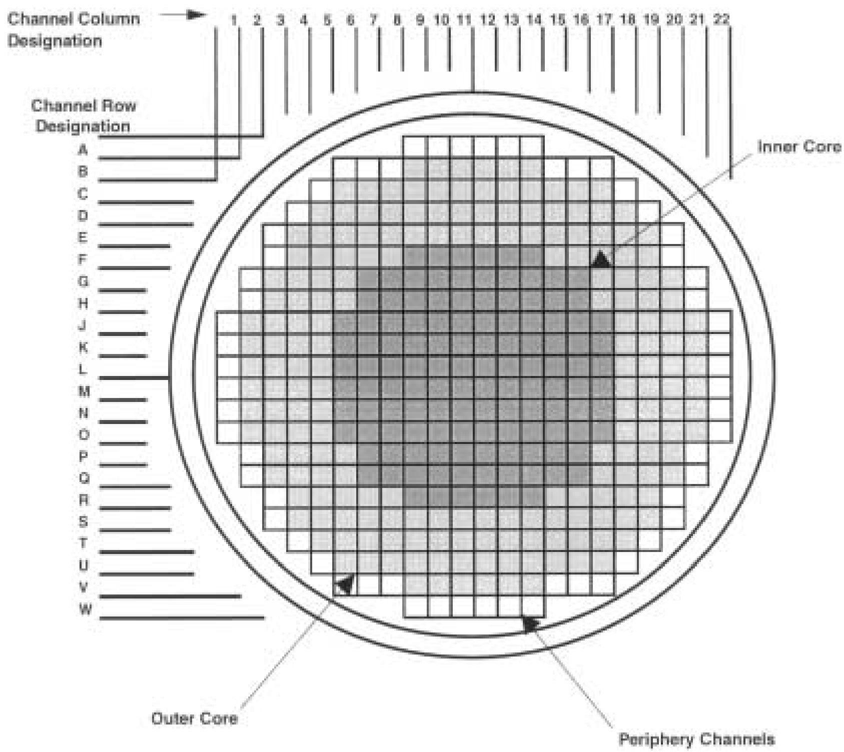


FIG. 189. Reactor core model of a CANDU 6.

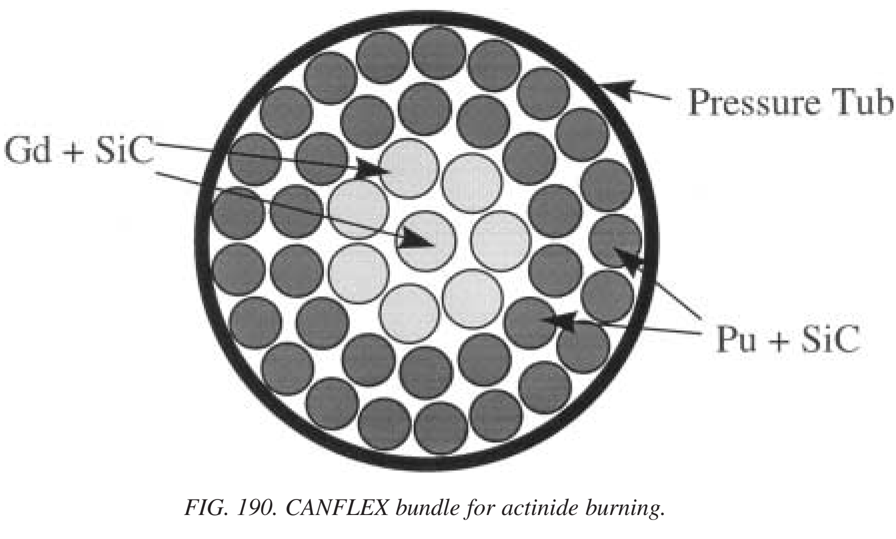


FIG. 190. CANFLEX bundle for actinide burning.

PWR fuel having an initial enrichment of 3.5% and a burnup of ~35 MW·d/kg HE. The initial material composition of the civilian (plutonium/actinide)–SiC fuel bundle is shown in Table XLV.

(a) Lattice reactivity of the civilian (plutonium/actinide)–SiC fuel

Figure 191 shows the variation of the lattice *k*-infinity as a function of the total energy produced in the civilian (plutonium/actinide)–SiC bundle, with and without gadolinium. Although natural UO_2 fuel is not used in this study, its *k*-infinity versus bundle burnup characteristics are also shown in Fig. 191 for comparison purposes. In this application, the energy output from the plutonium and the minor actinides is an important parameter, and only the minimum amount of gadolinium is used to suppress the initial reactivity. The reactivity of the fuel is suppressed by adding a relatively large portion of minor actinides to the plutonium–actinide fuel mix.

Coolant void reactivity is –4 mk for the civilian (plutonium/actinide)–SiC option. The power coefficient is also negative. There would be no power pulse in a postulated LOCA, and the safety and licensing analyses would be greatly simplified. The fuel temperature coefficient is very slightly positive, about 12 μk/°C. However,

TABLE XLV. INITIAL COMPOSITIONS OF CIVILIAN PLUTONIUM–ACTINIDE INERT MATRIX FUEL IN A CANFLEX BUNDLE

Isotope	Composition (g)			
	Ring 1 (1 pin)	Ring 2 (7 pins)	Ring 3 (14 pins)	Ring 4 (21 pins)
Pu-238			3.0	3.0
Pu-239			102.5	102.5
Pu-240			47.8	47.8
Pu-241			15.5	15.5
Pu-242			9.3	9.3
Total Pu			178.1	178.1
Am-241			15.37	15.37
Am-243			2.65	2.65
Np-237			14.98	14.98
Total minor actinides			33.00	33.00
Si (natural)	139.64	977.42	1399.01	2098.54
C (natural)	59.84	418.90	599.58	899.38
Gd (natural)	10.0	10.0		

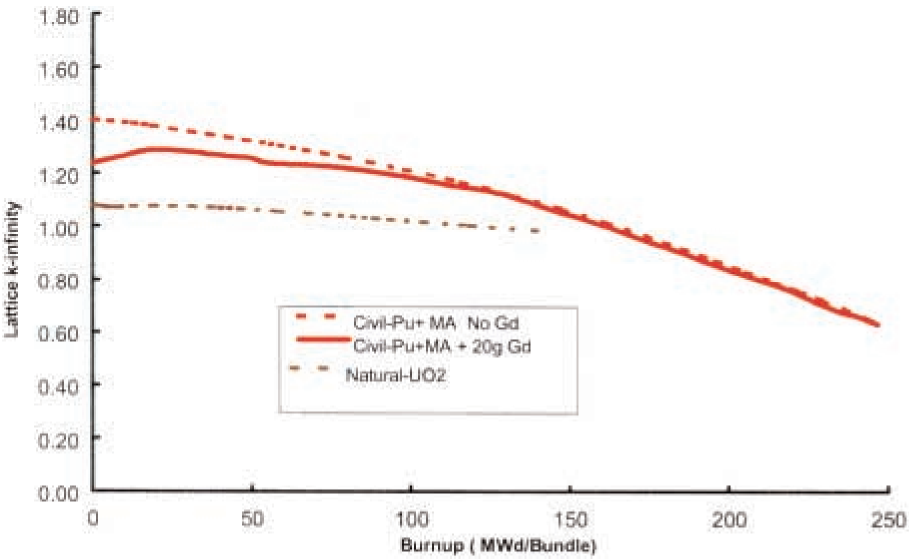


FIG. 191. Lattice k-infinity versus fuel burnup.

the sign of the fuel temperature coefficient is not relevant with this fuel: an increase in fuel power would result in a very small increase in fuel temperature because any increase in heat in the fuel would be immediately transferred to the coolant as a result of the high thermal conductivity of the SiC. The increase in coolant temperature would reduce the coolant density and produce a negative reactivity feedback owing to the negative coolant void reactivity effect.

(b) Full-core fuel management simulations

Detailed, realistic fuel management simulations were performed using RFSP for a standard CANDU 6 reactor employing civilian (plutonium/actinide)–SiC fuel in the whole core. Acceptable bundle and channel powers were obtained for both cases using a bidirectional, two bundle shift refuelling scheme. The refuelling rate is 9.5 bundles per full power day. Satisfying these fuelling requirements is easily within the capability of the current fuel handling system.

Figure 192 shows that the time average channel power distributions are very similar to those in the present natural UO_2 fuelled CANDU reactor. Figure 193 shows the time average bundle power distributions in a central channel, normalized to a channel power of 6.5 MW. Bundle powers are high at the inlet end for the civilian (plutonium/actinide)–SiC fuel because of the rapid depletion of the fissile plutonium.

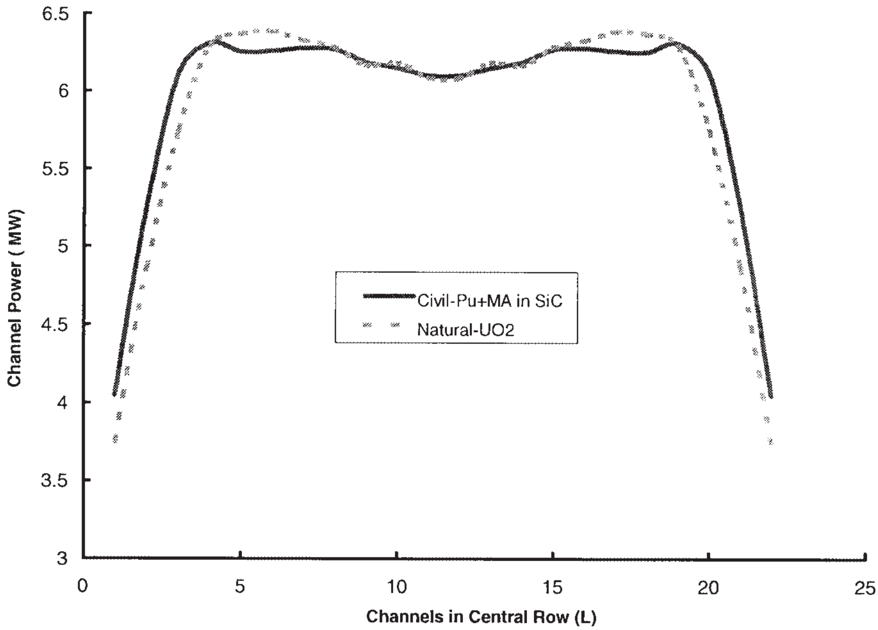


FIG. 192. Radial power distribution.

This axial power shape, which peaks towards the inlet (refuelling) end of the channel, will also result in improved thermohydraulic performance (higher critical channel powers). The maximum instantaneous channel power is less than 7.1 MW, and the maximum instantaneous bundle power is less than 990 kW. The fuel temperatures would be very low, even at these relatively high bundle powers, because the thermal conductivity of the SiC matrix is many times greater than that of UO_2. In fact, with a linear element rating of 55 kW/m, the peak centre line fuel temperature would only be ~100°C above coolant temperature.

(c) Destruction efficiency of plutonium and minor actinides

Table XLVI shows the plutonium and minor actinide content in fresh and discharged civilian (plutonium/actinide)–SiC fuel bundles. The discharge fuel burnup is 229 MW·d per bundle. In comparison, the fuel burnup of a natural uranium CANDU fuel bundle, which contains 133 g of ^{235}U, is about 136 MW·d per bundle.

The civilian (plutonium/actinide)–SiC option gives a net destruction rate of 1.2 Mg of plutonium and 0.13 Mg of minor actinide per GW(e)·a. About 97% of the

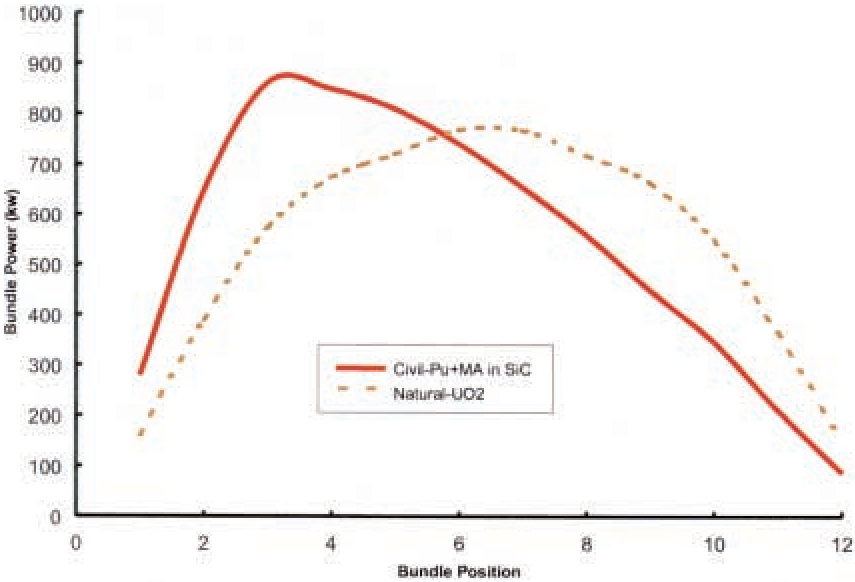


FIG. 193. Axial power distribution in a central channel.

TABLE XLVI. PLUTONIUM AND MINOR ACTINIDE CONTENT IN (PLUTONIUM/ACTINIDE)–SiC BUNDLES

Isotope	Civilian (plutonium/actinide)–SiC fuel content		
	Fresh (g/bundle)	Discharge (229 MW·d) (g/bundle)	Destruction (%)
Pu-238	5.9	5.5	6.8
Pu-239	205.0	6.2	97.0
Pu-240	95.6	57.5	39.9
Pu-241	31.0	15.8	49.0
Pu-242	18.5	41.5	
Total Pu	356.0	126.5	64.5
Np-237	29.96	14.76	49.3
Am-241	30.73	3.69	88.0
Am-243	5.30	9.94	
Cm-242		8.58	
Cm-244		3.96	
Total minor actinides	65.99	40.93	38.0

initial ^{239}Pu and 88% of the initial ^{241}Am are destroyed; the net destruction efficiency of the minor actinides is 38%. LWRs produce about 268 kg of plutonium and about 33 kg of minor actinides per GW(e)·a. One CANDU 6 reactor can effectively destroy much of the plutonium and minor actinides produced in three 1 GW(e) size LWRs.

There is scope for further optimization of the civilian (plutonium/actinide)–SiC option. All or some of the minor actinides could be segregated into elements in the centre of the bundle. It is conceivable that after the first pass through the reactor, these separate elements are removed and irradiated in the moderator (e.g. in the adjuster rod units) for an extended period, significantly increasing the net destruction efficiency of the minor actinides. These and other options will be the subject of further studies.

6.6. HWR MOX WITH PLUTONIUM FROM SPENT HWR NATURAL URANIUM FUEL

Resource and economic considerations are prime drivers in decisions taken on recycling. The availability and cost of fissile material (starting with natural uranium), the cost of processing (enrichment), the cost of fissile material recovery (reprocessing) and the cost of fuel fabrication are all determinants in the decision to recycle.

If spent fuel is considered to represent a mine of fissile material, then the spent natural uranium HWR 'ore' is dilute. The ^{235}U concentration in the spent fuel is at the same level as that of depleted uranium enrichment tails (~0.2%), therefore, there is currently no economic incentive for its recovery. The fissile plutonium is also dilute, typically 2.6 g/initial kg U. In contrast, in spent PWR fuel, depending on the initial enrichment and discharge burnup, the ^{235}U concentration is around 9 g/initial kg U, while the concentration of fissile plutonium is ~6 g/initial kg U. As the cost of recovery is dependent on the concentration of the fissile material (or on the amount of material that has to be processed), then clearly, spent PWR fuel will be a cheaper mine of fissile material than spent natural uranium HWR fuel.

Hence, the cost of its recovery would not warrant, from an economic perspective, plutonium recycle from spent natural uranium fuel for the foreseeable future. Neither would waste disposal considerations change this conclusion, as the geological disposal of spent HWR fuel has been shown to be both technically and environmentally sound, and the disposal of reprocessing wastes does not have any inherent advantages over the disposal of spent HWR fuel [150]. Moreover, it is important to establish that on any grounds — technical, economic, social, political or environmental — there is a viable and acceptable solution to the permanent disposal of spent fuel. This will most likely require the disposal of at least some of the stockpile of spent HWR fuel in order to establish confidence that the disposal of nuclear waste, which is a major issue with regard to the public acceptance of nuclear power, is adequately addressed.

Nonetheless, in some countries, such as India, a high value is placed on energy independence and security of supply, and the recycle of plutonium from spent HWR fuel has strategic importance, including provision of a source of fissile material for initiating the thorium cycle.

6.6.1. India: Steady state characteristics of MOX in HWR

With a view to making a slow, careful beginning aimed at acquiring experience in the use of plutonium fuel in the HWR, the first MOX design that is being considered uses plutonium (mixed with natural uranium) only in the inner 7 pins of the 19 rod cluster. The plutonium content is 0.4%. As long as this is the enrichment considered, the fuel design will not be changed. The discharge burnup achievable with this enrichment is 10.5 MW·d/kg.

The plutonium considered is that extracted from spent natural uranium fuel discharged from the HWR. The discharge irradiation ranges from 5 MW·d/kg HE to 10 MW·d/kg HE, occasionally declining to 4 MW·d/kg HE or rising to 13 MW·d/kg HE. The average burnup is about 6.2 MW·d/kg HE. The isotopic composition of the plutonium has been taken to be: ^{239}Pu (68.79%), ^{240}Pu (24.60%), ^{241}Pu (5.26%) and ^{242}Pu (1.35%).

With the addition of 0.4% plutonium in 7 pins out of 19, the discharge burnup increases from 6.2 MW·d/kg HE to 10.5 MW·d/kg HE. This will reduce the natural uranium requirement of the Indian HWR by about 27%.

Studies carried out with higher plutonium contents have shown greater uranium utilization. A composition of 1% plutonium in natural uranium was calculated to have a discharge burnup of 17.6 MW·d/kg HE, and to reduce the natural uranium requirements by 39%. Recycling the uranium from the reprocessed HWR fuel along with the plutonium gave still better savings — 1.3% plutonium with recycled uranium had a discharge burnup of 12.1 MW·d/kg HE and a uranium saving of 42%. The significantly lower discharge burnup would, of course, be reflected in higher fuelling cost.

Since the present emphasis is on the use of plutonium in the BWR, the work on HWR MOX is being pursued at a slower rate. In the beginning, only a few of the bundles described above, which has been named MOX-7, will be introduced. No major changes in the kinetic response of the reactor are expected with 20–25 bundles in the core. The issue which has been examined in the greatest detail is the transition of the natural uranium equilibrium core to a MOX loaded core.

In this study of the transition, some channels were fuelled with unenriched thoria, the idea being that this would assist in maintaining a desirable power distribution. During the transition stage, the following constraints were assumed:

- The bundle powers in the core should not exceed the value of 420 kW for the natural uranium and thoria bundles, and 384 kW for the MOX bundles.
- The maximum coolant outlet temperature should not exceed 297°C.
- The power ramps experienced by adjacent channels during refuelling operations should be within the limits imposed on the bundle, which in turn is a function of the bundle burnup.
- When a channel is loaded with thoria, the entire channel is converted to thoria.
- In the the case of other channels, an eight bundle shift fuelling scheme is followed.
- When a natural uranium channel is fuelled with MOX, the first refuelling introduces eight MOX bundles. The next time it is refuelled, another eight bundles of MOX will be introduced in order that the channel will be fully MOX. Thereafter, this channel will only be refuelled with MOX bundles.

In the transition scheme, the best strategy appeared to be to begin by converting the peripheral channels to MOX. Thereafter, the MOX region slowly expanded to encompass more and more of the continually shrinking central region of natural uranium channels. Figure 194 shows an intermediate stage in the transition.

6.7. HWR MOX FUEL FOR EX-WEAPONS PLUTONIUM DISPOSITIONING

6.7.1. Introduction

Disarmament efforts in both the USA and the Russian Federation have resulted in the creation of large inventories of weapons derived plutonium that require dispositioning. One option being followed involves the incorporation of plutonium into MOX fuel and its subsequent use as fuel in commercial power reactors. The use of CANDU power reactors currently operated by Ontario Power Generation is an option that is being considered as a mean of supplementing the plutonium dispositioning programmes being undertaken in the Russian Federation and the USA [252, 253]. The feasibility of the CANDU MOX fuel option was established in a study sponsored by the US Department of Energy. This feasibility study covered technical and strategic issues, schedule, and cost related parameters, with the objective of identifying an arrangement permitting consumption of 50 Mg of weapons derived plutonium as MOX fuel in CANDU reactors at the earliest date. Among the study's major conclusions are the following:

- Ontario Power Generation's Bruce A generating station is suited for this application because of its core design, its base load operating mode, and its existing safeguards and security infrastructures.

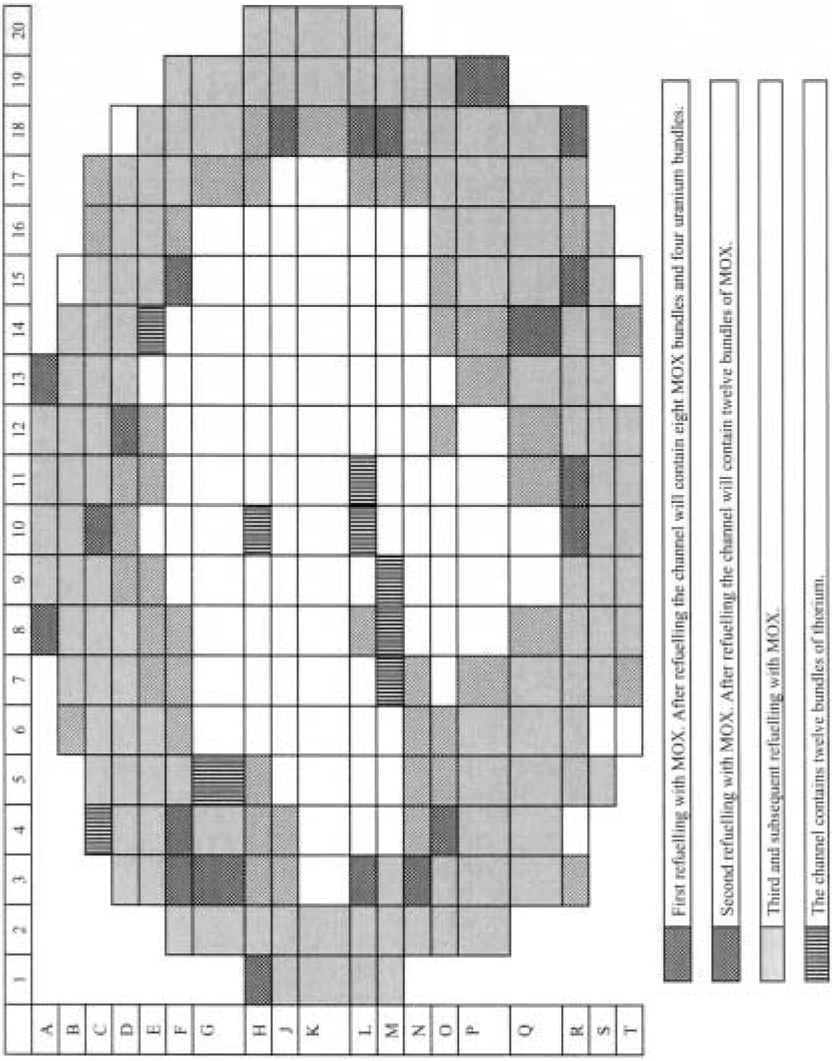


FIG. 194. Natural uranium to MOX conversion.

- To consume 50 Mg of plutonium in 25 years, the standard CANDU 37 element fuel bundle was used as the reference design. The outer 30 elements contain MOX fuel while the inner 7 elements contain dysprosium doped UO_2, which compensates for the extra reactivity due to the plutonium. This fuel would operate within the same burnup and power rating envelope as standard CANDU UO_2 fuel.
- The fuel would also possess nuclear parameters that allow the reactor to operate within its existing licensing envelope. No changes to the existing reactor system, other than for the provision of safe and secure storage of new fuel, are required.
- Instead of necessitating the mining and refining over 6000 Mg of uranium ore annually, the MOX fuel cycle consumes 2 Mg of plutonium and 250 Mg of depleted uranium (a by-product of enrichment plants). The resulting quantity of spent fuel is about 10% less, owing to the slightly higher average burnup which can be achieved with the full MOX core.

Overall, this study identified practical and safe options for the consumption of plutonium as MOX fuel in existing CANDU reactors. By designing the fuel and nuclear performance to lie within existing experience and safety and licensing envelopes, and by utilizing existing fuel fabrication and transportation facilities and methods, a low cost, low risk option for long term plutonium dispositioning/utilization has been developed. More detailed discussion of the major findings of the study are given in the following sections.

6.7.2. Conventional CANDU MOX fuel designs for dispositioning ex-weapons plutonium

The conventional MOX fuel is a near term, technically achievable, safe and economic option for the disposition of weapons derived plutonium. The main objective is not to destroy the plutonium, but to convert it to spent fuel, a form that has a high degree of diversion resistance, while producing electricity. The flexibility of the CANDU reactor allows the choice of several variants of this option, depending on the requirements and priorities. The important considerations are the timeliness of the deployment option, the plutonium disposition rate and the economics.

The first MOX option was optimized so as to minimize the implementation time. The reference fuel uses the standard 37 element geometry and is designed to perform within the operating and safety envelopes for natural uranium fuel. Depleted uranium (0.2 wt% ^{235}U) is the matrix material throughout the bundle. In the central element, and in the next ring of six elements, 5% dysprosium (a burnable poison) is mixed with the depleted uranium. Plutonium is confined to the outer two rings of fuel: 2.0% plutonium in the third ring of 12 elements, and 1.2% plutonium in the outer ring of 18 elements. The bundle average burnup of the reference MOX fuel is 9.7

TABLE XLVII. FUEL COMPOSITION: CANDU MOX OPTIONS

Nuclide	Option 1		Option 2		Option 3	
	0.0 MW·d/kg (g/bundle)	9.7 MW·d/kg (g/bundle)	0.0 MW·d/kg (g/bundle)	10.0 MW·d/kg (g/bundle)	0.0 MW·d/kg (g/bundle)	25.0 MW·d/kg (g/bundle)
Pu-239	218.0	78.8	313.8	172.9	443.4	106.8
Pu-240	13.5	57.6	19.4	69.3	27.4	108.1
Pu-241	0.3	13.2	1.2	13.1	1.7	25.2
Pu-242	0.1	3.8	1.3	2.9	0.1	13.0
Total	231.9	153.4	335.7	258.2	472.6	253.1

MW·d/kg HE, compared with 8.3 MW·d/kg HE for natural uranium fuel in Bruce A reactors. Peak element burnup is about the same as that for natural uranium. The fresh fuel contains 232 g plutonium per bundle, of which 94% is fissile (Table XLVII). The plutonium disposition rate in a Bruce A reactor is 1.0 Mg plutonium per year per reactor (assuming an 80% capacity factor). The MOX fuel fabrication capacity requirement is about 80 Mg per year per reactor.

The use of dysprosium as a neutron absorber in the inner elements results in negative coolant void reactivity. This compensates for the faster dynamic response of MOX fuel to reactivity perturbations resulting from a lower delayed neutron fraction and prompt neutron lifetime when using plutonium as opposed to natural uranium. However, the use of a neutron absorber in the centre of the bundle also increases the peak linear element rating in the outer two rings of fuel. The relative plutonium enrichment in rings 3 and 4 was chosen with a view to minimizing the peak ratings of both fresh fuel and discharge fuel, and optimizing the thermohydraulic performance of the bundle.

The reference MOX fuel design was chosen to provide the fastest start for the plutonium dispositioning mission. Two other CANDU MOX fuels have since been designed, to meet different objectives.

In the second MOX option, the plutonium disposition rate was increased by 50% by raising the amount of plutonium in the 37 element bundle, without significantly increasing the fuel burnup (Table XLVII). To compensate for the excess reactivity, the burnable poison content in the central elements was increased from 5% to 15%, and the purity of the coolant and moderator was downgraded from 99.75% to 97%. This is a good illustration of how the simplicity of the CANDU bundle design allows optimization of the composition to achieve specific objectives.

The resultant 37 element fuel bundle has a plutonium loading of 336 g confined to the two outer rings (3.1% plutonium in ring 3, 1.6% plutonium in ring 4), 15% dysprosium in the seven central elements, and depleted uranium as the base material throughout the bundle. The average discharge burnup is slightly greater than that of the earlier study, i.e. 10 MW·d/kg HE. The plutonium disposition rate in a Bruce A reactor is 1.5 Mg plutonium per year. The MOX fuel fabrication capacity requirement is 78 Mg per reactor per year. As with the earlier design, the reactor operates within the natural uranium licence envelope.

A third MOX design was considered to increase the energy output from the weapons derived plutonium, which could be important if the plutonium were viewed as a valuable resource (Table XLVI). This option uses a large amount of plutonium in the CANFLEX bundle. The plutonium is confined to the two outer rings of fuel: 4.6% plutonium in ring 3, and 2.6% plutonium in ring 4, mixed with depleted uranium. The eight central elements contain 7% dysprosium mixed with depleted uranium. This advanced MOX bundle contains 473 g plutonium in the fresh fuel. The CANFLEX bundle has 20% lower peak element ratings than does the 37 element bundle operating at the same bundle power, and has improved thermohydraulic performance. The lower ratings facilitate achievement of higher burnup using this option, which has a core average burnup of 25 MW·d/kg HE and a peak element burnup of 35 MW·d/kg HE. Nominal D_2O purity of 99.75% is used in the calculations. The plutonium disposition rate in a Bruce A reactor is 0.8 Mg plutonium per year. The higher fuel burnup reduces the MOX fuel fabrication requirement to 28 Mg per reactor per year, which will result in a significant reduction in the capital cost of the MOX fuel fabrication plant.

6.7.3. Fuel composition of conventional MOX options

Table XLVI shows the composition of the fresh and spent fuels for the three conventional MOX options. The fissile plutonium destruction efficiencies for the three options are 58%, 41% and 70%, respectively. The total amount of plutonium is reduced by 34%, 23% and 46%, respectively, for the three options.

6.7.4. Performance of reference CANDU MOX fuel

The performance of the reference MOX (option 1) fuel design was assessed in several ways. MOX power histories resulting from full-core reactor physics simulations were compared with those for natural uranium and showed that MOX fuel falls within the natural uranium envelope. One observation from this is that only relatively fresh fuel (burnup less than ~2 MW·d/kg HE) experiences power boosting during refuelling. This is good from the perspective of stress corrosion cracking, since low burnup fuel is resilient to power boosts (there are relatively few fission products and the cladding remains ductile).

Power ramp performance is assessed by comparing the power/power boost envelopes for the MOX fuel derived from a fuel management simulation to the stress corrosion cracking threshold curves. In the MOX fuel power boost envelope, the power boosts at low burnup are the result of power increases occurring during refuelling, as the bundles are shifted towards the centre of the core. The power boosts at higher burnup are the result of the ‘refuelling ripple’ effect resulting from the refuelling of neighbouring channels. The MOX power envelope exceeds the ‘ramped power’ threshold for stress corrosion cracking, while the power boost envelope for the MOX fuel is below this threshold curve. Since both the power and the power boost threshold curves have to be exceeded in order for there to be a non-trivial probability of stress corrosion cracking failure occurring, such failures are not predicted for the reference MOX fuel.

6.7.5. Nuclear design

The use of MOX fuel in a CANDU reactor introduces changes to the neutronic characteristics relative to the natural uranium core, including:

- Increase in the reactivity of the fuel lattice,
- Decrease in the reactivity depth of the control and safety systems,
- Decrease in the delayed neutron fraction,
- Decrease in the prompt neutron lifetime,
- Decrease in the negative fuel temperature feedback.

These changes result in nominally faster response to reactivity perturbations in the MOX core compared with the natural uranium core. The flexibility of the CANDU reactor design allows modifications to be made to the MOX fuel design such that the resulting MOX reactor operates within the safety and licensing parameters of the natural uranium reactor.

One significant innovation in the MOX fuel design is the strategic placement of a burnable poison in the fuel. The burnable poison suppresses the excess lattice reactivity and reduces the coolant void reactivity. Dysprosium is an excellent burnable poison for CANDU because its burnout rate matches the depletion rate of fissile materials such as ^{235}U and ^{239}Pu. In the fuel, dysprosium is mixed with depleted uranium in the two inner rings, and plutonium is mixed with depleted uranium in the two outer rings. The amounts of dysprosium and plutonium used are based on the plutonium consumption rate, MOX fuel production rate and maximum allowable burnup. The amount of dysprosium in the fuel also affects coolant void reactivity. In order to offset the faster neutronic response of the MOX fuel, the appropriate amount of dysprosium is added to the fuel to produce a slightly negative void reactivity. This adds stability to the MOX core and eliminates the need for modification of the control and safety systems used for the natural uranium core.

In CANDU reactors, fission is caused by thermal neutrons entering the fuel channel from the moderator. Fuel elements in the third ring are shielded from the thermal neutrons by the elements in the outermost ring, i.e. ring 4. This reduced neutron flux level results in a much lower power output in ring 3 compared with that in ring 4. The power output of the entire fuel bundle is often limited by the maximum allowable power rating of an individual fuel element. Therefore, it is desirable to design a fuel bundle such that all fuel elements operate at comparable power ratings. This is achieved through a graded enrichment scheme where the plutonium content of the elements in the third ring is higher than that in the fourth ring, 2% and 1.2%, respectively. The higher enrichment level in ring 3 compensates for the reduced flux, and enables the elements in ring 3 to operate at a power level comparable to that in ring 4.

The neutron flux level in the central seven elements, which contain 0.2% depleted uranium and 5% dysprosium (plutonium is absent), is so low that these elements do not produce any significant amount of power. Hence, all the power in the bundle is effectively produced by 30 elements instead of 37. This requires the MOX fuel bundle to operate at a lower maximum bundle power limit than that of natural uranium fuel. Using the two bundle fuelling scheme, the maximum bundle power in the MOX core is 780 kW compared with 950 kW for natural uranium.

The maximum fuel element burnup of the MOX fuel is calculated to be 15.5 MW·d/kg HE, which is essentially the same as the maximum fuel burnup attained by natural uranium fuel elements in Bruce A. The MOX fuelling rate is 15.5 bundles per full power day, which is lower than the present fuelling rate of ~18 bundles per full power day with natural uranium fuel.

6.7.6. Nuclear safety

The control and safety reactivity device worths in the reference MOX core are lower than those of the natural uranium core. This reduction is due to the increase in the cell average absorption in the MOX lattice over that in the natural uranium fuel lattice. However, the reactivity perturbations in the MOX core (i.e. refuelling and xenon effects), are also proportionately lower than those in the natural uranium core. Hence, control of the MOX core is not expected to be significantly different from that of the natural uranium core.

The major difference between a natural uranium fuelled reactor and a MOX fuelled one is the reactivity effect occurring during a hypothetical LOCA. Most perturbations in a CANDU reactor operating with natural uranium fuel are small and slow, and can easily be handled by the control system. The exception is a large LOCA during which the positive void reactivity of natural uranium generates an increase in power prior to operation of the shutdown systems. All current generation CANDU reactors are equipped with two independent fast acting shutdown systems, each of which is able to terminate quickly any increase in power occurring during the most severe LOCA. In the case of the MOX fuel lattice, when coolant is voided, the neutron scattering effect due to the coolant in the fuel channel is curtailed. This allows more neutrons to reach the centre of the fuel bundle, i.e. the poisoned region. The negative reactivity generated by the higher neutron absorption in the poisoned inner region more than compensates for the increase in reactivity in the MOX region. The overall coolant void reactivity effect is therefore negative. This negative void reactivity enhances the performance of the control and safety systems.

A key parameter when evaluating reactor safety is the energy deposited in the fuel during the first few seconds following a LOCA. Analysis of LOCA response compared the power transient and energy deposition in the natural uranium core with that in the MOX core for the first few seconds following initiation of a large LOCA. The analysis showed that energy deposited in the fuel is less for the MOX core, indicating that fuel heat-up is reduced and that existing safety design objectives are met.

To assess the performance of safety systems and quantify accident dose consequences, the decay heat level and radionuclide content of the MOX fuel were calculated. On the basis of these considerations, fractional releases caused by postulated accidents which could discharge small quantities of radionuclides from the fuel into the containment will be similar to those from natural uranium fuel.

Analyses of various accident scenarios indicate that:

- Shutdown system reactivity depth is maintained for the case of an in-core small break LOCA which dilutes the poison concentration in the moderator.
- Loss of reactivity control events are not significantly different in a MOX core than in a natural uranium core, although there is a change in the neutron kinetics. Reactor safety will not be affected negatively by changes in control characteristics.
- The decision and action time associated with establishing alternative heat sinks for secondary side failures is increased slightly in a MOX core as a consequence of the lower (short term) decay heat of MOX fuel, relative to that of natural uranium fuel.

6.7.7. Thermohydraulic design

The objectives of the thermohydraulic assessment were to determine the critical heat flux for the reference MOX (option 1) fuel design, and assess its impact on critical channel power. The regional overpower protection system in CANDU reactors protects the fuel from dry out resulting from overpowers that could arise from perturbations in the power shape, or from a slow loss of reactivity control power transient. The regional overpower protection system prevents fuel sheath dry out by limiting channel powers to below the critical channel power. The critical channel power values are used to establish the neutron flux detector trip settings for the reactor's regional overpower protection system.

In CANDU analyses, the critical channel power is calculated by assuming nominal (i.e. 100% full power) constant header to header pressure drop, inlet header temperature and outlet header pressure. The steady state thermohydraulics code NUCIRC calculates individual channel flows, cross-sectional averaged physical properties of the D_2O coolant at various locations in the PHTS, and dry out power for any number of channels. The different radial and axial power profiles of the MOX fuel relative to those of natural uranium were accounted for by the subchannel code ASSERT, which accounts for cross-mixing between adjacent subchannels, cross-flows generated by pressure differences between subchannels, and the local heat flux in the fuel.

Within the accuracy of the calculations, the calculated critical channel power is the same for both the MOX (option 1) core and the natural uranium core, and therefore no change to the regional overpower protection system trip set points would be required for the MOX fuel.

6.7.8. Modifications to plant installations

Engineering changes to existing CANDU plants that may be required to receive, store and handle the MOX fuel were assessed, and the most significant item identified was the need for enhanced security for the MOX fuel. Existing facilities and procedures used for natural uranium fuel are inadequate for MOX, mostly because of the low level of security required for natural uranium fuel. The new fuel storage building will be an IAEA Category 1 facility, with reinforced concrete floor, walls and roof, controlled access entrances, alarms and other security features.

As a result of the higher (compared with fresh natural uranium) radiation fields originating from fresh MOX fuel, MOX fuel bundles will be placed in shielding sleeves (water extended polyester surrounded by stainless steel) at the fabrication plant, and left in these until they are ready for being loaded into the reactor fuelling machine. This allows the handling of MOX fuel with equivalent or less personnel radiation doses relative to natural uranium.

The fuelling rate for the reference MOX fuel in Bruce A is 15.5 bundles per unit per day, lower than the rate for natural uranium at 18 bundles per Bruce A unit per day. The fuel handling system is therefore essentially the same.

6.8. PLUTONIUM ANNIHILATION

6.8.1. Introduction

This is a longer term technology for dispositioning ex-weapons plutonium. It could also be considered as a means of eliminating, or reducing, the stockpile of separated civilian plutonium, if that constitutes a particular national or global objective. This is a variant of the actinide burning application of the CANDU reactor that was considered in Section 6.5.5, in which the ex-weapons plutonium is embedded in an inert matrix carrier. Hence, three possible uses of inert matrix fuels in CANDU are: actinide burning, disposition of ex-weapons plutonium and disposition of civilian plutonium. The fuel design and reactor physics characteristics are similar for all of these applications. The reference inert matrix host selected by AECL is SiC, which has already been discussed in Section 6.5.5. An existing CANDU 6 reactor could destroy over 75% of the total plutonium, and more than 93% of the fissile plutonium, in a single pass. No other reactor technology has this flexibility.

6.8.2. Fuel design, fabrication and performance

The weapons derived Pu–SiC CANFLEX fuel bundle considered in this study contains 210 g of plutonium and 60 g of gadolinium [251]. The central element contains 20 g of gadolinium, the remaining 40 g of gadolinium being distributed uniformly throughout the seven elements comprising the next ring. Each of the two outer fuel rings contains 105 g of plutonium mixed with SiC. The plutonium content per element in the outermost fuel ring is two thirds that of the third fuel ring. This differential enrichment scheme improves the element power distribution and reactivity effect over the lifetime of the fuel bundle. Table XLVIII gives the initial material composition of the weapons derived Pu–SiC fuel bundle.

TABLE XLVIII. INITIAL COMPOSITIONS OF WEAPONS PLUTONIUM INERT MATRIX FUEL IN A CANFLEX BUNDLE

Isotope/element	Composition (g)			
	Ring 1 (1pins)	Ring 2 (7 pins)	Ring 3 (14 pins)	Ring 4 (21 pins)
Pu-239			98.72	98.72
Pu-240			6.11	6.11
Pu-241			0.14	0.14
Pu-242			0.02	0.02
Total Pu			104.99	104.99
Si (natural)	139.64	977.42	1399.01	2098.54
C (natural)	59.84	418.90	599.58	899.38
Gd (natural)	20.00	40.00		

6.8.3. Lattice reactivity of Pu–SiC fuel

Figure 195 shows the variation of the lattice *k*-infinity as a function of the total energy produced in the Pu–SiC bundle for weapons derived plutonium, with and without gadolinium. Natural UO_2 fuel is also shown in this figure for the purposes of comparison.

It is clear that the *k*-infinity of the Pu–SiC fuel lattice begins at a much higher value than that of fresh natural UO_2 fuel, and that it decreases very rapidly as the initial fissile material is depleted. This excessive rate of reactivity depletion, especially for the weapons derived plutonium fuel, will produce unacceptably high levels of reactivity and power perturbations in the channels being refuelled. As a consequence, it is necessary to use a relatively large amount of gadolinium in the weapons derived Pu–SiC bundles in order to minimize the refuelling perturbations and to achieve acceptable global and local power distributions. Incurring a fuel burnup penalty is not a concern in the weapons derived Pu–SiC option because the objective is to maximize the rate and degree of plutonium destruction.

Coolant void reactivity is –6 mk for the weapons Pu–SiC option, and the power coefficient is also negative. There would be no power pulse in a postulated LOCA,

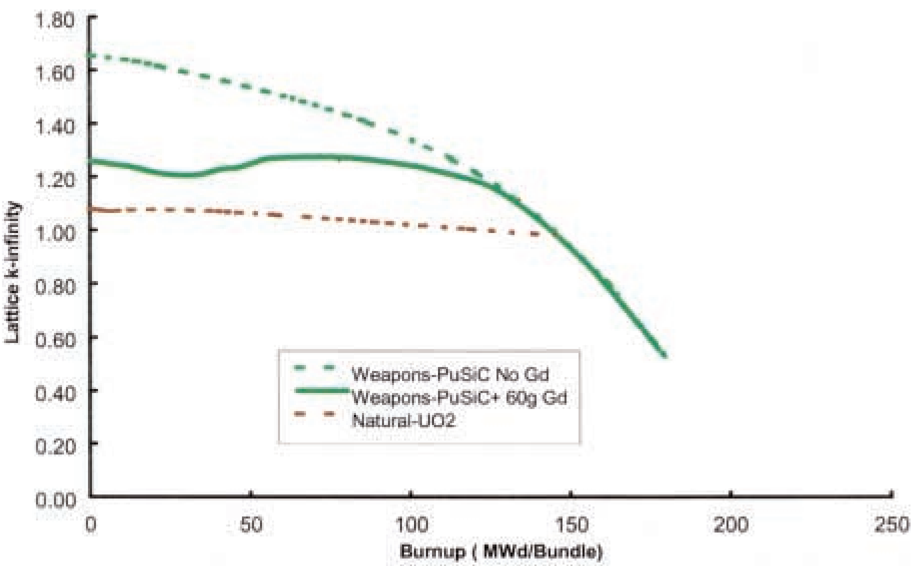


FIG. 195. Lattice k-infinity versus fuel burnup.

and the safety and licensing analyses would be greatly simplified. The fuel temperature coefficient is very slightly positive, about 12 μk/°C. However, the sign of the fuel temperature coefficient is not relevant with this fuel. An increase in fuel power would result in a very small increase in fuel temperature because any increase in heat in the fuel would immediately be transferred to the coolant as a result of the high thermal conductivity of the SiC. The increase in coolant temperature would reduce the coolant density and produce a negative reactivity feedback as a result of the negative coolant void reactivity effect.

6.8.4. Full-core fuel management simulations

Detailed, realistic fue management simulations were performed using RFSP for a standard CANDU 6 reactor using the weapons derived Pu–SiC fuel in the whole core. Acceptable bundle and channel powers were obtained using a bidirectional, two bundle shift refuelling scheme. The refuelling rate is 15 bundles per full power day, about the same as that for natural uranium fuel.

Figure 196 shows that the time average channel power distributions for the Pu–SiC core are very similar to those in the present natural UO_2 fuelled CANDU

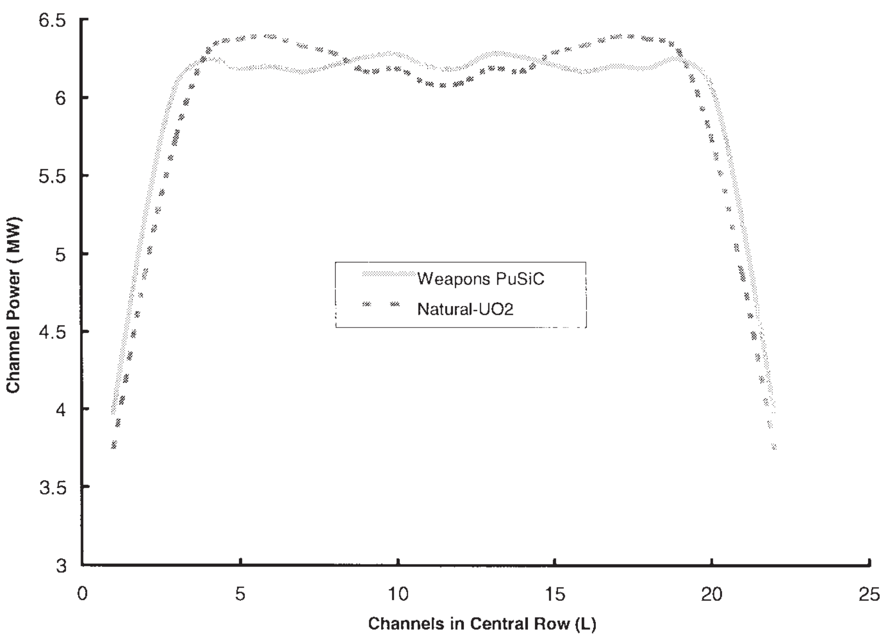


Fig. 196. Radial power distribution.

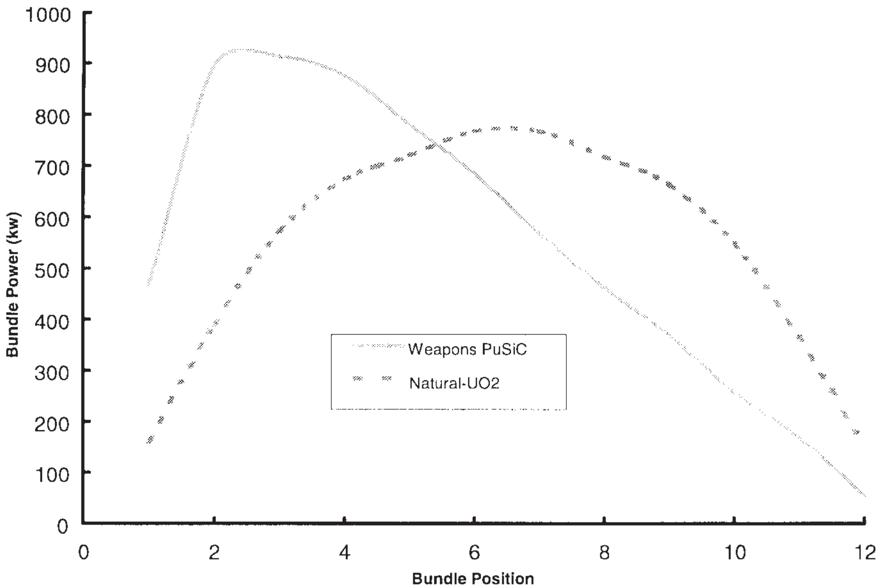


FIG. 197. Axial power distribution in a central channel.

reactor. Figure 197 shows the time average bundle power distributions in a central channel, normalized to a channel power of 6.5 MW. Bundle powers are high at the inlet end of the channel for the Pu–SiC fuel because of the rapid depletion of fissile plutonium. This axial power distribution, which peaks towards the inlet (refuelling) end of the channel, is also expected to result in improved thermohydraulic performance (higher critical channel powers). The weapons derived Pu–SiC option gives a higher peak time average bundle power than does the natural UO_2, 910 kW as opposed to about 800 kW. The maximum instantaneous channel power is less than 7.1 MW, and the maximum instantaneous bundle power is less than 990 kW. The fuel temperatures would be very low, even at these relatively high bundle powers, because the thermal conductivity of the SiC matrix is many times greater than that of UO_2.

6.8.5. Destruction efficiency of plutonium and minor actinides

The plutonium and minor actinide contents of fresh and discharged Pu–SiC bundles are shown in Table XLIX for weapons derived plutonium fuel. The discharge fuel burnup is 145 MW·d per bundle for the weapons derived Pu–SiC fuel. In

TABLE XLIX. PLUTONIUM AND MINOR ACTINIDE CONTENTS IN WEAPONS DERIVED Pu–SiC FUEL BUNDLES

Isotope	Weapons derived Pu–SiC Fuel		
	Fresh (g/bundle)	Discharge (145 MW·d) (g/bundle)	Destruction (%)
Pu-238			
Pu-239	197.44	4.78	97.6
Pu-240	12.22	31.13	
Pu-241	0.29	8.53	
Pu-242	0.05	8.38	
Total Pu	210.00	52.82	74.9
Np-237		0.074	
Am-241		0.649	
Am-243		0.068	
Cm-242		0.089	
Cm-244			
Total minor actinides		0.880	

comparison, the fuel burnup of a natural uranium CANDU fuel bundle, which contains 133 g of ^{235}U, is about 136 MW·d per bundle.

The disposition rate of the weapons derived Pu–SiC option is about 1.8 Mg plutonium per GW(e)·a. About 75% of the total initial plutonium inventory, 97% of the initial ^{239}Pu and more than 93% of the total fissile plutonium are destroyed. Two CANDU 6 reactors could, therefore, almost completely destroy 50 Mg of weapons derived plutonium in less than 25 years.

6.8.6. Conclusions

Good neutron economy and the use of on-power refuelling enable existing CANDU reactors achieve significant destruction of excess weapons derived plutonium using an inert matrix fuel. Detailed fuel management simulations conclude that existing CANDU reactors can operate with a full core of plutonium inert matrix fuel.

6.9. THORIUM

6.9.1. Introduction: Benefits of thorium

The thorium fuel cycle in HWRs is of strategic interest for several reasons [254]:

- The abundance of thorium in the earth's crust is about three times that of uranium. Hence, the thorium fuel cycle ensures long term nuclear fuel supply. In the case of countries with abundant thorium reserves, the thorium fuel cycle in HWRs would enhance both the sustainability of nuclear power and the degree of energy independence, using a single reactor type.
- In thorium fuel, ^{233}U is produced in-reactor through neutron capture in ^{232}Th, and subsequent beta decay of ^{233}Th and ^{233}Pa. The concentration of ^{233}U in the spent fuel is about five times higher than that of ^{239}Pu in spent natural uranium (UO_2) fuel. This isotope of uranium is a very valuable fissile material because of the high number of neutrons produced per neutron absorbed in a thermal neutron spectrum.
- The thermal conductivity of ThO_2 is about 50% higher than that of UO_2 over a large temperature range, and its melting temperature is 340°C higher than that of UO_2 [255, 256]. As a consequence, fuel operating temperatures will be lower for ThO_2 than for UO_2, and all thermally activated processes, such as creep and diffusion of fission gas from the fuel, will be lower. Fission gas release from the fuel should be lower for ThO_2 than for UO_2 operating under similar ratings.
- Thoria is chemically very stable and does not oxidize — a benefit for normal operation, postulated accidents and waste management.
- The isotope ^{232}Th produces fewer minor actinides than does ^{238}U. The resultant lower radiotoxicity of spent thorium fuel may be viewed as a benefit in waste management. However, in the Canadian concept of engineered geological disposal, the actinides contained in used fuel are not significant contributors to radiological risk, and therefore this benefit is judged to be small in that context [257].
- The amount of energy that can be extracted from mined uranium can be significantly extended by using thorium fuel cycles, and a wide range of thorium cycles is feasible in HWRs. In the limit, the self-sufficient equilibrium thorium cycle is independent of natural uranium and of any external supply of fissile material [258, 259]. The high neutron economy of HWRs makes this fuel cycle theoretically achievable. Hence, a single reactor technology can provide both short term and long term assurance of fuel supply. High conversion ratio HWRs utilizing thorium would also be synergistic with more expensive FBRs supplying fissile material.

- The OTT cycle in HWRs provides an evolutionary approach to exploiting some of the energy potential of thorium without recycling. The optimal OTT cycle is economic, both in terms of money and in terms of uranium resources. This cycle creates a mine of valuable ^{233}U, safeguarded in the spent fuel, for future recovery if desired.
- Since commercial thorium fuel recycling facilities have not yet been built, there is an opportunity to establish a new, proliferation resistant technology for recycling.

6.9.2. Fuel design, fabrication and irradiation performance

6.9.2.1. Canadian experience

AECL's past experience gained in fabricating ThO_2, $(Th,U)O_2$ and $(Th,Pu)O_2$ fuels includes the following fabrication routes: co-precipitation, dry powder blending/milling, wet powder blending/milling, sol–gel microspheres and extrusion. The latter two processes are very experimental and are not discussed here.

There are several methods of co-precipitating UO_2 and ThO_2 [260, 261]. AECL's study of co-precipitation has focused on the addition of ammonia to nitrate solutions. In this process, uranium and thorium are dissolved in a nitrate solution to form $UO_2(NO_3)_2$ and $Th(NO_3)_4$. Ammonia is added to the solution to precipitate $(NH_4)U_2O_7$ and $Th(OH)_4$. The precipitate is calcined to form blended UO_2 and ThO_2 powder, which is subsequently processed into fuel pellets. The microstructure and the quality of the pellets made by this process are generally very good.

Dry blending can be done by several methods, including mixing in a V-blender, dry ball milling, vibratory milling and jet milling. These tend to be dusty processes. There can be problems in maintaining uniformity on a microscopic scale, and intensive mixing is required. AECL investigated dry blending, using previously ground thoria powders that were mixed with highly enriched uranium powders and dry ball milled before pressing and sintering. The microstructure of fuel pellets made by this route was quite uniform and devoid of cracks and large pores.

As is the case with dry blending, there are several wet mixing methods, including wet ball milling and attrition milling. Wet processes have the advantage of not being dusty. In AECL's experience, it was found that product quality was poor and not reproducible with wet processes. The causes of the variations from batch to batch were not well understood. The fuel pellets used as reference fuels in the Whiteshell reactor 1 irradiations were manufactured using a process of attrition milling, pan drying (forming cakes), granulation, pressing and sintering. This process yielded pellets with non-uniform microstructures and evidence of residual granules from the fabrication process.

From the mid-1970s to the mid-1980s, AECL conducted an extensive programme to develop thoria fuels for use in CANDU reactors. Test irradiations were performed in the Whiteshell reactor 1 at AECL's Whiteshell Laboratories. Fission gas release from standard reference $(Th,U)O_2$ pellets in these irradiations was significantly higher than from other thoria fuel irradiations and was generally comparable to UO_2 under similar conditions.

Thoria is expected to have superior fission gas retention to that of UO_2, especially at higher powers (see Section 6.9.5.1). AECL investigated the fabrication of ThO_2 (with and without admixed enriched UO_2), and discovered that the primary cause of the non-uniform microstructure was the lubricant used for blending with the granulated fuel before final pressing. By removing the lubricant and using a wash to lubricate the pellet die, pellets with uniform microstructure free of residual granules and with high sintered density could be fabricated. Although the lubricant technique of lubricating dies has been very successful in UO_2 fuel fabrication, it does not appear to work well with ThO_2 based fuels.

As a result of AECL's investigation into factors controlling microstructure, a programme is under way to fabricate and irradiate thoria fuels having controlled microstructures in order to demonstrate the effect of microstructure on fuel performance and the superior performance of thoria fuels over those of UO_2.

In addition to the Whiteshell reactor 1 irradiations, AECL also irradiated four 19 element bundles in the NPD reactor between 1977 and 1987. Two of these bundles contained 2.6 wt% UO_2 (enriched to 93 wt% ^{235}U) in ThO_2, and the other two contained 1.45 wt% UO_2 in ThO_2. The irradiation was undertaken at low powers (less than 30 kW/m) to discharge burnups of approximately 40 MW·d/kg HE. The fission gas release from these fuels was typically less than 1%, demonstrating good performance at low powers.

6.9.2.2. Indian experience

(a) Thorium fuel fabrication experience

Thorium reactors form an important part of India's future nuclear programme and therefore India has devoted considerable effort this area and steady progress is being made [262, 263]. Indian experience with thorium dates back to the time when a 40 MW research reactor (CIRUS) was commissioned and thorium rods ('J-rods') were introduced into the reflector located outside the reactor vessel. Fabrication experience gained in making thoria and thoria based J-rods was further broadened when about 2.5 t of high density sintered thoria pellets was fabricated. Since then, this technology has been transferred to the commercial plant, i.e. the Nuclear Fuel Complex. Considerable quantities of thoria elements were fabricated there and supplied for use as a blanket in the LOTUS facility in Switzerland. These elements have now been transferred to France where they will be used for carrying out reactor physics experiments.

A moderate number of ThO_2 fuel bundles were fabricated and loaded into the two units at the Kakrapar Atomic Power Station, and more recently into the second unit at the Rajasthan Atomic Power Station (2.5 Mg). The characteristics of thorium fuel pins have been taken to be essentially the same as those of UO_2 fuel.

A major difficulty in the use of ^{233}U fuels is the high radiation and dose problem. The isotope ^{233}U is always associated with traces of ^{232}U, which gives rise to highly penetrating gamma emitting daughter products. Unlike natural uranium and thorium, fuel containing ^{233}U must be fabricated remotely, with proper containment inside either shielded gloveboxes or hot cells in order to keep the operator exposure at low levels.

The R&D work on thorium fuel fabrication in India is two pronged. One route is the powder metallurgy process, and the other is the sol–gel microsphere pelletization process. In the development of the sol–gel process, India started from what was developed at Kernforschungsanlage Jülich GmbH (KFA Jülich) in Germany for the pebble bed high temperature gas cooled reactor. The process that was used at KFA Jülich was modified in many ways. One major change was the addition of carbon black to the sol in order that the microspheres produced were more porous. The highest density that could be achieved without adding the carbon was only about 83% theoretical density. Another modification was the addition of about 1% calcium nitrate in the feed solution in order to improve the sinterability of the calcined microspheres.

The sol–gel microsphere pelletization process is ideally suited for remote and automated fabrication of thorium based highly radiotoxic oxide fuels. In this instance, sol–gel derived calcined oxide microspheres, rather than fine powder derived granules, are used for pellet pressing and sintering. Thus, radiotoxic dusts and aerosols are avoided and process losses minimized. This process is amenable to remote handling, because of the dust free, free flowing nature of microspheres. A high degree of microhomogeneity is attained in the fuel pellets because of the mixing of heavy metals in their nitrate solution forms. Finally, fuel pellets that combine high density with a desirable microstructure are obtained by this route. The sol–gel process is being increasingly used for making fine and homogeneous ceramic powders for a variety of applications, and can produce high quality microspheres. A recent innovation of the process involves the preparation of soft thorium oxide or thorium–uranium microspheres and their use in making fuel pellets. Superior microstructure and porosity distribution of the pellets obtained through this route have the potential to translate into superior performance in a reactor.

Thorium fuel has also been produced via two alternative routes: the cold pelletization and sintering route, and the pellet impregnation method used for (thorium–uranium) mixed oxide. This latter method uses a novel technique in which

ThO_2 pellets of low density are suspended in a high temperature uranyl nitrate bath in order to soak up ^{233}U. The conventional powder pellet route involves simultaneous mixing and grinding of ThO_2 powder with UO_2 or PuO_2 powders, granulation, cold pelletization of granules at ~350 MPa and high temperature (~1700°C) sintering of pellets in an atmosphere comprising argon plus hydrogen (8%). The ThO_2 and PuO_2 powders are produced by air calcination of the oxalates. These powders are extremely fine (<1 μm) and have poor flow characteristics. Likewise, the UO_2 powder produced via the ammonium diuranate route is also fine and is also not free flowing. Further, the platelet morphology of oxalate derived ThO_2 powder causes problems in achieving homogeneity while mixing with UO_2 or PuO_2 powders.

Some modifications were made to the conventional 'powder pellet' route in order to fabricate ThO_2–UO_2 and ThO_2–PuO_2 pellets of high density and good homogeneity. A thorium nitrate feed solution was doped with ~1 wt% magnesium sulphate or nitrate before precipitation of thorium oxalate in order to obtain ~0.4 wt% MgO which acts as a sintering aid in the calcined ThO_2 powder. Proper homogenization of oxide powder mixtures is ensured by premilling the oxalate derived ThO_2 powder in order to break the platelet morphology before co-milling with UO_2 or PuO_2. Approximately 0.25 wt% Nb_2O_5 is mixed with the ThO_2–UO_2 and ThO_2–PuO_2 powders to improve their sintering capabilities.

With MgO doped ThO_2 powder, it was possible to achieve high pellet density with ThO_2–UO_2 and ThO_2–4%PuO_2 pellets at a relatively low sintering temperature (<1500°C). The Th^{4+} ion is partially replaced by Mg^{2+} ions and this causes the formation of oxygen or anion vacancies which are thought to enhance the volume diffusion of thorium ions, leading to the rapid densification of ThO_2–UO_2 or ThO_2–PuO_2 at relatively low sintering temperatures.

Cermet fuel represents one of the advanced concepts that has been considered in the case of thorium. It envisages kernels of uranium oxide that are coated with nickel and chromium and dispersed in a matrix of thorium metal. A certain level of success has been achieved in coating uranium oxide microspheres with nickel.

Other R&D work that has been undertaken at Trombay on the use of thoria in power reactors is related to evaluating the thermal diffusivity and hot hardness of ThO_2–4%PuO_2 and ThO_2–2%UO_2. These data will be necessary to estimate the maximum heat rating that these fuel elements can withstand.

(b) Uranium-233 recycle experience

The Thorex process uses an organic solvent — tributylphosphate. As already mentioned, thorium rods, commonly referred to as J-rods, have been loaded into the CIRUS reactor since the early 1960s. A facility was set up at Trombay during the late 1960s to separate ^{233}U from the irradiated J-rods. Chemical reprocessing yields essentially pure ^{233}U with traces of ^{232}U. Although the 'ppm level' concentration of

^{232}U in the fresh ^{233}U recovered by chemical processing is not by itself of significance, a serious problem arises as a result of the decay products of ^{232}U. As a result of these decay products, the gamma activity of separated ^{233}U increases for about ten years. Owing to the very high activity level resulting from the buildup of ^{228}Th, it is very undesirable to reprocess irradiated thorium, recover the ^{233}U, and then store the recovered ^{233}U. It is essential to minimize the time interval between the separation of ^{233}U and the refabrication of the ^{233}U into fuel elements. Buildup of ^{228}Th necessitates not only shielding against the high energy gamma rays, but also provision of neutron shielding.

Alpha particles emitted from the decay of ^{233}U, ^{232}U and ^{228}Th interact with light elements such as carbon and oxygen, releasing neutrons in the process. Thus, it is important to refabricate the ^{233}U as early as feasible after its recovery. In summary, the discharged fuel should be stored and reprocessing should await the fabrication of ^{233}U into fuel elements. Another aspect of thorium reprocessing is related to the refabrication of the thorium separated from irradiated fuel. The separated thorium contains enough ^{228}Th to make the refabrication of the thorium as complex as the fabrication of ^{233}U bearing fuel. It has been observed that if the thorium is stored for about 10–16 years the activity would decrease to acceptable levels.

Laboratory studies on the reprocessing of thorium fuels started early in India. The method chosen was the solvent extraction process. Later on, in the 1960s, a pilot plant was set up in Trombay to reprocess the irradiated J-rods discharged from the CIRUS reactor. In the 1970s, the reprocessing programme at the Indira Gandhi Centre for Atomic Research set up a utility for reprocessing thorium fuel on a larger scale. Considerable work has also been done on the recovery and final purification of ^{233}U from irradiated thorium. The reprocessing was undertaken using tributylphosphate as extractant. In the initial stages, the emphasis was on the recovery of ^{233}U, while the thorium and fission products were allowed to go into a common solution.

While the emphasis has so far been on conventional reprocessing of spent thoria to recycle separated ^{233}U, great potential exists in more proliferation resistant recycle technologies that, for instance, only remove the neutron absorbing fission products (see Section 6.9.6). Extra handling and fuel fabrication costs due to the resultant high level of radioactivity of the fuel may be offset through a simpler recycle technology.

(c) Thorium irradiation experience in research reactors

The performance targets for thoria fuel will depend on the specific fuel cycle intended. The requirements for a low burnup, near breeder type of cycle will be different from those to be met by high burnup thorium fuel.

India has, for many years now, been following a policy of irradiating thoria whenever possible. The first scheme of this kind, as discussed previously, involved fabricating thorium rods and placing them in the reflector of the CIRUS research reactor. The 40 MW research reactor has 64 positions in which to load thorium fuel rods. A number of thorium metal and a larger number of thorium oxide rods have been fabricated and irradiated in the CIRUS reactor. Since the only neutron flux that they encountered was the leakage flux from the reactor, the ^{233}U buildup was quite slow. On the other hand, they did not form a load on the reactor, and therefore the ^{233}U was produced at no loss. These rods were subsequently reprocessed to recover the ^{233}U contained in them.

The CIRUS reactor has an in-core loop for fuel irradiation. This loop, termed the pressurized water loop (PWL), has been extensively used for testing thorium fuel. A fuel assembly containing ThO_2 and PuO_2 was irradiated in the PWL for almost three years from May 1985 to January 1988. The highest heat rating that the fuel pins experienced was 38.5 kW/m. The power was raised without restriction until the heat rating reached 32.5 kW/m. Subsequently, it was raised in steps of 0.4 kW/m with an interval of two hours elapsing between successive power increases. In addition, the fuel cluster was subjected to about 100 power cycles. When the cluster was removed, the thorium–plutonium mixed oxide fuel pins had undergone a burnup of about 18.5 MW·d/kg HE.

Further to this, a string of two, six pin clusters was loaded into the CIRUS during in the last week of 1992. One of these clusters comprises six pins of unenriched thoria; the other is a composite cluster comprising two ThO_2–PuO_2 pins, two UO_2–PuO_2 pins and two natural uranium pins. The plutonium content is about 4% and the heat rating of the ThO_2–PuO_2 pins is 50 kW/m. The objective of this irradiation is to prove the design of UO_2–PuO_2 and ThO_2–PuO_2 pins of HWR design at high burnups, to improve understanding of the fuel behaviour of advanced fuel cycles and to benchmark the fuel design codes. The next assembly to be installed in the PWL in CIRUS will be similar to this one, the difference being that the pellets will be made by different fabrication routes (low temperature sintering, sol–gel, etc.).

A few ThO_2 pellets have been used in the 100 MW research reactor at Dhruva.

(d) Thorium irradiation experience in power reactors

The Indian HWR is so designed that during the major part of its life some power flattening occurs in the central part of the core, which is achieved through a differential burnup distribution. In a newly started reactor, where the entire core is at zero burnup, the reactor will have to be derated unless some other measures can be taken to obtain a similar effect. Advantage of this fact was taken in the use of thoria bundles for power flattening in the initial core of new reactors.

The first regular use of thoria fuel in a power reactor in India was at Kakrapar Atomic Power Station. About 500 kg (35 bundles) of thorium fuel was used for power flattening in the initial core of unit 1, which was commissioned in 1993. The core has already seen 200 full power days of operation, and all thoria bundles have functioned well. The same scheme of power flattening has also been employed in the second unit. Thus, 500 kg of thoria is also present in Kakrapar 2, which attained criticality in December 1994. All earlier HWRs elsewhere in the world used depleted uranium in the initial core.

Before the extensive use of thoria at Kakrapar started, performance testing of thorium bundles in power reactors was undertaken by loading four thorium bundles in Madras Atomic Power Station in May 1985. These bundles were unloaded during the usual refuelling operation. By then, the thoria bundles had been in the core for 280 effective full power days. The highest burnup was 1.7 MW·d/kg HE.

Rajasthan 2 has been restarted after retubing. In this reactor, about 250 kg of thoria was loaded for initial power flattening. The thoria bundle distribution is different at Rajasthan 2 from that at Kakrapar 1 and 2. This is because Kakrapar uses shut-off rods for reactor scram, and therefore the thoria bundle locations have to be chosen in such a manner as not to cause any reduction in the worth of the primary or secondary shutdown systems. The Rajasthan reactor depends on moderator dump for shutdown and therefore the placement of thoria bundles has only to consider power flattening.

Thoria bundles have undergone a maximum of 700 full power days of operation, and there have been no fuel failures. A plot showing the discharge burnup versus the bundle power at discharge is shown in Fig. 198.

(e) Uranium-233 fuelled test reactors

A ^{233}U fuelled experimental reactor (PURNIMA 2) was commissioned at Trombay in 1984. This is a homogeneous, zero power, solution reactor experiment using ^{233}U as the fuel, light water as the moderator and beryllium oxide as the reflector. The objectives of this experiment were to validate ^{233}U cross-sectional data, gain experience in the handling of ^{233}U fuel and to provide an initial reactor physics understanding for the KAMINI neutron source reactor, which is described below.

The KAMINI research reactor was commissioned in 1996. The reactor is fuelled by ^{233}U. The fuel is in the form of ^{233}U–Al alloy having a uranium content of 20 wt%. This alloy is fabricated into flat plates. The total ^{233}U inventory of the reactor is 600 g. The coolant moderator is demineralized light water. There is a reflector of beryllium oxide 20 cm thick, beyond which there is a water reflector. This reactor is primarily used for the neutron radiography of active components. One of its chief uses is to examine the irradiated fuel from the fast breeder test reactor.

6.9.3. Thorium fuel cycle options

Thorium itself does not contain a fissile isotope, therefore neutrons must be initially provided by addition of a fissile material, either within or outside the ThO_2

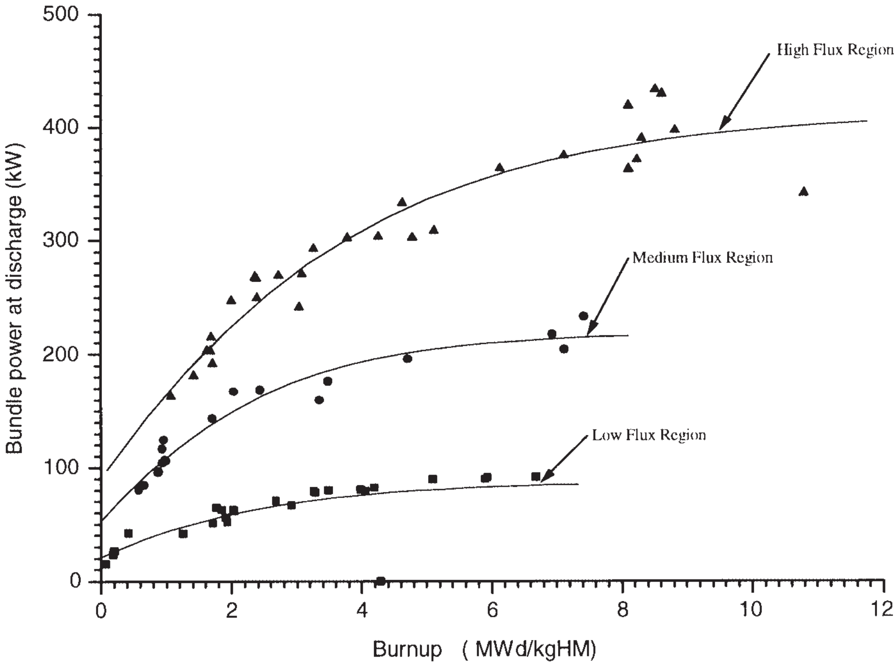


FIG. 198. Irradiation data for Indian ThO_2 bundles irradiated in power reactors.

itself. The manner in which the neutrons are initially provided defines a variety of thorium fuel cycle options in HWRs that will be examined in this section. These include the following:

- The OTT cycles, where the rationale for the use of thorium does *not* rely on recycling the ^{233}U (but where recycling remains a future option);
- Direct self-recycle of irradiated thoria elements following the OTT cycle (no reprocessing);
- Other recycling options, ranging from reprocessing to the selective removal of neutron absorbing fission products;
- The self-sufficient equilibrium thorium cycle, a subset of the recycling options in which there is as much ^{233}U in the spent fuel as is required in the fresh fuel;
- High burnup open cycles;
- The ThO_2–PuO_2 cycle, which can also be considered as an option for plutonium dispositioning.

6.9.3.1. OTT fuel cycles

The OTT cycle produces a mine of valuable ^{233}U in the spent fuel, at little or no extra cost, available for future recovery, predicted by economic or resource considerations.

High neutron economy, on-power fuelling, channel design and simplicity of the fuel bundle provide a great deal of flexibility in approaches to the OTT cycle. In the original OTT concept, it was termed the 'mixed channel' approach, whereby channels would be fuelled either with ThO_2 bundles or with 'driver' fuel, typically SEU [264]. The driver fuel would provide the neutrons required to convert ^{232}Th to ^{233}U in the thoria fuel. In such a system, the thoria would remain in the core much longer than the driver fuel would.

At low burnups, the thorium represents a load on the uranium, and therefore the presence of thorium causes a decrease in the energy obtained from uranium. With increasing thorium burnup, the ^{233}U which builds in produces power, and the sum total of the energy extracted from the SEU and the thorium can become larger than that achievable with SEU alone. At still higher burnups, the accumulated fission product poisons cause the energy extracted to decrease once again. This variation is shown clearly in Fig. 199. The total energy extracted will be the sum of the energy

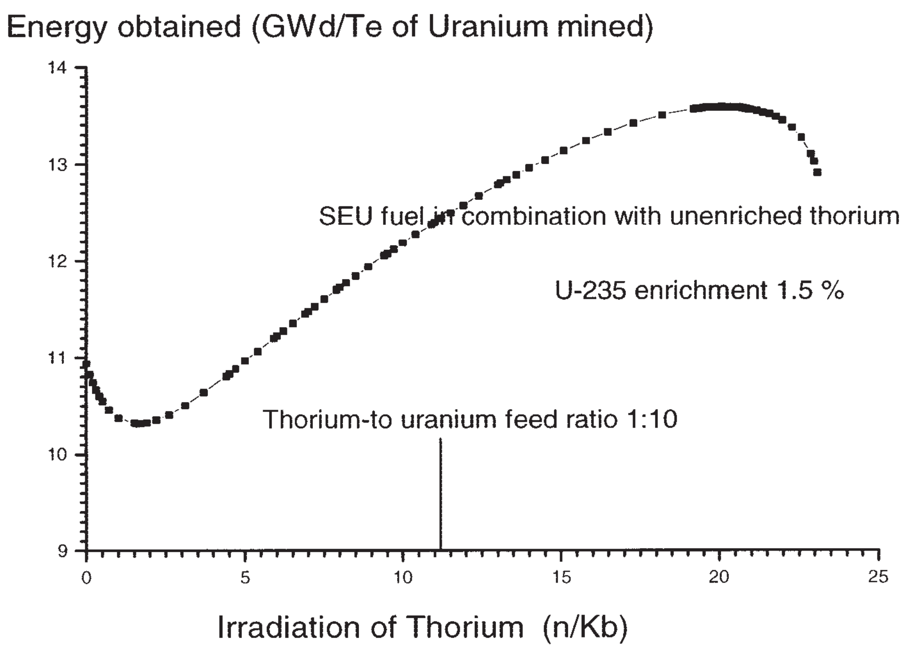


FIG. 199. OTT cycle in an HWR.

obtained from the thorium and the SEU. As the residence time of the thorium in the core increases, the energy obtained from a unit of mined uranium will first decrease, then, after passing through a minimum, will start to increase, finally becoming higher than it would have been had no thorium been present at all.

In the optimal mixed channel approach to the OTT cycle, a combination of feed rates, burnups, uranium enrichment and neutron flux level would be chosen in order that the cycle be economic (in terms of either resource utilization or money) compared with either natural uranium or SEU, without taking any credit for the ^{233}U produced. Simple ‘scoping’ studies (using a lattice code) have shown that such OTT cycles do indeed exist, although their implementation would pose technical challenges to fuel management because of the disparity in reactivity and power output between driver channels and thorium channels [264]. Other driver fuels could also be considered, such as DUPIC fuel from recycled PWR fuel or MOX fuel [265].

An alternative approach has been developed in which the whole core would be fuelled with mixed fuel bundles, which contain both thorium and SEU fuel elements in the same bundle. Figure 200 shows a CANFLEX ‘mixed bundle’ containing ThO_2 in the central eight elements and SEU in the two outer rings of elements. This mixed bundle approach is a more practical means of utilizing thorium in existing HWRs, while keeping the fuel and the reactor operating within the current safety and operating envelopes established for the natural uranium fuel cycle. Compared with natural uranium fuel, this option has better uranium utilization, comparable fuel cycle costs, lower void reactivity, higher thermohydraulic margins, a simpler fuel management scheme, and lower bundle and channel powers. However, the uranium utilization and fuel cycle costs are not as low as for SEU, or for an ‘optimized’ OTT cycle using the mixed channel approach. This mixed bundle option is a practical means of utilizing thorium in operating HWRs, within the current safety and operating envelopes, and does not involve making any significant hardware changes.

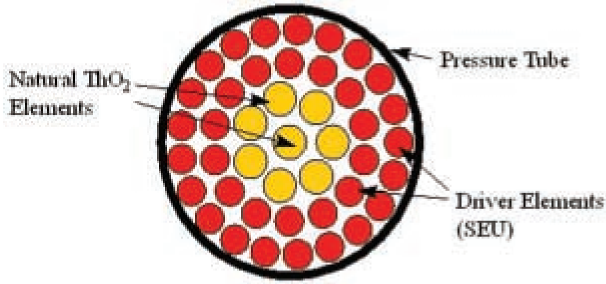


FIG. 200. The CANFLEX mixed bundle.

(a) Mixed bundle options

AECL has examined two mixed bundle strategies for burning thoria in an existing CANDU 6 reactor. In option 1, only one fuel type was used throughout the entire core, and the adjuster rods were removed. The reference fuel design for this study was a CANFLEX fuel bundle with 1.8% SEU in the outer 35 elements and natural ThO_2 fuel in the inner 8 elements. The initial fissile content was chosen to give UO_2 burnups that would be readily achievable without requiring significant development.

The second option illustrates the flexibility of existing CANDU reactors in accommodating both thorium fuel and adjuster rods. In option 2, each of the three regions shown in Fig. 201 contains a different type of thoria fuel bundle. The fuel in

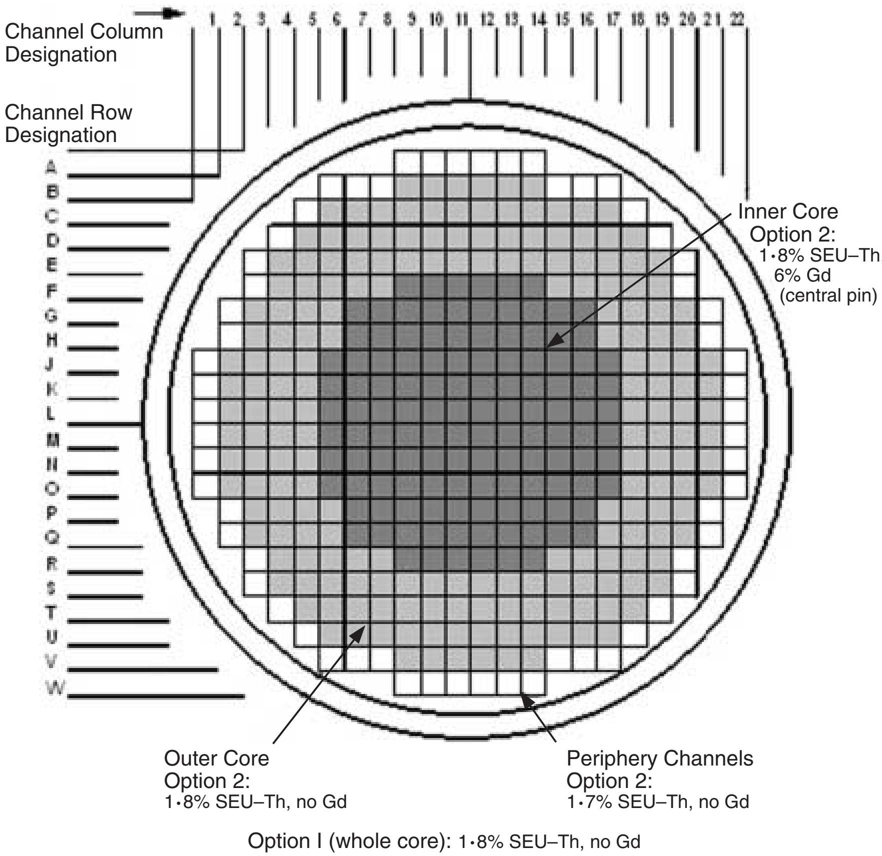


FIG. 201. Reactor core model of a CANDU 6 for the mixed bundle OTT cycle.

the 196 outer region channels is the same as that used in option 1. The fuel in the 124 inner region channels is identical to that in the outer region channels, except that the central ThO_2 element contains 6.0 wt% of gadolinium to shape the axial flux distribution. The gadolinium doped bundles are only used in the inner, adjuster rod region of the core. The 60 periphery channels contain thorium bundles designed to achieve burnups of over 50 MW·d/kg HE. These high burnup thorium bundles use natural ThO_2 in all 43 fuel elements. However, the initial fissile content in the outer 35 elements is increased from 0 wt% to 1.7 wt% using 20 wt% enriched uranium. These high burnup thorium bundles are located strategically at the edge of the core in order to utilize a large percentage of the leakage neutrons to produce power. This arrangement significantly increases the amount of thorium fuel in the core and improves the overall fuel efficiency of the thorium burning reactor.

(b) Lattice properties of mixed bundle options

Table L gives the initial fuel composition of the three types of thoria fuel bundle used in this study. The basic physics properties of these fuel lattices are shown in Fig. 202 (variation of lattice *k*-infinity), Fig. 203 (variation of fissile content) and Fig. 204 (variation of lattice void reactivity), as functions of bundle average fuel burnup. Although natural UO_2 and natural ThO_2 fuel bundles are not used in this study, their physics properties are also shown in these figures for the purposes of comparison.

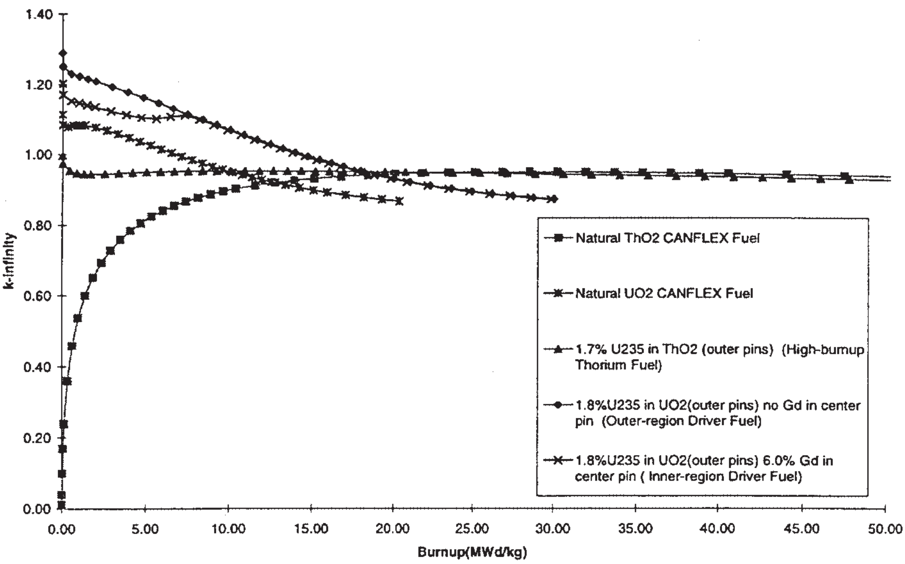


FIG. 202. Lattice k-infinity versus fuel burnup.

TABLE L. INITIAL FUEL COMPOSITION OF MIXED SEU–THORIUM BUNDLES

Ring number	Fuel compositon (kg/bundle)		
	1.8% SEU No Gd Whole core, option 1 Outer core, option 2	1.8% SEU 6 wt% Gd (central pin) Inner core, option 2	1.7% SEU No Gd Periphery channels, option 2
Ring 1:			
U-235			
U-238			
Th-232	0.510	0.510	0.510
Gd (natural)		0.0306	
Ring 2:			
U-235			
U-238			
Th-232	3.590	3.590	3.590
Ring 3:			
U-235	0.102	0.102	0.087
U-238	5.578	5.578	0.349
Th-232			4.700
Ring 4:			
U-235	0.153	0.153	0.131
U-238	8.367	8.367	0.523
Th-232			7.050
Total:			
U-235	0.255	0.255	0.218
U-238	13.945	13.945	0.872
Th-232	4.100	4.100	15.850
Gd (natural)		0.0306	

The initial fissile content of the high burnup thorium bundles in the core periphery in option 2 has been carefully chosen so that the depletion rate of the fissile material is almost the same as the conversion rate of fertile ^{232}Th into fissile ^{233}U. Consequently, the reactivity and the fissile content of the high burnup thorium bundles are almost constant throughout the entire lifetime. These high burnup thorium bundles can theoretically reside indefinitely in the reactor. The attainable fuel burnup is limited only by the mechanical integrity of the fuel bundle.

The main purpose of adding gadolinium to the central channels in option 2 is to shape the axial flux distributions such that the resultant axial bundle flux and power

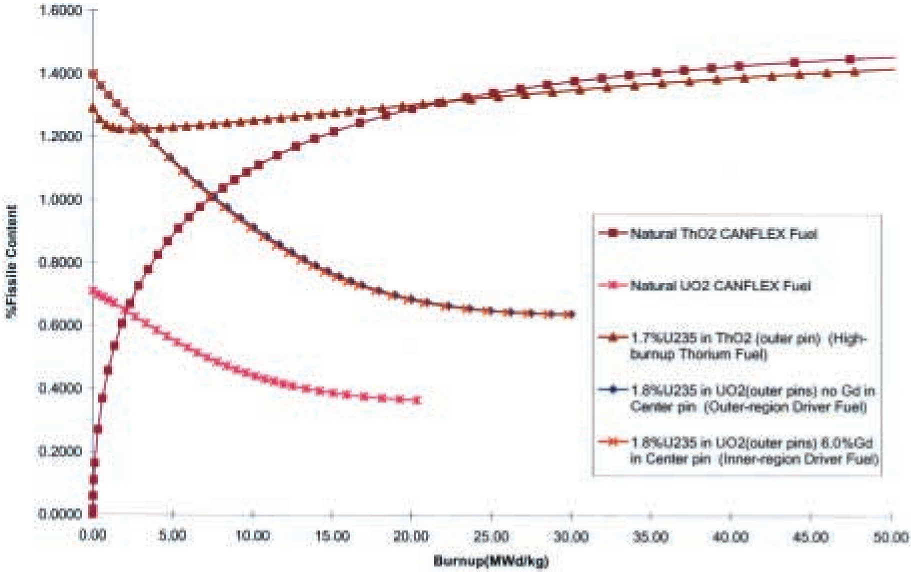


FIG. 203. Fissile content versus fuel burnup.

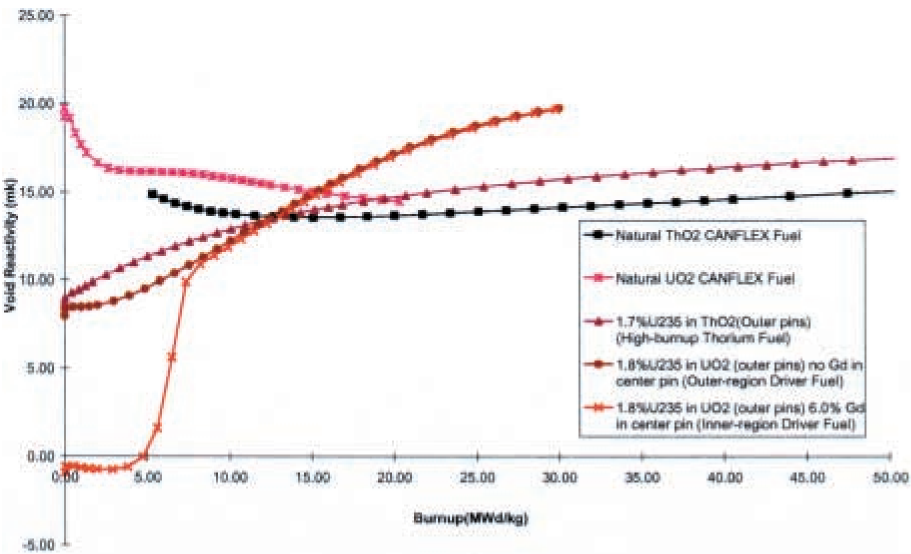


FIG. 204. Lattice void reactivity versus fuel burnup.

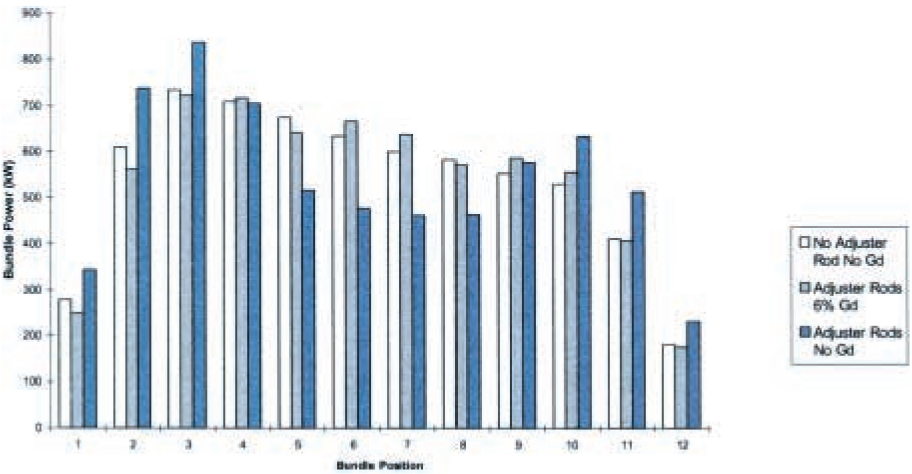


FIG. 205. Effect of adjuster rods and gadolinium on time average bundle power distribution.

distributions are similar to those in the thorium burning reactor without adjuster rods. Figure 205 shows that the gadolinium effectively eliminates the bundle power distortion caused by the adjuster rods. As expected, the effect of gadolinium on lattice reactivity is evident only during the initial stage of the fuel lifetime. The fast burnout rate of gadolinium suppresses the reactivity of the fresh bundle without incurring a significant burnup penalty over the lifetime of the fuel. This effect also reduces the channel and bundle power ripples caused by refuelling. The presence of a neutronic poison, gadolinium, in the central element also reduces coolant void reactivity [266]. This results in a significant reduction in the core averaged coolant void reactivity.

(c) Core characteristics of mixed bundle options

The RFSP code was employed using lattice properties taken from WIMS-AECL to perform time average core calculations for options 1 and 2 using a uniform two bundle shift fuelling scheme and a CANFLEX bundle. Instantaneous core calculations were conducted using randomly generated age patterns. Major reactor physics results for both options are summarized in Table LI. The fuelling rates, maximum channel power and maximum bundle power for both options are well within the limits established for current CANDU reactors using natural uranium fuel. There is also a significant reduction in the coolant void reactivity from that of a natural uranium reactor under comparable conditions. Option 1 gives 21% better uranium utilization than that of a natural uranium CANDU reactor. About half of the improved fuel efficiency is due to the removal of the adjuster rods. The other half can be attributed to

TABLE LI. SUMMARY OF WIMS-AECL/RFSP OTT FUEL MANAGEMENT STUDIES

Parameter	Core calculations: Option 1 (without adjuster rods)	Core calculations: Option 2 (with adjuster rods)
Uranium utilization (Mg natural U/GW(e)·a)	130	138
Volume of thorium in reactor core (%)	25	36
Core average fuel burnup (MW·d/kg HE)	22.1	20.3
Core average thorium burnup (MW·d/kg HE)	10.4	9.1
Fuelling rate (bundles/full power day)	5.5	6.0
Fuelling scheme (bundle shift)	2	2
Time average maximum channel power (kW)	6491	6505
Time average maximum bundle power (kW)	741	739
Instantaneous maximum channel power (kW)	6855	6849
Instantaneous maximum bundle power (kW)	785	781
Reactor leakage (mk)	30.6	25.8
Full-core coolant void reactivity (mk)	12.6	10.6

the energy produced in the thorium fuel. Option 2, which uses the existing adjuster rods, gives 14% better uranium utilization than that of a natural uranium reactor, with the additional advantage of a significantly lower coolant void reactivity.

Figure 206 shows that the channel power distributions of the two thorium burning reactors are very similar to those of a typical natural uranium CANDU reactor. Figure 207 shows that the axial bundle power distributions in the thorium burning reactors are flatter than those in a natural uranium CANDU and skewed towards the coolant inlet end. This skewed axial power profile should improve the thermohydraulic performance.

In summary, two options for implementing the OTT fuel cycle in existing HWRs were identified. In both options, uranium utilization is better than that of the natural uranium fuel cycle, but not as good as with SEU. The reactor and the fuel perform within existing envelopes without requiring that major modification be made to the current reactor design. Coolant void reactivity is significantly lower than that of a natural uranium reactor under comparable conditions.

(d) Effect of flux dependence on thorium physics calculations

The dependence of the lattice parameters of thorium bearing fuel on the flux and power history of the fuel has already been reported [267]. This dependence arises

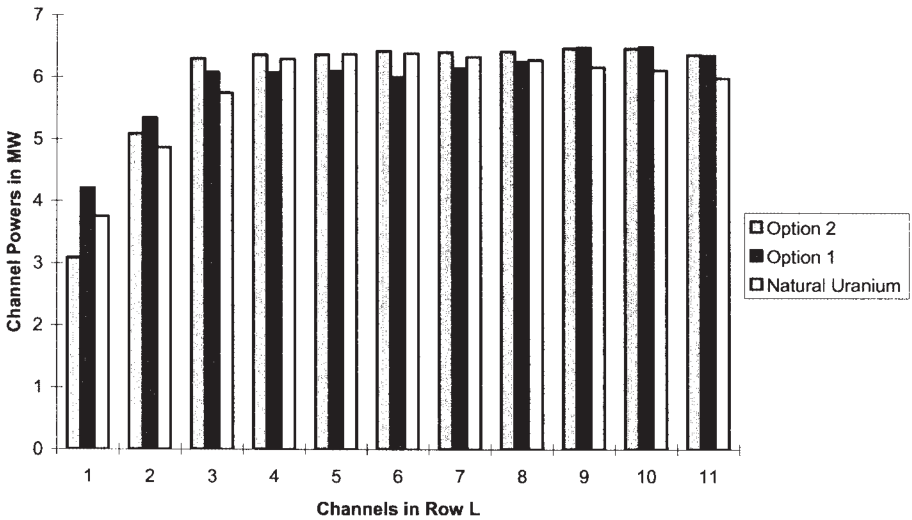


FIG. 206. Time average channel power distributions.

because the creation of the fissile isotope ^{233}U from the fertile isotope ^{232}Th is flux dependent. This process is analogous to the production of the fissile isotope ^{239}Pu from the fertile isotope ^{238}U in uranium based fuel cycles. The major difference is that the equilibrium level of ^{233}U in ^{232}Th is about 1.5%, and this is sensitive to the flux

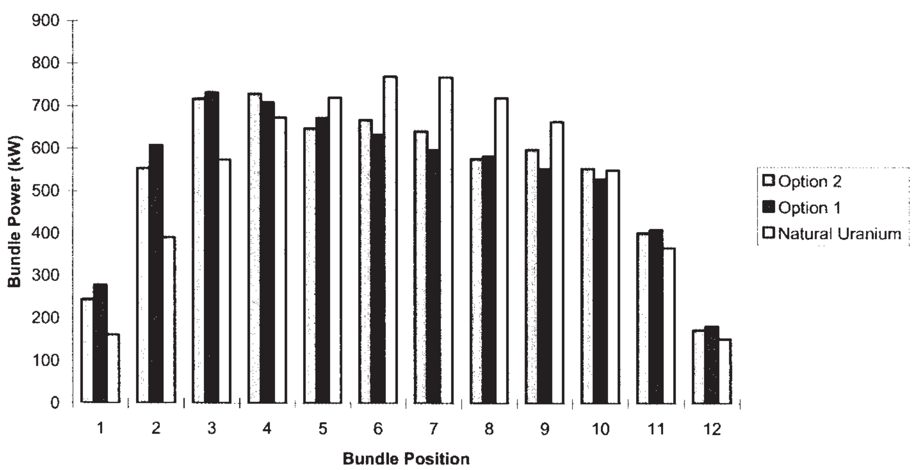


FIG. 207. Time average bundle power distributions in a high power channel.

level, whereas the equilibrium level of ^{239}Pu in ^{238}U is only about 0.4%, and this is relatively insensitive to the flux level.

The lattice parameters for the mixed SEU–thoria fuels detailed in the previous section were calculated with WIMS-AECL using a constant cell average thermal flux of 2.0×10^{14} n·cm^{-2}·s^{-1}. Those for the high burnup thorium fuel were calculated using a constant cell average thermal flux of 1.0×10^{14} n·cm^{-2}·s^{-1}. These flux levels are consistent with the flux levels obtained from RFSP core calculations. The results of the RFSP calculations are not expected to be significantly affected by the flux dependence of the lattice parameters because the thorium bearing fuel is placed in very low importance regions. The high burnup thorium fuel is located in the outermost channels and the thorium in the mixed SEU–thoria fuel bundles is limited to the two inner fuel rings, where the thermal neutron flux level is relatively low.

WIMS-AECL calculations were performed to assess the effect of flux dependence on the lattice *k*-infinity and fissile content of the mixed SEU–thoria fuel up to a core average fuel burnup of 22.1 MW·d/kg HE for option 1 and 20.3 MW·d/kg HE for option 2. Two methods of calculating the fuel burnup were used:

- Simulation of the movement of a thorium fuel bundle from the inlet to the outlet using a typical time average axial flux profile; the residence time at each position being calculated by RFSP.
- Use of a constant cell average thermal flux level consistent with the time average axial flux profile.

The results are summarized in Table LII. As is to be expected, there are small differences between these two sets of calculations. However, the variations in the ^{233}U content are not significantly greater than the variations in the fissile plutonium content. The sum of the ^{233}U and ^{233}Pa contents is the important quantity at discharge and is generally quite constant for each case type. In those cases which have the axial thermal neutron flux distribution modelled, the flux level drops appreciably during the final irradiation period. During this irradiation, the decay of ^{233}Pa to ^{233}U is enhanced and the level of ^{233}U at discharge is greater than that in the constant flux situation. This increases the *k*-infinity somewhat and also reduces the amounts of the other fissile nuclides that were used up. The maximum discrepancy in the lattice *k*-infinity is less than 10 mk at discharge. The uncertainty in the mixed SEU–thoria fuel lattice properties is expected to be small and should not have a significant impact on the results of the reactor core calculations.

The RFSP code was used to conduct time dependent refuelling simulations in the option 1 core for a period of 100 full power days using lattice parameters that were consistent with the power history of individual fuel bundles. These local lattice parameters were based on WIMS-AECL calculations using a very efficient computational scheme [268]. The maximum channel power and maximum bundle

TABLE LII. EFFECT OF FLUX DEPENDENCY ON LATTICE REACTIVITY AND FISSILE CONTENTS IN MIXED SEU–THORIUM FUEL BUNDLE

Option parameters	Value	
	WIMS calculations based on core average constant cell flux	WIMS calculations based on RFSP time average axial flux profile
Option 1:		
Core average burnup (MW·d/kg HE)	22.1	22.1
WIMS *k*-infinity	0.898	0.904
U-235 (g/kg HE)	20.6	20.5
U-233 (g/kg HE)	48.9	50.7
Pa-233 (g/kg HE)	4.6	3.1
Pu-239 + Pu-241 (g/kg HE)	46.8	47.1
Option 2 (outer region fuel):		
Core average burnup (MW·d/kg HE)	20.3	20.3
WIMS *k*-infinity	0.916	0.922
U-235 (g/kg HE)	26.6	26.6
U-233 (g/kg HE)	47.0	48.6
Pa-233 (g/kg HE)	4.2	2.8
Pu-239 + Pu-241 (g/kg HE)	46.6	46.8

power occurring during this simulation period using history dependent lattice parameters were 6.8 ± 0.1 MW and 780 ± 10 kW respectively. These results are very similar to those calculated by the RFSP code using traditional non-history based lattice parameters.

6.9.3.2. Direct self-recycle

Additional energy can be derived from the thorium by recycling the irradiated thoria fuel elements (which contain ^{233}U) directly, without any processing, into the centre of a new mixed bundle. Recycle of the central eight thoria elements results in an additional burnup of ~20 MW·d/kg HE from the thoria elements, for each recycle. The reactivity of these thoria elements remains remarkably constant during irradiation for each recycle. This direct, self-recycling results in a significant improvement in uranium utilization compared with OTT: after the first recycle, the uranium requirements are ~35% lower than those of the natural uranium cycle, and more than 10% lower than those of the optimal SEU cycle, and remain fairly constant with further recycling. The cumulative uranium requirement averaged over a number of cycles is 30–40% lower than that of a natural uranium fuelled CANDU reactor.

A key enabling technology in this direct, self-recycle option is the so-called 'demountable bundle', of which several embodiments of this concept exist. For many years, fuel researchers at AECL have made use of a demountable 37 element bundle for irradiation testing of advanced fuels in the NRU research reactor. The bundle is designed to enable any of the 18 elements in the outer ring of fuel to be removed, remotely, with the fuel bundle under water in the reactor spent fuel bays. Thus, elements can be removed at different burnups and new elements added to the bundle. Elements that develop defects can be removed, and the irradiation of the remaining elements continued. The design has proven to be durable and practical. A demountable CANFLEX bundle has recently been designed, and is undergoing qualification testing before use in NRU. This design provides even more flexibility than the demountable 37 element design, allowing access to the smaller elements in the outer ring and to the seven larger elements comprising the inner ring.

A final example of such technology is the 'advanced carrier bundle' [269]. This bundle was designed for irradiating specimens of pressure tube or calandria tube material in a commercial CANDU power reactor. In this bundle, two of the elements in the ring of six elements in a 37 element bundle are replaced by a tube occupying the space of those two elements. The tube is perforated to allow access to the coolant, and the specimens are mounted inside the tube. To achieve high neutron fluence, the bundle is designed to allow removal of the tube, once the bundle has been discharged into the spent fuel bays. The tube would be mounted into a fresh carrier bundle, under water in the bays, which would then be 'back fuelled' into the reactor to continue the irradiation of the specimens. This demonstrates the demountable bundle concept, its handling under water in the spent fuel bays, and its reintroduction into the fuelling machines and the reactor via back fuelling.

6.9.3.3. Other recycling options

The burnup, and the energy derived from the thoria elements, could be increased even further by removal, before recycling, of the rare earth, neutron absorbing fission products from the spent fuel. Conventional reprocessing, the so-called 'thorex' solvent extraction process, separates the uranium and the thorium from the fission products and other actinides. The radiation fields caused by the presence of ^{232}U (which emits copious amounts of α particles) and its daughter products (particularly ^{208}Tl, which emits a 2.6 MeV γ ray) provide a degree of self-protection and increase the proliferation resistance of recycled fuels containing ^{233}U. However, the absence of a commercial thorium recycling industry opens up the opportunity to develop a novel, simpler, more proliferation resistant recycle technology. For example, AECL has conceived a simple means of removing neutron absorbing, rare earth fission products, one which has a higher degree of proliferation resistance than the conventional thorex process and which would be much less costly.

In this process, the spent fuel would be dissolved in nitric acid and the pH adjusted. Uranium and thorium precipitate at similar values of pH, levels at which the rare earth fission products remain largely in solution (precipitation of these requires a much higher pH). By adjusting the pH of the solution, the uranium, thorium and some radioactive fission products can be removed from solution, leaving the parasitic, neutron absorbing, rare earth fission products behind.

The resultant fuel would be highly radioactive, and this processing, as well as the subsequent fuel fabrication and handling, would be done remotely. This would greatly enhance the proliferation resistance of the fuel, since it would have a distinct radioactive signature for monitoring and would prove difficult to access for purposes of diversion. Moreover, the simplicity of the CANDU fuel bundle design would facilitate processing, remote fabrication and handling, and reduce the cost relative to more complex fuel designs. This recycle option would be more expensive than either the simple OTT cycle, or the direct self-recycle of irradiated thoria fuel elements into new bundles.

The benefit of removing the fission products from the spent thoria fuel can be seen from the extensive studies that were performed on thorium recycling options in the 1970s and 1980s. In Ref. [270], a special version of WIMS-AECL was used to analyse and compare the resource utilization of various CANDU reactor fuel cycles, including once through natural uranium and SEU fuels, and recycle options based on both uranium and thorium. In these studies, conventional reprocessing was assumed, and all fission products were removed from the recycled fissile material. The thorium cycles considered only homogeneous mixtures of ThO_2 and fissile material, the initial fissile material being either ^{235}U or plutonium. The ^{233}U and any remaining fissile topping material were recycled from the spent fuel, and new fissile topping material were added to maintain burnup. A range of burnups was analysed. The results show that for thorium cycles, the largest improvements in uranium utilization are realized in replacement reactors that inherit the ^{233}U produced in reactors which initially use thorium. For such systems in equilibrium, savings in natural uranium requirements of up to 90% (compared with once through fuelling with natural uranium) were indicated.

Thorium reactors are an important part of India’s future nuclear programme, and the recycle of ^{233}U forms an essential part of ensuring a long term energy supply.

6.9.3.4. Self-sufficient equilibrium thorium cycle

The ultimate uranium conserving fuel cycle would be the self-sufficient equilibrium thorium cycle, in which no fissile topping (and hence, no natural uranium) would be required in equilibrium, i.e. the ^{233}U concentration in the recycled fresh fuel matches the ^{233}U concentration in the spent fuel [271]. Further improvements in neutron economy would be required to achieve this: reducing the fuel rating to lower the flux and hence neutron capture in ^{233}Pa, increasing the moderator purity, removing the adjuster rods from the core, enriching the zirconium used in the pressure and calandria tubes to remove most of the high cross-section isotope, ^{91}Zr. However, the following studies do not give credit for improvement.

The major shortcoming of the self-sufficient equilibrium thorium cycle is its low burnup, between 10 and 15 MW·d/kg HE, which will not be economic in a cycle which requires reprocessing and remote fabrication of the ^{233}U bearing fuel. To address this issue, a small amount of ^{235}U make-up could be added to each cycle, allowing the burnup to be increased as desired. Figure 208 shows the energy production in a ^{233}U 'near sustaining' cycle with ^{235}U make-up in the form of 0.9% SEU. Since there is no net destruction of ^{233}U, it can be assumed that all the energy is derived from the destroyed ^{235}U. As the discharge burnup increases, so the energy extracted decreases. Also shown in the same figure is a similar curve for SEU; the immense advantage of the thorium near sustaining cycle is clearly evident.

6.9.3.5. High burnup open cycle

The high burnup thorium open cycle avoids the issues related to closing the fuel cycle with reprocessing. In this cycle, the burnup is increased by trading off the conversion ratio. The thorium is enriched with ^{235}U to give whatever burnup the fuel can achieve. The spent fuel is not recycled (although this option would not be precluded). High burnup is equally possible with SEU, but the advantage of thorium over SEU lies in the fact that for very high discharge burnups, the initial fissile content required is lower with thorium fuel. Figure 209 shows Indian calculations of

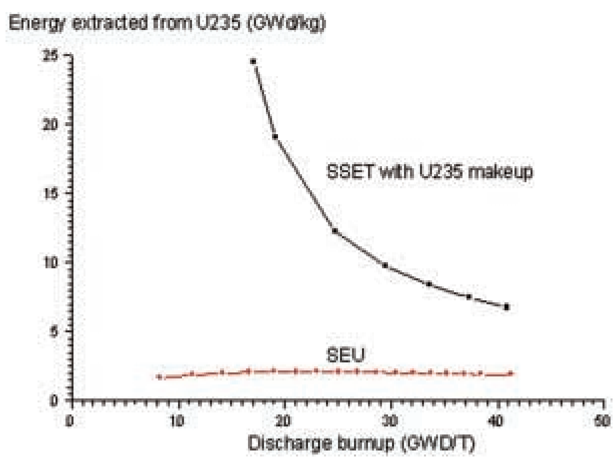


FIG. 208. Comparison of self-sufficient equilibrium thorium and SEU fuel cycles.

the discharge burnup as a function of initial ^{235}U content for SEU fuel, and for thorium topped with ^{235}U. In the case of low enrichments, SEU gives a higher discharge burnup for a given ^{235}U enrichment, but with very high discharge burnups, the enrichment required for the thorium fuel is lower than that required for SEU. In this theoretical assessment, pure ^{235}U is added to the thoria.

The main advantage of this thorium cycle compared with an equivalent enriched uranium cycle stems from the fact that as ^{235}U is burnt, so ^{233}U is built up, and as ^{233}U is a superior fissile material than ^{235}U, the reactivity versus burnup curve falls off more gradually with thorium than it does with enriched uranium. This means that to attain the same discharge burnup, the initial ^{235}U content can be lower in the thorium cycle. Consequently, power peaking problems are easier to manage with the thorium cycle. To achieve a discharge burnup of around 66 MW·d/kg HE, SEU requires an enrichment of 4.5%, whereas thorium needs only 3.5% (in total HE). Added to this is the fact that thermal neutron absorption in thorium is about three times that in ^{238}U, and that consequently the initial reactivity in the thorium core will be well below that of the SEU core for the same discharge burnup. This leads to lower reactivity swings, which is a definite operational advantage. This cycle is also an attractive method of plutonium annihilation (as discussed in Section 6.9.3.6), it having a very high plutonium destruction efficiency.

6.9.3.6 Plutonium–thoria as a plutonium dispositioning option

A special application for thoria that has recently received attention is its use as a matrix material for the annihilation of weapons derived plutonium [252, 272]. This is a responsible, forward looking strategy that uses plutonium to convert ^{232}Th to ^{233}U, which would be available as a future energy resource, if and when it is needed.

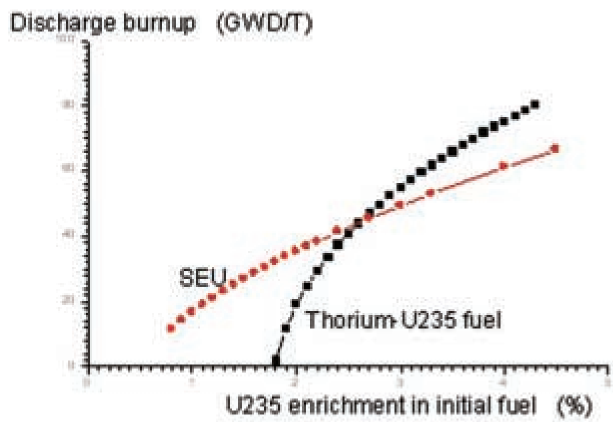


FIG. 209. Comparison of thorium and uranium cycles for high burnup.

The ^{233}U would be safeguarded in the spent fuel, benefiting from all the proliferation resistant features of spent UO_2 or MOX fuel. As noted above, the radiation fields caused by the presence of ^{232}U (which emits copious amounts of α particles) and its daughter products (particularly ^{208}Tl, which emits a 2.6 MeV γ ray), provide a high degree of self-protection and render ^{233}U unattractive as a weapons material. The ^{233}U could be recovered in the future using a proliferation resistant technology, when warranted by the price of uranium and other factors.

The assessment of Pu–ThO_2 for plutonium management was limited to reactor physics lattice calculations, using the multigroup lattice code WIMS-AECL. Actinide inventories were calculated using a fully coupled multiregion WIMS-AECL/ ORIGEN-S code package [273]. Reactor calculations and fuel management simulations were not performed. However, given the CANDU flexibility in fuel management, no technical feasibility issues are anticipated.

A somewhat different approach was taken in designing the Pu–ThO_2 fuel bundle for this application. To maximize the destruction of the plutonium, good neutron economy was desired. A reduction in void reactivity was also sought, to compensate for the faster dynamic behaviour of the fuel (shorter neutron lifetime and smaller delayed neutron fraction). To achieve these two objectives, the central elements in a CANFLEX bundle were replaced with a large central graphite displacer. Plutonium at 2.6% (354 g per bundle) was mixed with thorium in the remaining 35 elements in the two outer fuel rings of the CANFLEX bundle. Enrichment grading in the outer two fuel rings would result in peak element ratings comparable to those in a 37 element bundle with natural uranium fuel. The resultant burnup was 30 MW·d/kg HE, a burnup for which there is research reactor experience with Pu–ThO_2 fuel. Void reactivity was 8.6 mk, which is judged to be acceptable with the current shutdown system. Computer simulations also showed that using SiC instead of graphite in the central displacer reduces the magnitude of the void reactivity somewhat.

Addition of a small amount of burnable poison to the central displacer would further reduce void reactivity, increase the plutonium loading per bundle as well as the absolute amount of plutonium destroyed, but would decrease the plutonium destruction efficiency. The plutonium destruction efficiency would be reduced from about 77% to 71% by poison addition that reduces void reactivity from about 8.6 mk to zero.

Table LIII shows the composition of the fresh and spent fuel: 77% of the total plutonium is destroyed, and 94% of the fissile plutonium, a destruction efficiency similar to that for plutonium annihilation in an inert matrix (see Section 6.8). Fissile ^{233}U (including its parent ^{233}Pa) is produced to the extent of 168 g per bundle, and can be recovered and recycled in a proliferation resistant fashion. For example, the ^{233}U, remaining plutonium and ^{232}Th could be co-extracted, without separating the ^{233}U, and fresh weapons derived plutonium added to maintain burnup, and further destroy plutonium stockpiles. This option achieves high plutonium destruction

TABLE LIII. FUEL COMPOSITION FOR PLUTONIUM–THORIA

Isotope	Fuel composition (g/bundle)	
	0.0 MW·d/kg HE	30 MW·d/kg HE
Pu-238	0.2	0.3
Pu-239	331.3	5.3
Pu-240	21.2	45.2
Pu-241	1.4	14.5
Pu-242	0.2	15.9
Total Pu	354.3	81.2
U-233 + Pa-233		167.7
Np-237		0.004
Am-241		0.59
Am-243		2.21
Cm-242		0.26
Cm-244		0.42

efficiency and has a high energy yield, even without recycling the ^{233}U. Recycling the ^{233}U would increase the energy yield manyfold.

6.9.4. Power plant optimization for thorium use: The Indian AHWR

India is presently working on the design of an AHWR designed specifically for utilizing thorium, including reprocessing of spent thorium and refabrication of $(Th,^{233}U)O_2$ fuel. The main constituent of the core is fuel made of mixed oxides of thorium and ^{233}U. The reactor will be of the pressure tube type, with boiling light water used as coolant. The discharge burnup of the $(^{232}Th,^{233}U)O_2$ can be increased later on, depending upon fuel performance limits. The ^{233}U enrichment in the thorium is so adjusted that the system will be self-sustaining in ^{233}U. Since the burnup of the $(^{232}Th,^{233}U)O_2$ is planned to be increased to fuel performance limits, and the ^{233}U enrichment is determined by self-sustaining considerations, it is clear that this lattice would be subcritical. In order to make the reactor critical, a few driver zones are provided; these regions consisting of highly reactive $(Th,Pu)O_2$ fuel. This region forms a part of each cluster. In order to obtain negative void reactivity, and also to obtain the benefit of a hard spectrum for plutonium, the lattice pitch has been reduced.

This reactor also has 100% heat removal effected by natural circulation, and passive safety is ensured by engineered safety features such as gravity driven water pool, isolation condenser, and large volumes of water that can totally submerge the core in the event of an accident.

The design objectives of the AHWR are, briefly, as follows:

- About 75% of the power produced by the AHWR should be contributed by thorium,
- The system should have a negative void coefficient of reactivity,
- The discharge burnup of the fuel should be more than 20 MW·d/kg HE,
- The plutonium consumption and initial plutonium inventory should be as low as possible,
- The system should be self-sustaining in ^{233}U,
- The total thermal power of the reactor should be 750 MW(th).

The objective stating that the system should be self-sustaining in ^{233}U indicates that the starting point will be the self-sufficient equilibrium thorium cycle described in Section 6.9.3.4. In order to extend the burnup of the self-sufficient equilibrium thorium cycle, plutonium is used as make-up fissile material. Plutonium-239 has a rather high capture to fission ratio in the soft spectrum of the HWR. In order to place plutonium in a more favourable spectrum, the lattice pitch is reduced, resulting in an undermoderated spectrum compared with that of the HWR.

Thorium oxide has very good fuel performance characteristics, and is capable of achieving very high burnups. Since this has to be matched by reactivity considerations, the initial plutonium enrichment could be very high. This would have the undesirable consequence of too high a fraction of the power being derived from plutonium. This has been addressed by concentrating the plutonium in a small number of pins. In the AHWR cluster, which comprises 54 rods, only 24 pins contain plutonium, the other 30 being $(^{232}Th,^{233}U)O_2$. The characteristics of the AHWR in its current stage of evolution is given in Table LIV, while the fuel assembly is detailed in Table LV and illustrated in Fig. 210.

The thermal absorption of thorium is three times that of ^{238}U. As a result of this, the deleterious effects of parasitic absorption are less in thorium systems, and therefore the use of light water coolant can be considered. This opens up the possibility of in-core boiling. In order to design for 100% heat removal by natural circulation and for passive safety, the reactor has to be positioned vertically. The possibility of registering a positive void coefficient of reactivity has been countered by the lattice being undermoderated, using a burnable absorber in the fuel cluster.

In summary, the Indian nuclear power programme aims to maximize the energy potential of its indigenous resources. Various domestic fuel cycle related development programmes address this objective. Since India has abundant thorium reserves, efforts are being made to increase the energy extracted from thorium. Thorium utilization in Indian HWRs has already been initiated with the use of ThO_2 bundles for initial flux flattening in PHWRs. The proposed AHWR is another step towards the establishment of thorium cycles.

TABLE LIV. SPECIFICATIONS OF THE AHWR AT THE PRESENT STAGE OF EVOLUTION

Parameter	Value
Reactor power (MW(th))	750
Reactor power (MW(e))	220
Fuel description:	
Number of pins	54
Number of Pu bearing pins	24
Number of Th–U-233 pins	30
Number of burnable absorber rods incorporating water tube for ECCS injection	1
Plutonium content in MOX (%)	3.0
U-233 content in thorium	Self-sustaining
Average coolant water density (g/cm^3)	0.50–0.55
Total number of channels	452
Number of fuelling zones	3
Discharge burnup (MW·d/kg HE)	~24
Lattice pitch (cms)	29.4
Active fuel length (cms)	350
Moderator and reflector	D_2O
Scatterer in the moderator	Pyrocarbon
Calandria radius (cms)	430
Number of adjuster rods	12
Worth of adjuster rods (mk)	15
Number of SDS1 rods	36
Worth of SDS1 (mk)	56.3
Worth of liquid poison (SDS2) (mk)	62.3
Performance data for equilibrium core:	
Radial form factor	1.37
'Hot spot' factor in the cluster	1.3
Maximum channel power (MW)	2.30
Maximum to minimum channel power factor	2.16
Fraction of power from thorium (%)	74.9

6.9.5. Waste management aspects

6.9.5.1. Physical properties

Thoria based fuels are also attractive from a waste management perspective because ThO_2 is chemically stable and almost insoluble in groundwater. By far the most important chemical difference between ThO_2 and UO_2 is the fact that thorium

TABLE LV. SPECIFICATIONS OF THE AHWR FUEL ASSEMBLY

Parameter	Value
Fuel pellet radius (cm)	0.49
Sheath outer radius (cm)	0.56
Pellet cladding gap (mm)	0.1
Number of pins in cluster	54
Number of plutonium bearing pins	24
Number of Th–(U-233) pins	30
Fuel pellet density (g/cm^3)	9.6
Pin arrangement	Three concentric rings
Cladding thickness (cm)	0.06
Coolant tube inner radius (cm)	6.0
Coolant tube outer radius (cm)	6.36
Calandria tube inner radius (cm)	7.7
Calandria tube outer radius (cm)	7.9
Cladding material	Zircaloy
Coolant tube material	Zr–2.5wt%Nb
Calandria tube material	Zircaloy
Active fuel length (cm)	350
Mass of HE in an assembly (kg)	120
Coolant material	Light water
Average coolant density (g/cm^3)	0.5–0.55
Coolant in water tubes	Light water
Water tube water density (g/cm^3)	0.771
Burnable poison rod outer diameter (cm)	3.8

is present in its maximum oxidation state, Th(IV), whereas uranium is not. Therefore, oxidative dissolution of the matrix is not an issue with thoria fuel. Redox conditions could affect the leachability of ^{233}U from irradiated thoria, but this would be limited to surface dissolution and is unlikely to be a major concern.

The inertness of thoria to oxidation is also relevant to the interim dry storage of irradiated fuel prior to geological disposal. In contrast with UO_2, air oxidation of the fuel matrix in defective elements is not an issue with thoria based fuels. Moreover, the thoria structure can easily accommodate oxidation of minor solid solution components such as uranium and plutonium. Thus, fuel oxidation is unlikely to be a concern during the dry storage of thoria based fuels, and hence the maximum storage temperature would be limited by some other factor, probably cladding degradation [274].

The solubility of crystalline thoria in aqueous solution at 25°C and pH > 5, and in the absence of complexing agents, is extremely low. The release of actinides and those fission products that are retained by the thoria matrix is expected to be limited by the solubility of ThO_2. Such release would be exceedingly slow in an engineered

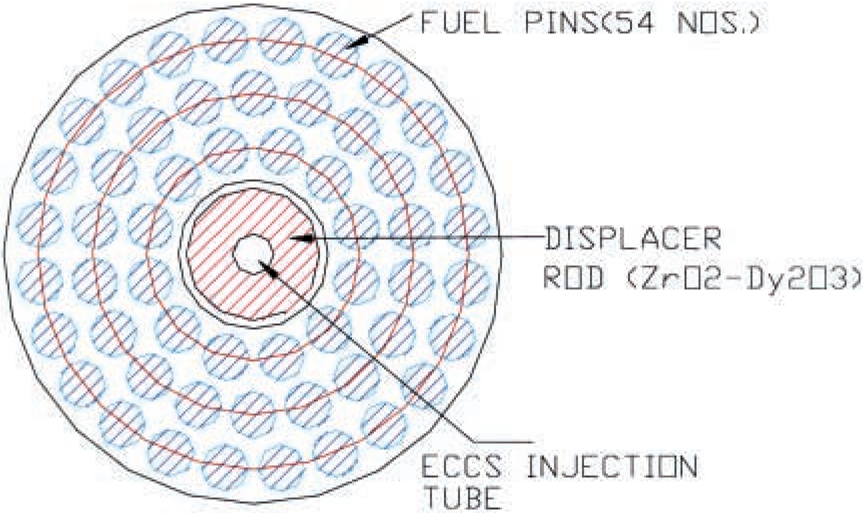


FIG. 210. Cross-section of AHWR fuel cluster.

disposal vault of the type envisaged for CANDU UO_2 fuel. No credible aqueous or geochemical process has been identified that would greatly accelerate ThO_2 fuel matrix dissolution under disposal conditions [275].

Thoria crystallizes with the fluorite structure, as do all other actinide dioxides. Extensive solid solution occurs between these oxides, and the fluorite structure can also accommodate substantial levels of actinides in other oxidation states, as well as many fission products. Thus, no phase segregation of actinides is expected to occur within the fuel, either during operation or after disposal, and it is reasonable to assume that release of actinides will be controlled by the slow dissolution rate of the thoria matrix, provided that the fuel is initially homogeneous.

Calculated environmental releases and subsequent radiation doses arising from a CANDU UO_2 fuel disposal vault are dominated by the 'instant' release of soluble and mobile fission products (in particular ^{129}I) from the fuel sheath gap of the fuel. Grain boundary inventories may also be released rapidly, as compared with matrix dissolution. It is likely that similar findings would emerge from a detailed assessment of thoria fuel disposal, especially given the expectation of extremely slow matrix dissolution. Therefore, it is important to consider the irradiation history and microstructural behaviour of the fuel, and to have reliable information on the segregation of mobile fission products to the gap and grain boundaries in thoria based fuels.

Grain growth in the central region of fuel pellets is a major cause of fission gas release to the gap, because the gases and other incompatible elements are swept from their original resting places in the fuel matrix and become concentrated at the grain boundaries. There, they form features such as fission gas bubbles and noble metal particles [276]. Interlinkage of fission gas bubbles on grain boundary intersections eventually creates tunnels that permit venting of other fission products to the fuel cladding gap. Thoria is a better thermal conductor than UO_2; it also has a higher melting point and experiences slower cation diffusion. Therefore, for a given power rating and fuel geometry, it would be expected to run cooler and to undergo less grain growth.

Fission gas release rates are expected to be somewhat lower for thoria based fuels than for UO_2 fuels with comparable geometry, microstructure and power history. This conclusion is based on the lower diffusion rate of xenon in ThO_2 compared with UO_2 and the smaller burst release in ThO_2 [277–279]. The expected low gas release rates from thoria based fuels are supported by in-pile experiments on ThO_2 and $(Th,U)O_2$ fuel assemblies. Goldberg et al., measured fission gas release in a set of 51 thoria based fuel rods over a range of linear powers, burnups and compositions [280, 281]. The authors gave an expression for the rate of fission gas release, which suggests that rates are significantly lower than for UO_2 under comparable operating conditions.

In many cases, the segregation and, hence, the leachability of volatile, non-gaseous fission products, such as caesium and iodine, is correlated with fission gas release and thus the release of these fission products is expected to be lower for a thoria based fuel than for one based on UO_2 [282, 283]. Jones et al., reported low gas releases for $(Th,U)O_2$ fuels, and also noted that fission product release from defected thoria elements was between one and two orders of magnitude lower than for UO_2 [284]. Experimental data obtained by Matzke supports this notion; Ref. [285] reports that the release of bromine, caesium and rubidium from thoria was generally slower than from UO_2.

Diffusion of fission products in UO_2 and ThO_2 remains poorly understood, but generally appears to involve uranium or thorium ion vacancies. The results of high temperature, out of pile annealing experiments on lightly irradiated or ion implanted samples appear to be consistent with modestly lower fission product diffusion rates in ThO_2 than in UO_2 — roughly paralleling the difference in cation lattice diffusion [276, 285–288]. Fission product migration in-reactor involves further complexity. Indeed, Matzke has suggested that five different diffusion coefficients are required to model fission gas transport [276]. Nonetheless, the overall trend is evidently maintained: under equivalent operating conditions, fission product segregation and release tend to be lower for ThO_2 fuels than for UO_2 fuels.

In summary, the high degree of chemical stability and the low solubility of thoria make irradiated thoria based fuels attractive as waste forms for direct geological disposal. Moreover, there is good reason to expect lower fission gas releases (and correspondingly lower gap and grain boundary inventories of other fission products) in thoria fuels than in UO_2 fuels with a comparable power history. In order to realize these beneficial qualities of thoria based fuels, an appropriate fuel fabrication process must be utilized to achieve an acceptable degree of microscopic homogeneity. Detailed post-irradiation examination and leaching studies of thoria based fuels, coupled with a thorough understanding of their physical and chemical properties, are needed to support these preliminary conclusions.

6.9.5.2. Radiotoxicity

One of the advantages that is often cited for the thorium fuel cycle is the significant reduction in the production of long lived transuranic actinides. It has lower 'radiotoxicity' than uranium based fuels, and therefore the source term in the waste management vault will be lower.

However, in assessing the risk to humans and to the environment posed by radioactive spent fuel, it is important to consider not only the source term, but also the pathways to exposure. Thus, in the Canadian concept for the geological disposal of spent fuel, the actinides are not the major contributor to risk after tens or hundreds of thousands of years. This is because they are relatively insoluble in the reducing environment of the vault, and are transported only slowly if they do dissolve.

Nonetheless, many proponents of the thorium cycle promote this feature, and indeed, many detractors of the nuclear industry cite the long term radiotoxicity of spent uranium based fuels as a major issue.

6.9.6. Non-proliferation and safeguards considerations

Proliferation resistance aspects were discussed for each of the thorium fuel cycle options described in Section 6.9.3. The OTT cycle (Section 6.9.3.1) does not involve thorium recycle, and therefore has a very high degree of proliferation resistance, as has the once through, high burnup thorium fuel cycle (Section 6.9.3.5). The direct self-recycle option applied to the OTT cycle (Section 6.9.3.2) would involve the recycle of irradiated thoria fuel elements into new fuel bundles at the reactor site. With no alteration of the elements or pellets, this fuel reconfiguration, recycle option would also have a very high degree of proliferation resistance.

In the context of fission product removal from spent thoria fuel as a means of further increasing the energy derived from the recycled ^{233}U, the absence of a commercially established facility or even process for reprocessing thorium fuel opens up the possibility of incorporating the highest degree of proliferation resistance right from the start in the design of such a process. For example, rather than extracting ^{233}U from the spent fuel, only to mix it with fresh ^{232}Th, the thorium/uranium from the spent fuel could be co-precipitated, and only the high neutron aborbing rare earth fission products removed (Section 6.9.3.3).

Finally, even with the thorex flowsheet for thoria reprocessing, which results in separated ^{233}U, a degree of proliferation resistance is provided by the presence of ^{232}U in the spent fuel, which is always present with the ^{233}U and which renders the ^{233}U unattractive for diversion. Uranium-232 is a copious emitter of alpha particles, an undesirable feature for a weapon. Moreover, the decay chain following alpha emission from ^{232}U to produce ^{228}Th includes ^{212}Bi and ^{208}Tl, both strong gamma emitters. The presence of these two isotopes makes ^{232}U a difficult material to handle and thereby to divert (although the ^{208}Tl daughter product builds in with a half-life of several years, and therefore would not be present immediately in chemically separated uranium).

Finally, if the thorium is mixed with ^{238}U to begin with, the ^{233}U that is produced will be contaminated with ^{238}U, from which it cannot easily be separated by chemical methods. The proportions of thorium and ^{238}U can be so adjusted that the final uranium does not have a fissile content exceeding 12%, or whatever other value is felt to be acceptable. It is difficult to come up with a similar solution for the plutonium produced in the uranium cycle. If the thorium cycle involves the addition of topping material to maintain a desired burnup, the use of 'denatured' material would be another feature that would enhance the proliferation resistance of the cycle.

6.10. HWR/FBR SYNERGISTIC FUEL CYCLES [289]

In the very long term, the availability of economic nuclear fuel resources can be ensured through the use of FBRs. However, their high cost relative to thermal reactors dictates that their principal use would be in satisfying the fissile requirements for lower cost, high conversion, thermal reactors — the most efficient in this respect being the HWR. Each FBR could satisfy the fissile requirements of about eight HWRs. There is sufficient fertile material to fuel the FBRs indefinitely, in the form of depleted uranium from enrichment plant tails. Alternatively, the FBR could use thorium as the fertile material, producing ^{233}U which could be very effectively burned in the soft HWR spectrum.

Since such synergistic systems have not yet been developed, there is an opportunity to ensure that a holistic approach to this fuel cycle is used, one that optimizes all key criteria. Proliferation resistance should be an objective from the start. For example, the FBR could contain an integral fuel processing unit, which would produce both its own fuel and the enriched fuel for the HWR cycle. The direct recycle of fuel from one reactor type to the other may also be feasible (a variant of the DUPIC cycle, or the direct self-recycle thorium fuel cycle option, applied to the HWR/FBR system).

It should be noted that the type of thermal reactor should have a high conversion ratio, in order that as many cheap, thermal reactors could be operated on the fissile material produced by the single, expensive FBR. In this manner, the HWR provides a bridge to long term energy security. Although these concepts are still in their infancy, they would form a fruitful area for international collaboration.

6.11. SUMMARY OF HWR FUEL CYCLE STRATEGIES AND TECHNOLOGY DEVELOPMENTS REQUIRED

This report describes the advanced fuel and fuel cycle options that can be exploited in the HWR. The general time-frame for deployment, and the status of technological development, have been indicated for each fuel cycle option. The key fuel cycle drivers that will determine the choice, and timing, of advanced fuel cycles have been discussed in Section 6.2.2. It is worth restating that the particular fuel cycle chosen by any country will depend on a range of local and global factors, and that there is no unique fuel cycle path appropriate for all countries. The HWR provides the fuel cycle flexibility to enable any country to tailor its fuel cycle to fit its own unique requirements. Moreover, advanced fuel cycle technology provides a means of achieving significant improvements: reduction in capital cost; reduction in fuel cycle costs, both front and back end; reduction in spent fuel volumes; enhancement in passive safety; means of extending plant life; increases in operating margin; extension of uranium reserves and, ultimately, independence from natural uranium; and a means of achieving specific national or global objectives, such as the timely and effective dispositioning of ex-weapons plutonium, and plutonium and actinide burning.

This section summarizes likely, or possible trends in the implementation of advanced fuels or fuel cycles in commercial power reactors over the time-frames indicated. The level of confidence in the projections, of course, decreases as they recede further into the future.

Finally, while some fuel cycle options will be introduced into operating reactors on the basis of their own merit, others will be introduced in new reactors as part of an integrated reactor design.

6.11.1. The next ten years

This time-frame will see more HWRs reaching ‘middle age’, and advanced fuel designs, such as CANFLEX and CARA, may be used by operating utilities, partly to increase operating margins (particularly, critical channel power). In the specific case of CANFLEX, the bundle is currently undergoing a small scale demonstration irradiation in a power reactor; all development and qualification testing are nearing completion, prior to large scale, commercial implementation. Further improvements to the bundle may take place over this time-frame, to further enhance the critical channel power margins.

Enrichment may gradually be introduced into operating stations, with the initial enrichment level being around 0.9%. Recycled uranium from reprocessed spent PWR fuel may also witness large scale usage, if the costs are significantly lower than those for SEU. Both SEU and recycled uranium are expected to provide compelling benefits to operating stations:

- Reduced fuel cycle costs (which will become a very important factor in reducing operation, maintenance and administration costs in an increasingly competitive, deregulated electricity supply market);
- Improved operating margins (specifically thermohydraulic) and, if necessary, reduced void reactivity through the low void reactivity fuel configuration;
- Power uprating capability (when the reactor has sufficient heat removal capability — an attractive opportunity to be considered when retubing);
- Reduced quantities of spent fuel produced.

Very little development is required for SEU with an enrichment of around 0.9%. Further in-reactor, power ramp performance data at extended burnup will be accumulated in research reactors, and this will support even higher enrichments and burnups, since the database at extended burnup is sparse (although it must be recognized that the vast majority of power ramp data are acquired through power reactor experience, not through irradiations in research reactors). SEU fuel with around 0.9% ^{235}U will undergo full qualification. A small scale demonstration irradiation in a power reactor should commence within five years. In the case of recycled uranium, more development may be undertaken on the conversion process (to ceramic grade UO_2). Depending on the conversion route, some limited in-reactor testing in a research reactor of pellets produced through the chosen conversion route may be prudent.

In new reactors, SEU will be used to optimize HWR reactor design, in order to minimize capital costs. One means of achieving this will be through flattening of the channel power distribution across the core, deriving more power from a given size of reactor. This will still enable the use of natural uranium fuel, although possibly at the risk of incurring a penalty. More aggressive capital cost reductions in new plants will also be enabled over this time-frame through the use of enrichment, such as a reduction in heavy water inventory through the use of light water coolant and a tighter lattice pitch.

A small scale demonstration irradiation of SEU in an operating station, and preferably, conversion to a full core of SEU, will provide utilities with the confidence to take advantage of new reactors optimized for the use of SEU. Hence, HWR power reactor experience gained with SEU will be a very important step.

It is still possible that MOX fuel, containing plutonium from dismantled warheads, will be used in the Bruce reactors in Canada, towards the end of this period. A fuel qualification programme has been developed that is consistent with this time-frame. Burnup initially would be within natural uranium experience (~10 MW·d/kg HE), and may progress to higher values, depending on the overall economics. However, if this mission goes ahead, the timing will be determined by political, not technical, considerations.

Over this time-frame, R&D will continue to provide the technological basis for more advanced fuel cycles. It is likely that international collaboration will become increasingly important in advancing these fuel cycles. For example, an international centre for thorium fuel cycle research could be envisaged. India will continue to acquire experience in the irradiation of thoria fuel bundles for reactivity suppression in new reactor cores. Technology related to the front and back end of the thorium fuel cycle will be developed over this period. Reactor physics data for thorium lattices, including improved cross-section measurements, will be extensively generated during the verification of the Indian AHWR design.

Research will continue on the DUPIC fuel cycle, to establish technical feasibility. The next phase of DUPIC development, prior to commercialization, will depend on national, strategic considerations, as well as on the results of the current assessment, and the costs of alternative technologies. Extensive development of remote processing and fabrication technologies, safeguards technology and in-reactor irradiations will be required, at a fairly intense pace, and at an international level, in order for DUPIC to be commercialized. Research will be undertaken on other advanced recycling options having a higher degree of proliferation resistance than conventional reprocessing, driven by the potential for simpler and cheaper technologies.

Research on actinide burning in the HWR will remain at a low level, until a specific client is identified.

6.11.2. Ten to twenty years

It is anticipated that this time-rame will witness the widespread use of SEU/recycled uranium in HWRs. Enrichment may increase to 1.2% SEU, driven in part by lower enrichment costs and advanced enrichment technologies. This high enrichment will follow naturally from the experience gained with 0.9% SEU. Power ramp testing will have been performed in the previous time-frame.

Advanced HWRs will be built in this time-frame. The first Indian AHWR will provide valuable experience with the use of thorium based fuel.

Testing will be done on the next generation of advanced fuel bundles, which are likely to feature greater subdivision (as much as 61 elements), high burnup capabilities, higher sheath temperatures (although fuel operating temperatures will be lower, either through low linear element ratings resulting from greater subdivision, or the use of graphite discs), and other advanced features.

Beyond this, particular fuel cycle developments will be driven by national considerations. DUPIC could be commercialized in this period. Similarly, MOX from recycled LWR fuel could be implemented in this time-frame in specific countries, in HWRs optimized for this purpose.

In countries such as India, having thorium but only limited uranium resources, limited implementation of thorium fuel cycles, such as the OTT cycle, could be envisaged. This will require development of the infrastructure for commercial thoria fuel fabrication (from mining and refining, to fuel fabrication and continued irradiation testing).

6.11.3. Options beyond twenty years

Advanced HWRs, having the target of half the total unit energy cost of current plants, would be implemented in this time-frame. Advanced fuel designs will be central to these advanced HWR designs, for example, HWR fuel allowing high sheath temperatures to achieve higher thermodynamic efficiency through higher coolant temperature; negative void reactivity to enhance passive safety; ‘cool fuel’ features which result in low fuel temperatures; and advanced welding techniques (such as resistance brazing). These fuel designs will evolve as advances are incorporated into the fuel designs of operating plants. The trend will be towards higher power output, while increasing operating and safety margins.

Other developments will be strongly influenced by local and national strategic considerations. High burnup MOX (using new recycling technologies specifically exploiting the HWR–PWR synergism) and high burnup SEU will likely be used. Thorium cycles employing novel recycling technologies may be employed. The first demonstrations of HWR–FBR synergism would not occur before this time-frame. Inert matrix fuel for actinide burning in HWRs could be implemented if conventional reprocessing of spent PWR fuel continues.

In India, a reactor system which is optimized for utilizing fissile material generated in the Indian FBR programme will be under design and development. Most of the input required for developing these new reactor systems will come from the experience gained with operating AHWRs.

6.11.4. National perspective: India

Given its unique situation, the Indian strategy for nuclear power deployment is summarized separately. This strategy is primarily based on nuclear fuel resources available within the country. India has modest uranium reserves, but abundant thorium reserves. A three stage nuclear power programme has been drawn up by the Department of Atomic Energy with the aim of maximizing the energy potential of indigenous resources. The three stages of this programme incorporate HWRs based on natural uranium fuel, FBRs utilizing plutonium obtained from HWRs as fuel along with thorium in blankets for producing fissile ^{233}U, and advanced reactors mainly based on the Th–^{233}U fuel cycle. In the Indian power programme, reprocessing and recycling in thermal/fast reactors are viewed as being integral parts of the fuel cycle.

The broad policy and actual implementation differ at some points owing to the fact that some Indian reactors are under international safeguards. These reactors are

Rajasthan 1 and 2, built with Canadian assistance, and Tarapur 1 and 2, built as a turnkey project by General Electric (USA) and using imported enriched uranium. The plutonium produced from safeguarded reactors will only be used in safeguarded reactors; thus, the plutonium extracted from the spent fuel of Rajasthan 1 and 2 is being used at Tarapur 1 and 2 (plutonium recycled as MOX fuel in BWRs). This provides an opportunity to 'shrink' the spent fuel volume in safeguarded reactors and to gain valuable experience in MOX fuel technology. Studies regarding the utilization of plutonium from spent HWR fuel are also to be seen in the same context. These studies indicate that the equivalent ^{239}Pu value is not reduced significantly with one recycle in thermal reactors.

Ultimately, the plutonium from spent HWR fuel will be utilized as fuel for FBRs in India. This forms the second stage of the Indian nuclear power programme. The Fast Breeder Test Reactor has operated well with a UC–PuC core for the past several years. The design for the Prototype Fast Breeder Reactor (PFBR), which will have an output of 500 MW(e), is at an advanced stage. The PFBR will use MOX (UO_2–PuO_2) fuel. The construction of the PFBR is expected to start soon and will prepare the way for future FBRs. A series of FBRs is needed for the multiplication of fissile inventory (both plutonium and ^{233}U).

The utilization of thorium is at present limited to initial flux flattening in HWRs. Various thorium based cycles in HWRs are being studied. The irradiations in HWRs are expected to allow a full understanding to be gained of all the activities in the thorium fuel cycle. To sustain the technological base of pressure tube type reactor technology and to master thorium fuel cycles on an industrial scale, the design of the AHWR is being pursued. The AHWR is expected to address the problems of thorium fuel cycle technologies well in advance of the third stage of the Indian programme, and to provide the opportunity to gain operating experience with the fuel cycle facilities. The feasibility report for the AHWR has been completed and a detailed project report is being prepared. The type and design of reactors using thorium based fuels alone is not yet finalized. However, the experience gained with the current HWR and the proposed AHWR will be used to take an informed decision on thorium based reactors during the large scale thorium utilization phase of the Indian nuclear power programme.

In summary, the Indian vision for the future development of fuel cycles and reactor systems with HWR technology is strongly linked to the achievement of flexible fuel cycles. The use of recycled plutonium and thorium for conversion to ^{233}U in existing HWRs depends on the specific conditions prevailing at the time. In addition, the development of the AHWR is a major step towards the development of the fuel cycle technologies and advanced reactor system designs which may be utilized in the third phase of the Indian nuclear power programme—based on a synergistic FBR–thorium fuelled reactor system.