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CONCEPTUAL DESIGN OF A LARGE HEAVY WATER REACTOR
FOR U. S. SITING

By
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September 1979

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Volume 3

CONCEPTUAL DESIGN OF A LARGE HEAVY
WATER REACTOR FOR U.S. SITING

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SEPTEMBER, 1979

The conceptual heavy water reactor design presented in this report is based upon the pressurized heavy water Canadian deuterium uranium reactor (CANDU-PHW) and, in particular, upon the standard CANDU 600 reactor (References 5.2-1 and 5.2-2) currently under construction at the Gentilly-II and Point Lepreau station in Quebec and New Brunswick, respectively.

One of the objectives established for development of the conceptual HWR design was to minimize changes to existing CANDU-PHW designs so as to enhance the prospects for early deployment and minimize R&D requirements. As a result, design modifications were restricted to those that fit into the following categories:

- Modifications which would improve fuel utilization or result in significant reduction in capital costs.
- Modifications arising from consideration of current U.S. licensing requirements and U.S. construction practices, for a HWR on a U.S. site.

It should not be inferred that the modifications described in this section in any way indicate inadequacies in the design or performance of the currently operating CANDU's, but were incorporated to provide cost reductions or convenience in licensing and construction for an HWR on a U.S. site.

The major changes incorporated into the conceptual HWR design with respect to the standard CANDU 600 reactor to improve uranium utilization or economic performance were discussed in Section 4.0 and are summarized below.

- A net electrical output of 1260 MW(e), as compared to 600 MW(e) for the standard CANDU design.

- The use of slightly enriched (1.2 w/o) uranium fuel rather than natural uranium.
- Primary system coolant pressure of 2250 psi, rather than 1600 psi, in order to increase net electrical efficiency from 29 to 31.3%. In order to accommodate this increase in primary system pressure, pressure tube wall thickness has been increased to 5.79 mm.
- A D₂O reflector thickness of 33 cm, about half the thickness employed in the standard CANDU design in order to reduce D₂O inventory.
- A two-loop primary system configuration incorporating four steam generators and four primary coolant pumps, in contrast to a three-loop configuration which might be considered for large CANDU plants to more fully utilize components of proven size. The use of the two-loop configuration with larger steam generators allows containment size and equipment costs to be reduced, thereby reducing the plant capital costs.

Other design modifications are described in the following sections.

Containment

The containment was modified from the CANDU design to a U.S. type containment structure incorporating the following features:

- Elimination of the dousing tank located near the top of the containment shell and incorporation of a U.S. PWR type ECCS system to facilitate licensing in the U.S.

- Replacement of epoxy coating in the containment with a carbon steel liner. The carbon steel liner can be provided with test channels to assure maintenance of a vapor leak tight containment at all times. A metal liner permits complete sealing of containment penetrations to the liner, thus providing a homogenous liner material that can be fully inspected for flaws and pin hole leaks.
- Seismically designed for a Safe Shutdown Earthquake of 0.25g and an Operating Basis Earthquake of 0.125 g.

Reactor Service and Fuel Handling Building

This structure was redesigned and resulted in a smaller structure. Some of the functions performed in the CANDU design were transferred to other structures. The following major changes were incorporated:

- Designed as a seismic Category I reinforced concrete structure rather than as a structural steel structure with siding.
- Reduced the size of the spent fuel storage pool to 4/3 rds. core from the conventional Candu spent fuel storage pool holding the equivalent of 10 years operation worth of spent fuel.
- Relocated the control room to the CAD building.
- Loading area for all equipment and supplies entering or leaving the safety related structures of the power plant. This is a controlled access area.
- Contains equipment and components for the engineered safeguards systems.

Control, Auxiliary, and Diesel Building

This structure is new to the CANDU design. Functions from various locations in the CANDU system were combined to provide an arrangement that would meet criteria of the USNRC regulations. Some of the features included in this design are:

- Redundant emergency electric power diesel generators are provided in the lowest level of the CAD building. These diesel generators each provide 100 percent of the electric requirements for safety related systems that must operate during and after an abnormal plant condition.
- Control room provided with life support system, including redundant air intakes, to enable operators to perform their duties under abnormal plant operating conditions. A secondary control room is located in the CAD building to provide a backup means of controlling the reactor systems in the unlikely event that the main control room would become uninhabitable.
- Redundant cable spreading rooms, one above and one below the control room, are provided. Fire detection and fire fighting systems are located in these rooms to assure immediate response in the unlikely event of a fire occurring.
- Major switchgear is located in the CAD building.
- The structure is designed to seismic Category I requirements.
- Security and control of personnel within the reactor plant is controlled from the CAD building.

Penetrations Building

The penetrations building is new to the CANDU system and provides:

- A seismic Category I reinforced concrete structure to protect the main steam and boiler feedwater containment isolation valves.
- Seismic Category I structure to house the moderator cooling heat exchangers and moderator circulating pumps. These were previously located in the containment.
- A seismic Category I structure to contain and support the moderator cooling heat exchanger cooling system. This system is an ammonia absorption system.

Annulus Building

The annulus building is located between the containment and the CAD building. This is a new structure to the CANDU system and provides:

- Seismic Category I reinforced concrete structure to house the cables for the control and powering of various electrical and instrumentation equipment in the power plant complex. Structural steel framing is provided in the annulus to support and separate cable trays and conduits.
- A structure that has walkways to permit controlled personnel access to the containment, penetrations building, and to the reactor service and fuel handling building.

Turbine Building

This is a non-seismic Category building constructed from structural steel framing with siding. Safety related systems and equipment previously contained in this structure was relocated into seismic Category I structures.

Security Systems

The major features of the system incorporated in the HWR design are:

- Guardhouse at entrance to property perimeter.
- Fencing of entire property and of zones within the property boundaries.
- Security building through which all employees and visitors entering and leaving the power plant must pass. This building contains the monitors and alarms of the security systems.
- Covered access walkways leading from the security building to either the CAD building or the turbine building.
- Access from the CAD building to other structures containing safety related equipment can only be made from the operating floor access area in the CAD building.

Engineered Safeguards Systems

The following systems either were incorporated or replaced comparative systems in the CANDU system:

- Containment Spray System

This system replaces the dousing tank containment spray system used in the CANDU system. The spray water is chemically treated to aid in the removal of iodine that may be in the containment atmosphere.

- High-Pressure Safety Injection System (HPSI)

High pressure safety injection pumps (HPSIP) and safety injection tanks have been incorporated to provide core recovery for a full spectrum of reactor coolant system break sizes. To insure that all reactor coolant tubes will be cooled during shutdown cooling and refilled following an accident, eight discharge lines from each safeguard pump were included.

- Shutdown Cooling System (SCS)

The shutdown cooling system (SCS) employed in the conceptual HWR design is similar to systems employed in PWRs. In particular, the shutdown cooling heat exchangers are designed for operation at primary system pressures below 650 psia, in contrast to operating CANDU plants where a full system pressure SCS is provided.

The shutdown cooling system (SCS) does not interface with the containment spray system (CSS). To meet the CANDU rapid cooldown capability, the capacity of the low pressure safety injection pumps and shutdown cooling heat exchangers (SDCHX) have been significantly upgraded with respect to the PWR. Both the safety injection system and shutdown cooling system have two separate and redundant trains.

- Containment Isolation System

Every pipe or duct penetrating the containment is provided with valves to isolate the containment in the unlikely event of a pipe rupture. The isolation valve operators are designed to fail in a "safe" position should their power source fail.

- Emergency Water Storage

The condensate storage tank and the demineralized water storage tank are designed to seismic Category I requirements. They are reinforced concrete with steel liners and are buried. The tanks are provided with reinforced concrete covers designed for missile protection.

Ultimate Heat Sink

Redundant cooling towers together with necessary pumps, piping, valves and heat exchangers are provided. The towers, structures, equipment, and components are designed to seismic Category I requirements to assure that they will be available in the unlikely event of an abnormal plant operating condition. The electrical systems will be supplied from redundant sources.

Holding Pond

This is provided as a means of collecting and monitoring all liquid effluents from the plant prior to discharging to the environment.

Primary Component Cooling Water System (PCCW)

This is a new system relative to the CANDU system. It is a redundant safety related system designed to seismic Category I requirements. This is a closed system that operates under normal or abnormal plant operating conditions. The system is designed

to provide cooling water to those heat exchangers of systems required to perform a safety function or for those systems containing tritiated water.

The PCCW system heat exchangers are connected to the circulating water cooling towers during normal operations and to the ultimate heat sink cooling towers during abnormal plant operating conditions.

Combustible Gas Control System

A redundant combustible gas control system is provided to prevent the accumulation of hydrogen from reaching combustible concentrations in the containment following a loss of coolant accident.

Moderator Cooling System

An ammonia absorption system was incorporated into the design of the HWR plant. Based on an engineering evaluation of the problems associated with maintaining the moderator pressure and temperature to an acceptable level without pressurizing the calandria, a refrigeration system was selected for cooling the moderator fluid.

Since the HWR is using cooling towers and an intermediate closed circuit cooling loop (PCCW), the economics of cooling the moderator fluid with conventional water-cooled heat exchangers, as used in the CANDU system, became apparent. The cooling system has a duty of 748.2×10^6 Btu/hr and cools 27,100 gpm of D_2O from 160°F to 110°F. A conventional mechanical compression system would require a large electrical consumption for the compressor drive motors, as well as major maintenance.

A final engineered selection was made based on the ammonia absorption system. This reduced the electrical consumption from in excess of 60,000 HP for the compressors to two pumps each requiring a 250 HP motor. The driving force for the refrigeration effect is furnished by steam from the auxiliary steam boiler.

The problem of ammonia entering the moderator is precluded by the use of double tube moderator heat exchangers. The moderator is contained in the tube side and the ammonia in the shell side of the exchanger.

Reactor Coolant System (RCS)

- Four channel actuation is provided for protection functions. Two trips of the four are required for actuation of a protective function vs. two of three for CANDU. The advantage of two of four channel actuation is increased plant availability and reliability. Plant operation can continue without an increase in spurious trip signals with one channel out of service.
- Stainless steel wetted surfaces have been used in place of carbon steel wetted surfaces of CANDU. The advantage of stainless steel is its corrosion and erosion resistance. The feeder tubes from the headers to the pressure channels will be stainless steel and the major pipe sections, headers, pump casings and steam generator primary head and nozzles will be clad on the interior with austenitic stainless steel. Any connecting piping will be austenitic stainless steel.
- In accordance with DGC-4, a complete system of supports for the RCS have been incorporated. The supports are included for the steam generators, pumps, major pipe sections, and headers. The supports will be designed to withstand the effects of loss-of-coolant incident with a simultaneous safe shutdown earthquake.

Reactor Coolant Pressure and Inventory Control System

The following systems either were incorporated or replaced comparative systems in the CANDU system:

- The inventory control, or charging and letdown, was changed to include positive displacement pumps as presently used on PWR light water NSSS. The advantage of this type of pump is constant flow rate at varying discharge pressures.
- Overpressure protection was modified to meet the requirements of Regulatory Guide 1.67. This includes power operated relief valves (PORV's). These PORV's are located upstream (RCS side) valves of the surge line isolation valves so that they cannot be isolated from the RCS. The actuation signal will be provided by protective channels on the reactor inlet header.
- The pressurizer will be protected from overpressurization when isolated by the addition of a safety valve on the steam space. This valve will be sized to protect the pressurizer and surge line from overpressurization due to inadvertent heater actuation when the pressurizer is isolated.
- So that the surge line isolation valves meet single failure criteria, redundant isolation valves were included in each line to the RCS loops. These valves will be actuated in such a manner that no single failure will prevent at least one valve from closing in the affected loop following a loss-of-coolant incident.
- The system has been modified to include a pressurizer spray for normal operations as well as for cooldown. The spray will inject subcooled D_2O into the steam space at a rate sufficient to prevent a high pressure reactor trip during any normal plant evolution.

The modifications indicated in the preceding discussion represent the significant changes that have been incorporated in the conceptual HWR design relative to the CANDU 600 PHW. Modifications that resulted primarily from increased size of the plant (1260 MWe vs. 600 MWe or the arrangement of the primary coolant system) have not been included since these modifications do not represent deviations from the basic CANDU concept.

References

- 5.2-1 "CANDU 600 Station Design", Atomic Energy of Canada Limited, May 1976
- 5.2-2 "Point Lepreau", Nuclear Engineering International, June 1977

5.3 Capital Cost Estimate

5.3.1 Introduction

This section summarizes the total base construction cost estimate for the conceptual heavy water reactor (HWR) plant described in Section 5.1, and reflects the design modifications as indicated in Section 5.2. A detailed tabulation of component costs can be found in Appendix 7.3.

The cost estimate reported herein are for an established unit (nth-of-a-kind), and hence do not include contingency costs for the equipment, material and labor components. Escalation and interest during construction are also excluded. Other items not included are listed in Table 5.3-1. This estimate is considered baseline for "Middletown USA", and would require adjustments for regional variations in material and labor rates, different construction schedule durations, and for escalation and interest rates incurred during construction.

The bases for determining the costs are as follows:

- ° The point of departure for developing capital costs were the previous USDOE studies: C00-2477-4 "Licensing Assessment of the CANDU Pressurized Heavy Water Reactor", and; C00-2477-13 "Commercial Electric Power Cost Studies, Capital Cost, Pressurized Heavy Water Reactor Plant". Both reports are dated June 1977. Capital costs developed in these studies were modified to reflect the design changes developed in Section 5.1.
- ° The Capital Cost Estimate is for an nth-of-a-kind (NOAK) reference HWR plant (1260 Mwe) at a Middletown, USA site.

- ° A detailed engineering design was not prepared for this reference plant. Where possible, costs were estimated explicitly for HWR plant components. Where little or no information was available, design data from a similar size Pressurized Water Reactor (PWR) plant was used as a basis for estimating HWR capital cost.
- ° Cost data is based on prices effective as of January 1, 1978.
- ° The licensing and design criteria in effect as of January 1, 1978 was utilized.
- ° The main heat rejection system utilizes mechanical draft wet cooling towers. The nuclear ultimate heat sink is redundant mechanical draft wet cooling towers.
- ° Escalation and interest during construction are not included in the cost estimate.
- ° The plant has an onsite spent fuel storage capacity of 4/3 cores as opposed to 10 years capacity for a Canadian sited CANDU reactor.
- ° The plant design life is thirty years.
- ° A single unit on a new site with sufficient land area to accommodate a second unit.

5.3.2 Economic Evaluation

The estimated total base construction cost for the 1260 MWe conceptual HWR dry (exclusive of heavy water) plant is \$740,124,000 or \$587/kWe based on U.S. dollars and prices in effect as of January 1, 1978. In this report the general practice followed is to include the cost for the initial charge of heavy water with the fuel cycle cost and present the capital costs for a dry plant. The heavy water can be eliminated from the fuel cycle cost and included in the capital costs by adding, 1,802,900 lbs of D_2O at \$97 per pound to the dry value, which gives a wet plant cost of \$915,006,000 or \$726/kWe. Either way of treating the heavy water cost will yield the same total power costs.

Table 5.3-2 presents a summary cost estimate in accordance with the expanded AEC Code of Accounts (U.S. AEC Report NUS-531). The cost data are separated into direct and indirect accounts, where the directs cover the capital cost of land, structures and plant equipment, and indirects cover construction and engineering services. A listing of the contents of these accounts is given in Table 5.3-3. Appendix 7.3 contains a detailed listing of costs by accounts along with a direct craft labor summary.

One of the main objectives of this study was to develop a conceptual HWR design which would be more economically competitive with U.S. Light Water Reactors. Previous evaluations have indicated that the capital costs of a natural uranium fueled HWR plant, modified toward U.S. licensing criteria and construction practices, are higher than those of the PWR even though both systems share many common design features (containment, steam generators, etc.). A study reported in Reference 5.3-1 established a capital cost of \$587/kWe for a 1100 MWe U.S. sited HWR, based on a scaled up CANDU 600 design with modifications to allow for consideration of U.S. licensing requirements. This capital cost is about 18%

higher than the \$499/kWe estimated for a 1139 Mwe Pressurized Water Reactor in Reference 5.3-2.

Basically the capital cost difference between the HWR and PWR can be attributed to the following: the reactor and primary system piping; the necessity of incorporating additional components such as the dual refueling machines, moderator heat rejection system, and D₂O upgrader; and the lower net efficiency of the HWR which results in a lower electrical output for a specific thermal power rating.

HWR efficiencies are lower than those of the PWR for two reasons: because of the unproductive loss of energy to the moderator, and because of the lower primary and secondary operating pressures and temperatures. While little can be done to reduce the loss of efficiency which results from the former, it appears feasible to increase coolant conditions to those comparable with the PWR, so that the thermal energy deposited in the coolant can be converted to electrical energy with an efficiency more comparable to that of the PWR. Consequently, the conceptual HWR plant has been designed to operate at a somewhat higher primary system pressure than do present Candu reactors (2250 psi vs. 1600 psi for the Candu 600) in order to allow primary system coolant temperatures and secondary pressures to be increased. As a result, station electrical efficiency is increased from the 29.3% of the standard Candu 600 design to 31.3%. This increase in thermal efficiency is significant since it allows approximately 6.8% more electrical energy to be generated per unit of thermal energy produced, and affords a similar cost (\$/kWe) reduction in many HWR components.

In addition to the increased pressure and temperature, other modifications have been incorporated into the conceptual design for the purpose of reducing capital cost. Where these modifications have had an apparent impact on the capital cost is shown on Table

5.3-4. This table presents a comparison between the costs for the three loop natural uranium fueled, 1160 MWe HWR design based on the design described in Reference 5.3-1, adjusted to January, 1978 dollars and to a comparable thermal output, and the 1260 MWe conceptual design employing two primary coolant loops and slightly enriched uranium fuel. Although a precise intercomparison is difficult because of the numerous design difference between the plants (some of which decrease plant cost but others of which increase plant cost) and the charges in material costs which have occurred since the earlier capital cost study was performed, a number of observations can be made.

As indicated by this comparison, all accounts except 20 and 92 (land and home office services) show some differences in costs, with an overall reduction of the order of eleven million dollars. This reduction was realized even though modifications were incorporated to improve efficiency and thus increase the net electrical power output. A major portion of this cost reduction results from the consolidation of structures, elimination of dousing tank in the containment, smaller containment, and smaller spent fuel storage capacity, and is reflected in the decreased costs in accounts 21, 91, and 93. Other accounts show small variations which would be expected for plants of similar size. However, even though only a small difference is indicated, in plant equipment (account 22), cost reductions were achieved in certain areas which were compensated for by increases in others. For example, the utilization of a two loop primary coolant system reduced the number and cost of required components and piping, as compared with a three loop system. The incorporation of an ammonia system for moderator cooling in lieu of a water system also reduced costs. Compensating increases resulted from inclusion of an emergency core cooling system similar to that currently employed by U.S., PWR's and the increased cost of zircaloy pressure and calandria tubes.

Information from component vendors have indicated that recent increases in Zirconium prices have been far in excess of any traditional escalation rates which might be assumed. It is factors like this and others such as scope and sources of cost estimates that make cost comparisons, at best, difficult to access. In addition many of the improvements incorporated into the conceptual design could be applied to the natural uranium plant as well. Thus the conclusion reached from this comparison is that an improvement in the electrical energy generated per unit of thermal energy produced can be realized without a significant impact on the total base cost of the plant.

A second capital cost comparison is presented in Table 5.3-5, in which the HWR conceptual design is compared to a 1256 MWe PWR, both at the hypothetical Middletown, U.S. site. The data for the PWR is based on information contained in a 1978 update to USNRC Report NUREG-0241/USDOE Report COO-2477-5. The reference PWR plant in the study had a core power of 3425 MWt. In order to facilitate the comparison, the PWR was scaled-up to 3800 MWt (1256 MWe). The resulting capital cost, \$543/kWe, is about 8% lower than that of the conceptual HWR.

The comparison of capital costs of the 1260 MWe HWR conceptual design with other plant concepts, such as the U.S. PWR has additional uncertainties, which make comparisons even more difficult than that previously discussed. Examples of these uncertainties are as follows:

- ° Although an effort was made to incorporate U.S. licensing requirements into the conceptual HWR design, further modifications may be necessary as a result of licensing review. The extent and cost impact of the modifications are unknown. In contrast the PWR licensing requirements are well established.

- ° The PWR industry is well established in the U.S. thus more precise construction and fabrication costing information exists. In contrast HWR costs for a plant located in the U.S. are subject to greater uncertainties.
- ° The nuclear industry is very competitive in the U.S., whereas, the costs presented for the HWR were not based on competitive bids.
- ° The design on which the HWR was costed is conceptual; design details could influence component and construction costs.

Due to the above, any conclusions drawn from this comparison must be kept general. A review of the comparison indicates that the capital cost for the conceptual HWR is generally higher than that for the PWR with differences primary due to the reactor plant equipment (account 22), the main heat rejection system (account 26) and indirect costs, with variations in other accounts being small. The difference in the reactor plant result from those items mentioned previously, namely the reactor and primary system piping, dual online refueling machines, moderator heat rejection system and the D₂O upgrader. The main heat rejection system differences results from the larger required capacity due to the heat load from the moderator. These items impact the engineering and construction services and are reflected in the indirect costs.

Based on the information developed for this study and as reported herein the following general conclusions with respect to capital cost have been reached:

- ° The baseline capital cost (dry, excluding interest and escalation during construction, owner's cost) of the conceptual HWR design is \$587/kWe. The cost of the initial D₂O inventory (at \$97/lb) adds \$139/kWe to this baseline cost.

- ° The capital cost of the conceptual HWR plant (dry) is about 8% higher than that of a U.S. PWR, with the most significant difference being attributed to the cost of the reactor plant equipment.
- ° The improvement in electrical energy production resulting from higher plant efficiency will result in the HWR being more economically competitive with U.S. Light water reactors, with a breakeven occurring at a lower uranium price than would occur at the standard CANDU efficiency (See Section 5.4).
- ° The modifications incorporated into the conceptual HWR plant resulted in about a 10% cost reduction as compared to previous DOE cost estimates for a U.S. sited HWR.

TABLE 5.3-1

HWR CAPITAL COST ESTIMATE EXCLUSIONS

1. Main Transformer, Switchyard and Transmission Facility Costs
2. Owner's Costs, Including Consultants, Site Selection, etc.
3. Waste Disposal Costs
4. Fees and Permits - Federal, State, Local
5. State and Local Taxes
6. Spare Parts
7. Nuclear Liability and Other Insurance
8. Initial Fuel Loading
9. Interest During Construction
10. Escalation
11. Contingency
12. Heavy Water Inventory

TABLE 5.3-2
COST ESTIMATE SUMMARY FOR
1260 MW(e) CONCEPTUAL HEAVY WATER REACTOR PLANT

Site: Middletown, USA
Cost base: January 1978 US dollars

Direct Costs

20 Land and land rights				2,240,000
Physical plant	<u>Equipment</u>	<u>Materials</u>	<u>Labor</u>	<u>Total</u>
21 Structures and site facilities	6,831,000	52,317,000	61,350,000	120,498,000
22 Reactor/boiler plant equipment	154,081,000	10,719,000	30,488,000	195,288,000
23 Turbine plant equipment	96,060,000	5,719,000	25,699,000	127,478,000
24 Electric plant equipment	15,896,000	9,169,000	20,191,000	45,256,000
25 Miscellaneous plant equipment	9,444,000	805,000	5,419,000	15,668,000
26 Main Heat Rejection System	<u>22,916,000</u>	<u>1,417,000</u>	<u>4,958,000</u>	<u>29,291,000</u>
Sub-total (physical plant)	305,239,000	82,385,000	148,105,000	533,479,000

Indirect Costs

91 Construction services	75,630,000
92 Home Office services	97,800,000
93 Field Office Services	<u>30,975,000</u>
Sub-total (indirect costs)	204,405,000
Total Plant Cost (see note 1)	740,124,000

Notes:

1. Total plant cost excludes cost of heavy water inventory, initial fuel charge, and other items as indicated in Table 5.3-1.
2. For contents of accounts, see list in Table 5.3-3.

TABLE 5.3-3
CONTENTS OF ACCOUNTS

<u>Code-of-Accounts</u>	<u>Title</u>
20.	<u>Land & Land Rights</u>
21.	<u>Structures & Improvements</u>
211.	° Yardwork
212.	° Reactor Containment
213.	° Turbine Room & Heater Bay
214.	° Security Building
215.	° Reactor Service & Fuel Handling Building
216.	° D ₂ O Upgrading Structure
218A.	° Control Room & Diesel-Generator Building
218B.	° Administration & Warehouse
218D.	° Fire Pump House
218J.	° Penetrations Building
218K.	° Pipe Tunnels
218S.	° Holding Pond
218T.	° Ultimate Heat Sink Structure
218V.	° Control Room Emergency Air Intake Structure
22.	<u>Reactor Plant Equipment</u>
220A.	° Nuclear Steam Supply
221.	° Reactor Equipment
	° Reactor Support
	° Calandria & End Structure
	° Fuel Channels
	° Reactor Control Devices
222A.	° Main Heat Transfer Transport System
	° Reactor Core Coolant System
	° Pumps
	° Piping Systems
	° Steam Generators
	° Pressurizing System

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

- ° D₂O Feed & Bleed System
 - ° Pumps
 - ° Piping
 - ° Coolant Collection & Storage
 - ° Pumps
 - ° Tanks
 - ° Heat Exchangers
 - ° Piping System
 - ° Instrumentation & Controls
 - ° Purification System
 - ° Heat Exchangers
 - ° Purification Equipment
 - ° Piping System
 - ° Deuteration & Dedeuteration
 - ° Pumps
 - ° Tanks
 - ° Piping Systems
 - ° N₂/H₂ Addition System
 - ° D₂O Sampling Systems
- 222B.
 - ° Moderator Circuit
 - ° Moderator Cooling System
 - ° Heat Exchangers
 - ° Pumps
 - ° Calandria Head Tank
 - ° Piping Systems
 - ° Moderator Purification
 - ° Heat Exchangers
 - ° Purification Equipment
 - ° Piping Systems

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

- Deuteration & Dedeuteration
 - Pumps
 - Tanks
 - Purification Equipment
 - Piping Systems
 - Cover Gas System
 - Compressors
 - Heat Exchangers
 - Recombiners
 - Piping Systems
 - Moderator Collection System
 - Pumps
 - Tanks
 - Piping Systems
 - Instrumentation & Controls
 - D₂O Sampling Systems
 - Tanks
 - Piping Systems
 - Moderator Liquid Poison System
 - Pumps
 - Tanks
 - Piping
- 223. ◦ Safeguards System
 - Shutdown Cooling System
 - Pumps
 - Heat Exchangers
 - Piping Systems

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

- Moderator Poison Injection System
 - Tanks
 - Injection Nozzles
 - Piping Systems
- Containment Spray System
 - Tanks
 - Piping Systems
- Combustible Gas Control System
 - Piping Systems
 - Recombiner
 - Blowers
 - Heat Exchangers
- Emergency Core Cooling System
 - Pumps
 - Heat Exchangers
- 224. ◦ Radwaste Processing
 - Liquid Radwaste (D_2O)
 - Pumps
 - Tanks
 - Purification Equipment
 - Piping Systems
 - Miscellaneous Waste
 - Pumps
 - Tanks
 - Heat Exchangers
 - Purification Equipment
 - Piping Systems

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

- Detergent Waste
 - Pumps
 - Tanks
 - Purification Equipment
 - Piping Systems
- Chemical Waste
 - Pumps
 - Tanks
 - Piping Systems
- Steam Generator Blowdown
 - Pumps
 - Heat Exchangers
 - Tanks
 - Purification Equipment
 - Piping Systems
- Regenerative Chemical Waste
 - Pumps
 - Tanks
 - Purification Equipment
 - Piping Systems
- Miscellaneous Liquid Radwaste
- Gaseous Waste
 - Compressors
 - Tanks
 - Purification Equipment
 - Piping Systems
 - Volume Reduction System

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

- 225.
 - ° Fuel Handling System
 - ° Fuel Handling Tools & Equipment
 - ° Cranes & Hoists
 - ° Fuel Handling Equipment and Fueling Machine Heads
 - ° Fuel Transfer Systems
 - ° Service Platforms
 - ° Fuel Storage Equipment
 - ° New Fuel Storage Racks
 - ° Spent Fuel Storage Racks
 - ° Spent Fuel Pool Cleaning & Purification Equipment

- 226.
 - ° Other Reactor Plant Equipment
 - ° Inert Gas System
 - ° H₂/N₂ Gas Supply System
 - ° D₂O Upgrading System
 - ° Annulus Gas System
 - ° Moderator Refrigeration System
 - ° Pumps
 - ° Heat Exchangers
 - ° Tanks
 - ° Piping Systems
 - ° Auxiliary Cooling Systems
 - ° Nuclear Service Water System
 - ° Pumps
 - ° Heat Exchangers
 - ° Piping Systems
 - ° Primary Component Cooling Water System
 - ° Pumps
 - ° Heat Exchangers
 - ° Tanks
 - ° Piping Systems

TABLE 5.3-3 (Cont.)

<u>Code-of- Accounts</u>	<u>Title</u>
	<ul style="list-style-type: none"> ° End Shield Cooling System <ul style="list-style-type: none"> ° Pumps ° Heat Exchangers ° Tanks ° Purification Equipment ° Piping Systems ° Gland Seal Cooling System ° N/S Chilled Water System <ul style="list-style-type: none"> ° Pumps ° Piping Systems ° Tanks ° Purification Equipment ° Chiller Units ° Maintenance Equipment ° Laundry Equipment ° Sampling Equipment
227.	<ul style="list-style-type: none"> ° <u>Reactor Instrumentation & Controls</u> <ul style="list-style-type: none"> ° Benchboard, Panels & Racks ° Process Computer ° Monitoring Systems ° Plant Control Systems ° Tubing & Fittings
228.	<ul style="list-style-type: none"> ° <u>Reactor Plant Miscellaneous/Items</u> <ul style="list-style-type: none"> ° Miscellaneous Suspense Items ° Standard NSSS Valve Package ° Reactor Plant Insulation

TABLE 5.3-3 (Cont.)

<u>Code-of- Accounts</u>	<u>Title</u>
23.	<ul style="list-style-type: none"> ° <u>Turbine Plant Equipment</u> <ul style="list-style-type: none"> ° Turbine Generator & Accessories ° Foundations ° Lubricating Oil System <ul style="list-style-type: none"> ° Tanks ° Piping Systems ° Gas Systems <ul style="list-style-type: none"> ° Hydrogen Storage System ° Carbon Dioxide System ° Moisture Sep/Reheater Drain System <ul style="list-style-type: none"> ° Tanks ° Piping Systems
233.	<ul style="list-style-type: none"> ° <u>Condensing Systems</u> <ul style="list-style-type: none"> ° Condenser Equipment <ul style="list-style-type: none"> ° Condensers ° Condensate System <ul style="list-style-type: none"> ° Pumps ° Tanks ° Piping Systems ° Foundations ° Gas Removal System <ul style="list-style-type: none"> ° Pumps ° Piping Systems ° Foundations ° Turbine Bypass System <ul style="list-style-type: none"> ° Piping Systems ° Condensate Polishing <ul style="list-style-type: none"> ° Pumps & Blowers ° Tanks ° Purification Equipment

TABLE 5.3-3 (Cont.)

<u>Code-of-Accounts</u>	<u>Title</u>
234.	<ul style="list-style-type: none"> ° <u>Feedwater Heating System</u> <ul style="list-style-type: none"> ° Feedwater Heaters <ul style="list-style-type: none"> ° Heat Exchangers ° Feedwater Systems <ul style="list-style-type: none"> ° Pumps ° Piping Systems ° Foundations ° Extraction Steam System <ul style="list-style-type: none"> ° Piping Systems ° Feedwater Heater Vent & Drain System <ul style="list-style-type: none"> ° Pumps ° Tanks ° Piping Systems
235.	<ul style="list-style-type: none"> ° <u>Other Turbine Plant Equipment</u> <ul style="list-style-type: none"> ° Main Steam System <ul style="list-style-type: none"> ° Piping Systems ° Turbine Auxiliaries <ul style="list-style-type: none"> ° Piping Systems ° Turbine Building Closed Cooling Water System <ul style="list-style-type: none"> ° Pumps ° Heat Exchangers ° Tanks ° Piping Systems ° Demineralized Water Make-Up System <ul style="list-style-type: none"> ° Tanks ° Piping Systems ° Pumps ° Filters ° Foundations

TABLE 5.3-3 (Cont.)

<u>Code-of-</u> <u>Accounts</u>	<u>Title</u>
	<ul style="list-style-type: none"> ° Chemical Treatment System <ul style="list-style-type: none"> ° Pumps ° Tanks ° Piping Systems ° Neutralization System <ul style="list-style-type: none"> ° Pumps ° Tanks ° Piping Systems
236.	<ul style="list-style-type: none"> ° <u>Instrumentation and Control</u> <ul style="list-style-type: none"> ° Process IC Equipment ° I&C Tubing
237.	<ul style="list-style-type: none"> ° <u>Turbine Plant Miscellaneous Items</u> <ul style="list-style-type: none"> ° Miscellaneous Suspense Items ° Turbine Plant Insulation
24.	<u>Electric Plant Equipment</u>
241.	° Switchgear
242.	° Station Service Equipment
243.	° Switchboards
244.	° Protective Equipment
245.	° Wiring
246.	° Power & Control Wiring
25.	<u>Miscellaneous Plant Equipment</u>
	<ul style="list-style-type: none"> ° Cranes & Hoists <ul style="list-style-type: none"> ° Turbine Building Crane ° Reactor Containment Crane ° Reactor Service Bldg. Crane

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

- ° Intake Structure Crane
- ° Circ. Water Pump House Crane
- ° Diesel Building Cranes
- ° Compressed Air System
 - ° Compressors
 - ° Tanks
 - ° Piping Systems
- ° Containment Bldg. Instr. Air. System
 - ° Compressors
 - ° Tanks
 - ° Driers
 - ° Piping Systems
- ° Service Water System
 - ° Pumps
 - ° Piping Systems
- ° Yard Fire Protection System
 - ° Diesel & Motor Driven Pumps
 - ° Pumps
 - ° Heat Exchangers
 - ° Tanks
 - ° Piping Systems
 - ° Hose, Hydrants, Sprinklers, Deluge Systems, Etc.
- ° Potable Water System
 - ° Piping Systems
- ° Auxiliary Steam System
 - ° Boilers
 - ° Piping Systems
- ° Feedwater System
 - ° Pumps

TABLE 5.3-3 (Cont.)

Code-of-
AccountsTitle

- ° Auxiliary Deaerator & Makeup System
 - ° Pumps
 - ° Tanks
 - ° Piping Systems
- ° Auxiliary Chemical Feed System
 - ° Pumps
 - ° Tanks
 - ° Piping Systems
- ° Auxiliary Steam & Condensate Return
 - ° Pumps
 - ° Tanks
 - ° Piping Systems
- ° Auxiliary Boiler Stacks & Ducts
- ° Auxiliary Boiler Blowdown System
 - ° Tanks
 - ° Piping Systems
- ° Auxiliary Steam System I&C
- ° Plant Fuel Oil System
 - ° Tanks
 - ° Foundations

253.

- ° Communications Equipment
 - ° Local Communications System
 - ° Signal Systems

TABLE 5.3-3 (Cont.)

Code-of-
Accounts

Title

254.

- ° Furnishings and Fixtures
 - ° Safety Equipment
 - ° Chemical Lab & Instrument Shop
 - ° Office Equipment & Furnishings
 - ° Change Room Equipment
 - ° Environmental Monitoring Equipment
 - ° Dining Facilities

TABLE 5.3-4
COMPARISON OF COSTS
1100 MWe HWR (NATURAL U-FUEL)
VERSUS 1260 MWe HWR (1.2% LEU FUEL)
NUCLEAR POWER PLANTS

<u>Acct. Number</u>	<u>Description</u>	<u>Natural U 1160 MWe HWR</u>	<u>1.2% LEU 1260 MWe HWR</u>
20.	Land & Land Rights	\$ 2,240,000	\$ 2,240,000
21.	Structures & Site Facilities	128,097,000	120,498,000
22.	Reactor/Boiler Plant Equipment	194,129,000	195,288,000
23.	Turbine Plant Equipment	126,608,000	127,478,000
24.	Electric Plant Equipment	46,088,000	45,256,000
25.	Miscellaneous Plant Equipment	16,345,000	15,668,000
26.	Main Heat Rejection System	<u>26,640,000</u>	<u>29,291,000</u>
	TOTAL DIRECT COST	\$540,147,000	\$535,719,000
91.	Construction Services	80,780,000	75,630,000
92.	Home Office Services	97,800,000	97,800,000
93.	Field Office Services	<u>32,335,000</u>	<u>30,975,000</u>
	TOTAL INDIRECT COST	\$210,915,000	\$204,405,000
	TOTAL BASE COST	\$751,062,000	\$740,124,000

TABLE 5.3-5
COMPARISON OF COSTS
1254 MWe PRESSURIZED WATER REACTOR
VERSUS 1260 MWe CONCEPTUAL HWR

<u>Acct. Number</u>	<u>Description</u>	<u>1254 MWe PWR Plant</u>	<u>1260 MWe HWR Plant</u>
20	Land & Land Rights	\$ 2,200,000	\$ 2,200,000
21	Structures & Site Facilities	121,300,000	120,500,000
22	Reactor/Boiler Plant Equipment	153,100,000	195,300,000
23	Turbine Plant Equipment	128,000,000	127,500,000
24	Electric Plant Equipment	44,700,000	45,300,000
25	Miscellaneous Plant Equipment	15,300,000	15,700,000
26	Main Heat Rejection System	<u>21,900,000</u>	<u>29,300,000</u>
	TOTAL DIRECT COSTS	\$486,500,000	\$535,800,000
	TOTAL INDIRECT COSTS	<u>\$195,620,000</u>	<u>\$204,400,000</u>
	TOTAL COSTS	\$682,120,000	\$740,200,000

References

- 5.3-1 C00-2477-13, "Capital Cost Pressurized Heavy Water Reactor Plant",
June 1977.
- 5.3-2 C00-2477-5, "Capital Cost Pressurized Water Reactor Plant",
May 1977.

5.4 Fuel Management Characterization

5.4.1 Introduction

The purpose of this section is to characterize the fuel management of the conceptual HWR design. Discussed are U_3O_8 and/or other fissile material requirements, separative work requirements, and fuel cycle and net power cost economic performance. The slightly enriched once-through uranium fuel cycle is addressed in Section 5.4.2, while Section 5.4.3 discusses the uranium fuel cycle with plutonium recycle. Lastly, Section 5.4.4 provides similar information for the conceptual HWR operating on the thorium fuel cycle. Addressed are the denatured thorium fuel cycle with uranium recycle and U-235 makeup, the denatured thorium fuel cycle with uranium recycle and U-233 makeup, and the plutonium-enriched thorium fuel cycle. More detailed reactor fuel charge and discharge inventory data can be found in Appendix 7.2.

5.4.2 Slightly Enriched Uranium Once-Through Fuel Cycle

5.4.2.1 Description of Reference Fuel Cycle

The fuel enrichment evaluation described in Section 4.1 established the effect of initial enrichment on uranium resource and fuel cycle cost performance. This evaluation concluded that an initial enrichment of 1.2 w/o was near optimal from the standpoint of both uranium utilization and fuel cycle costs, and that limiting the initial enrichment to 1.2 w/o minimizes fuel performance and core power peaking problems. Consequently, an initial enrichment of 1.2 w/o was selected as the reference value for this fuel management characterization.

Table 5.4.2-1 provides a summary of the general reactor performance specifications which are pertinent to the fuel cycle characterization (a more detailed description of the conceptual HWR can be found in Section 5.1). Of particular note is the assumed capacity factor of 75%, specified to facilitate intercomparison of resource utilization with other DOE evaluations, and the use of on-line refueling. Fuel assembly specifications are summarized in Table 5.4.2-2. Since the fuel design assessment (cf. Section 4.8) indicated that the discharge exposures anticipated for 1.2 w/o initial fueling could be accommodated by existing Candu fuel design, these specifications are unaltered from those normally employed in Candu reactors.

As discussed in Section 4.1, the discharge exposure predicted by C-E calculations for a 1.2 w/o initial enrichment is 21,700 MWd/t, while Canadian calculations predict a burnup about 10% lower (Canadian predictions, normalized to the operating conditions of the conceptual HWR, are 19,750 MWd/t). Thus, there is approximately a 2000 MWd/t uncertainty in the predictions of achievable discharge burnup. It is impossible at this time to conclude definitively

which of these estimates are more nearly correct; more detailed calculations might help to resolve this discrepancy, but definitive conclusions will probably have to await operating reactor performance data. For the purpose of the fuel management characterization, we have elected to employ the lower of the discharge burnups, i.e., to employ a discharge burnup of 19,750 MWd/t, for the following reasons:

1. Canadian calculational techniques have been subject to much more thorough benchmarking than have the C-E calculational tools. Although this benchmarking has been limited largely to natural uranium-fueled systems, one might nevertheless expect the Canadian calculations to be more precise.
2. The procedure used to define achievable discharge exposure (described in Section 4.1) is only approximate. Although multidimensional calculations would be required to completely eliminate the ambiguities in the definition of end-of-cycle, the Canadian procedure for defining the achievable burnup is somewhat more complex and accurate than that employed in this analysis. The C-E procedure is thought to overestimate the achievable discharge exposure by about 500 MWd/t.
3. Although the design modifications required to evolve the conceptual HWR design from the standard Candu 600 plant have been included in the fuel cycle analysis, it is recognized that many design uncertainties remain, and that further design changes, should they be necessary, will almost invariably lead to some reduction in achievable discharge exposure. For example, considerable uncertainty exists in properties of the zirconium pressure tube at the elevated temperatures and pressures of the conceptual design, and some increase in pressure tube thickness may be necessary, depending upon the results of experimental programs to establish the characteristics

of the pressure tubes at these altered operating conditions. Irradiation demonstration programs may also identify the need for modifying fuel rod design to accommodate higher discharge exposure, perhaps necessitating some increase in clad thickness. Additional absorber rod strength could also be required in order to balance the power distribution in the larger core size of the conceptual HWR. Changes such as these will reduce the achievable burnup somewhat from that predicted by simple lattice cell calculations.

In view of the considerations mentioned above, a discharge exposure of 19,750 MWd/t has been assumed as the reference discharge exposure for the conceptual HWR. The use of the lower discharge exposure should provide a degree of conservatism in the fuel management characterization to accommodate uncertainties in the fuel cycle analysis and decreases in achievable burnup which may result from design changes.

Fuel Cycle facilities required for the slightly-enriched once-through fueling mode are illustrated in Figure 5.4.2-1; also shown in the figure are the annual fuel throughputs for the equilibrium cycle of operation. The fuel cycle facilities required are essentially identical to those needed for light-water reactors, and consist of an isotopic enrichment plant, a facility to convert UF_6 to UO_2 , a fabrication plant, and provisions for either the long-term storage or permanent disposal of spent fuel. These facilities differ from those required for natural uranium fueling primarily in the addition of the isotopic enrichment and conversion facilities. Existing (natural uranium) fabrication facilities could probably be utilized to fabricate the slightly-enriched fuel, but some modification to these existing lines may be required in order to meet the accountability requirements associated with the use of slightly-enriched fuel and to ensure that adequate margins to criticality are maintained throughout the fabrication

line. The spent fuel storage or permanent disposal facility, although identical in concept to that required for natural uranium fueling, would need to consider as a design basis the higher decay heat production due to the higher discharge exposure achieved with slightly-enriched fueling.

Fuel management information for the slightly-enriched once-through cycle is summarized in Table 5.4.2-3. The reference fuel cycle employs 1.2 w/o enriched fuel for all reloads, but natural uranium fuel is utilized for the initial core loading. Natural uranium, rather than 1.2 w/o enriched fuel, is utilized to fuel the initial core to avoid the necessity of excessive reactivity holddown until an equilibrium burnup distribution is attained; the use of natural uranium fuel also improves fuel cycle cost, by reducing head-end inventory requirements. Normally (once an equilibrium cycle is attained) there is little excess reactivity present in the HWR because of on-line refueling. With on-line refueling, fresh fuel is being charged and spent fuel discharged in an essentially continuous fashion so as to maintain a constant value for the core average exposure, and hence to maintain essentially zero excess reactivity. In the case of the initial core (which initially has an average burnup of zero), soluble boron poison control must be used to control excess reactivity until the equilibrium core average exposure is attained, in a fashion similar to the reactivity depletion control employed in PWRs. If 1.2 w/o fuel were used in the initial core, the reactor would have to be operated in this poison control mode for approximately 1.5 years until a core average exposure of about 10,000 MWd/t was obtained (the equilibrium core average exposure is one-half the discharge exposure, for on-line refueling). The use of slightly enriched fuel in the first core would consequently involve a rather long period of poison control during which significant amounts of reactivity would be tied up by control poisons. A more satisfactory initial cycle can be obtained by using natural

uranium in the first core. With natural uranium, the HWR would operate for about 0.5 years in the poison control mode, after which on-line refueling with slightly enriched fuel would be initiated. The use of a natural uranium first core thus minimizes the period of poison control, and allows the more resource efficient on-line refueling to be initiated at an earlier date. Because natural uranium fuel is less expensive than slightly enriched fuel, first core costs are also reduced, resulting in lower fuel cycle costs because of reduced carrying charges on the initial core loading.

As indicated in Table 5.4.2-3, the reactor is initially loaded with 166.1 metric tons of natural uranium fuel contained in 8,800 fuel assemblies. This initial loading allows the reactor to operate for approximately one-half year without refueling, during which soluble boron in the moderator is used for reactivity control. After the first half year of operation, on-line refueling is initiated. After this time, fresh fuel having an initial enrichment of 1.2 w/o is charged into the reactor, while the most highly exposed fuel is discharged and sent to the spent fuel storage facility. Approximately 11 fuel assemblies must be charged/discharged for each full-power day of operation, resulting in an annual throughput of fuel of approximately 55.9 metric tons.

U_3O_8 and separative work requirements and the quantities of fissile material discharged in spent fuel are also shown in Table 5.4.2-3. Under the assumption of a 1% fissile loss during fabrication and a 0.5% loss during conversion, the annual U_3O_8 requirements* (at 75% capacity factor) are about 114 ST/GWe (88 MTU/GWe) while 30-year cumulative U_3O_8 requirements total 3,545 ST/GWe; these requirements are roughly 40% lower than those of current LWRs operating on the once-through uranium fuel cycle. It should be

*Throughout this report, U_3O_8 (uranium) requirements are generally expressed in short tons (ST) of U_3O_8 . One ST U_3O_8 is equal to 0.77 metric tonnes uranium metal (MTM or MTU).

noted that uranium utilization improvements to the light water reactor are under active development, and hence the uranium utilization of the LWR used in this section should not be interpreted to represent the best performance which can be obtained in the LWR; rather the LWR information is included solely to provide perspective to the HWR evaluation.

Separative work requirements are zero for the initial core because of the use of natural uranium fuel and are 39.0 and $1,152 \times 10^3$ SWU/GWe for the equilibrium annual and 30-year cumulative requirements, respectively; these requirements are roughly 1/3 of those of currently-operating LWRs. U-235 and fissile plutonium discharge quantities are also given in Table 5.4.2-3. Of particular note is the rather low concentration of fissile plutonium in the spent fuel--about half the concentration of plutonium in LWR spent fuel. The annual discharge fissile plutonium, 149 kg/GWe, is somewhat lower than that of currently operating LWRs. This contrasts with natural uranium fueling where considerably higher annual fissile plutonium discharges occur, and is attributable to the improved fissile utilization of the slightly enriched HWR which results in larger quantities of plutonium being consumed in situ. The plutonium discharged from the slightly enriched HWR also contains considerable quantities of Pu-240, and hence is unattractive for use in nuclear weapons. Pu-240 constitutes about 37% of the total plutonium discharged, and the fissile content is about 58% of the total plutonium content. Pu-240 constitutes a larger fraction of the plutonium discharge than is observed in the LWR because of the higher thermal-to-fast flux ratio of the HWR. Because of the higher thermal flux, more Pu-240 is produced while the lower fast flux reduces the destruction rate of Pu-240 which captures neutrons primarily in the epi-thermal energy range.

5.4.2.2 Economic Evaluation

Total busbar power costs for the slightly enriched (1.2 w/o) conceptual HWR are summarized in Table 5.4.2-4, while this information is shown graphically in Figure 5.4.2-2. Also shown in this table and figure are the power costs for a standard light water reactor, typical of large plants currently in operation. This light water reactor power cost information is included in order to provide perspective to the HWR economic evaluation, but should not be interpreted to represent the best economic performance which can be obtained in the LWR. Light water reactor fuel cycle improvements are currently under active development, and when implemented should result in some reduction in LWR fuel cycle costs.

The analysis summarized in Table 5.4.2-4 is based upon 30-year levelized fuel cycle cost analyses, and consequently the results differ slightly from the more simplified batch fuel cycle cost analyses employed in the optimization evaluation and reported in Section 4. Economic assumptions have also been changed somewhat, reflecting more recent revisions to the NASAP data base (Ref. 5.4-1). These economic assumptions are summarized in Table 5.4.2-5.

The busbar power costs summarized in Table 5.4.2-4 are comprised of three components: capital cost, O&M, and fuel cycle cost. The capital cost component is based upon the HWR capital cost of \$587/kWe and the LWR capital cost of \$543/kWe developed in Section 5.3. These plant capital costs are in January 1, 1978 U.S. dollars, and are exclusive of interest during construction, escalation during construction, contingency allowance and owner's cost. In addition, the cost of the heavy water inventory has been excluded from the plant capital cost, and included as a component of the fuel cycle cost. The contingency allowance has been excluded since it is assumed that the unit is one of an

established standard plant design; if one wished to include this component, an increase in capital cost of 10% for both units would be appropriate.

In order to establish power costs, interest during construction and owner's costs must be added to the base capital cost of \$587 and \$543/kWe for the HWR and LWR, respectively. Interest during construction will vary depending upon the length of time required for construction, the schedule of payments during this period, and the cost of money (interest). For the purpose of this assessment, a 10-year total construction period and the schedule of payments specified in Reference 5.4-2 were employed for both reactor types, along with a 4.525% cost of money (deflated effective interest rate). These assumptions result in an interest-during-construction cost of 21% of the base capital cost. Owner's costs were assumed to be 7.5%, based upon Reference 5.4-1. Adding interest during construction and owner's costs to the base capital cost results in a total capital cost for the HWR of \$764.8/kWe and of \$706.5/kWe for the LWR; the total capital cost of the HWR is thus evaluated to be approximately 8.3% greater than that of the LWR. These values of capital cost are utilized to establish the capital cost contribution to the total power cost presented in Table 5.4.2-4. It should be noted, that since the power cost analysis is performed in constant 1978 dollars, escalation during construction is properly excluded from the capital cost. The capital cost of the plant delivered at some future date will, of course, be higher when viewed in terms of the then current dollars because of escalation (inflation). For example, the cost of a plant for 1990 operation will be about \$1720/kWe in 1990 dollars, assuming an 8% escalation rate; however, a 1990 dollar would be worth only about \$0.40 in 1978 dollars if there were an 8% inflation rate. Other fuel cycle cost components would also be expected to escalate at the 8% rate, so that the relative contributions to the total power cost from the plant capital cost and various fuel

cycle cost components will be unaltered by the assumption of constant 1978 dollars.

In order to convert the total capital cost to mills/kW-hr, it is necessary to assume a plant capacity factor. For the LWR, a capacity factor of 75% is assumed, consistent with NASAP economic ground rules. However, a slightly higher value, 77.5%, was employed as a reference value for the HWR because of the increase in capacity factor that should be obtainable due to the on-line refueling feature of the HWR.* Light water reactors must be shut down periodically for refueling, and annual refueling intervals are typical at present, although many utilities are considering refueling at 18-month intervals. These refueling outages are utilized both to refuel the reactor and also to perform scheduled maintenance and inspections. The length of the refueling outage in the LWR has typically been dominated by the time required to perform maintenance and inspection, rather than by refueling itself; the duration of the refueling outage has varied considerably from reactor to reactor and has at times been quite long due to maintenance requirements. It is difficult to assess the increase in availability which should properly be attributed to the on-line refueling feature of the HWR, since it is hard to assess how much refueling/maintenance outage times would decrease if only maintenance operations were performed; if maintenance were the critical path item throughout the outage, outage times would not be reduced by eliminating refueling operations, and since similar maintenance operations must be performed on the HWR, no improvement in availability would be expected. In practice, some reduction in either the duration or frequency of the refueling/maintenance outage should be anticipated with on-line refueling.

*The lifetime average capacity factor of Ontario Hydro Candu - PHWR's is nearly 20 percentage points greater than the world average of PWR plants (Reference 5.4-7). This difference in capacity factor is not fully understood, and is believed to be due, at least in part, to factors other than inherent difference in HWR and LWR reactor plant design. The 2.5% percentage point difference in capacity factor employed in this evaluation is based on identifiable differences in design features, as discussed in subsequent sections of the text.

Although many LWR refueling/maintenance outages have been of excessive duration due to maintenance problems, outages of 6-to-7-week durations are not atypical when maintenance problems are routine, and outages of 3 to 4 weeks in duration have long been an industrial target. A recent evaluation of refueling/maintenance outages performed for DOE (Ref. 5.4-3) indicated that with existing LWR equipment refueling outages of about three weeks were possible and that outage times of 4.5 weeks were "realistic", with further reductions possible with equipment modifications. For the purpose of this evaluation, a five-week outage duration is assumed for LWR's; in view of the 6-to-7-week outage-times currently observed when maintenance is routine, the five-week duration assumed here does not seem unreasonable for the longer term. Under the assumption of a 75% capacity factor and a five-week refueling/maintenance outage, the availability between refuelings would be about 83%. It is difficult to establish precisely how much this five-week refueling/maintenance outage would be decreased for the HWR where refueling operations are eliminated through the use of on-line refueling, since this depends upon the amount of time where refueling operations are on the critical path, i.e., on the amount of time where refueling operations delay the required maintenance procedures. Estimates of one to two weeks however, appear reasonable for the additional time required for refueling above that required for maintenance. Assuming a one- or two-week reduction in the duration of the outage for the HWR results in capacity factors of 76.6 and 78.2%, respectively, assuming that the availability between refuelings is equal for both units (83%) and that outages are annual for both reactor types. If refueling/maintenance outages were extended to an 18-month interval, the HWR capacity factor which corresponds to a 75% capacity factor for the LWR would be 76% and 77% for one- and two-week reductions in the duration of the outage, respectively.

A second consideration is the potential for increasing the frequency between maintenance outages in the HWR. Presently, operating Candu plants employ an 18-month maintenance schedule, with a longer-term goal of delaying maintenance to 24-month intervals.

With such extended maintenance intervals, one might anticipate that the maintenance outage would be of somewhat longer duration, perhaps the full five weeks assumed for the LWR. If one assumes the same 83% availability between refuelings, a five-week maintenance outage, and a 24-month frequency, HWR capacity factor would increase to 77.5%, two and one-half percentage points greater than that of the LWR on an annual maintenance schedule. For the purpose of the reference economic assessment, an HWR capacity factor of 77.5% has been assumed; information on the sensitivity of power cost to capacity factor assumptions is also discussed in subsequent sections of this chapter. The resulting contribution of capital cost to the total busbar power cost is shown in Table 5.4.2-4. Although the HWR capital cost is evaluated to be about 8.3% higher than that of the LWR, the capital cost contribution to power cost is only 4.8% greater for the HWR because of the assumption of a 2.5 percentage point higher capacity factor.

The operational and maintenance costs (O&M) used in the analysis were obtained from Reference 5.4-1. Although no attempt was made to develop O&M cost in detail, the costs quoted in this reference were evaluated and are thought to be reasonable. Actual O&M costs for light water reactors in the United States for 1976 were 2.3 mills/kw-hr. This high average O&M cost was caused by major problems associated with a few plants; when the worst ten units were excluded, the 1976 O&M cost is 1.2 mills/kw-hr (or 1.4 mills/kw-hr in 1978 dollars). The 1.76 mills/kw-hr O&M cost for the LWR used in this analysis is thus not unreasonable for an average LWR in the future, assuming that the major problems which have plagued a number of plants are resolved. The O&M costs for the HWR are about 21% higher than the equivalent O&M costs for the LWR. The primary difference between LWR and HWR O&M costs is the cost of D₂O makeup and upgrading, the cost of valve and seal maintenance to ensure low D₂O leakage, and the cost of maintenance of the refueling machines. Other components of the O&M costs,

such as allowance for staff requirements, maintenance materials, supplies and expenses, nuclear liability insurance and operating fees, and administrative and general overhead expenses are expected to be similar for the two reactor types. Irrecoverably D_2O losses, which are assumed to be 1% of the inventory per year, would account for 0.20 mills/kw-hr of the O&M cost difference between the HWR and the LWR, while the cost of upgrading recoverable losses constitutes approximately 0.1 mills/kw-hr of the O&M cost. The remaining differential in O&M cost between the HWR and the LWR, about 3% of the total HWR O&M cost, appears reasonable to cover the added HWR maintenance requirements.

As Table 5.4.2-4 illustrates, fuel cycle costs for the HWR are lower than those of the LWR, and tend to compensate for the somewhat higher HWR O&M and capital cost components. HWR fuel cycle costs are lower primarily because of the reduced requirements for uranium (the uranium component includes both the cost of U_3O_8 and also of conversion) and of separative work. Carrying charges are also lower for the HWR, primarily because of the lower U_3O_8 and Separative Work Unit requirements. The cost of heavy water represents a substantial fraction of the HWR fuel cycle cost, particularly at low U_3O_8 prices. D_2O costs are partially, but not completely, compensated by the lower separative work requirements of the HWR. The fuel fabrication cost component of the HWR is roughly comparable to that of the LWR, because of the lower fabrication cost for HWR fuel (\$72/kg for the HWR compared to \$106/kg for the LWR) compensates for the higher throughput of fuel which results from the lower discharge burnup of the HWR. In the case of spent fuel disposal costs, where the price of spent fuel disposal is assumed to be essentially equal for both reactor types (\$145/kg for the HWR vs. \$150/kg for the LWR), the higher throughput of fuel in the HWR results in a fuel cycle cost component due to spent fuel disposal nearly twice that of the LWR.

At a U_3O_8 price of \$40/lb, the total busbar power cost from the HWR is evaluated to be slightly greater than that of the standard LWR, by about 0.46 mills/kw-hr. The HWR, however, is less sensitive to the price of U_3O_8 because of its lower U_3O_8 requirements. As a result, for higher U_3O_8 prices the total busbar power cost for the HWR will be lower than that of the LWR. The sensitivity of busbar power cost to the price of U_3O_8 is illustrated in Figure 5.4.2-3. Under the assumption of a 77.5% capacity factor for the HWR (with a 75% capacity factor for the LWR), equal power costs are obtained at a uranium price of about \$55/lb; if uranium can be purchased for less than this amount, LWR power costs will be lower than that of the HWR, while if the price of uranium exceeds this value, the HWR will prove more economic. Figure 5.4.2-3 also indicates that the break-even U_3O_8 price will rise to about \$75/lb if the capacity factor of the HWR proves to be equal to that of the LWR.

Obviously, the perception of the economic competition between the HWR and the LWR depends upon a large number of economic assumptions, many of which are quite uncertain, particularly with respect to their projection into the future. The sensitivity of the busbar power costs for the LWR and the HWR to various economic assumptions is illustrated in Figures 5.4.2-4 through 5.4.2-11. In each of these figures, an economic parameter is varied to establish the change in busbar power costs, while all other parameters remain unchanged from their reference value, as specified in Table 5.4.2-3.

The effect of HWR capital cost with respect to that of the LWR on power cost is illustrated in Figure 5.4.2-4. In these calculations, the LWR capital cost is held constant (and hence the LWR power cost does not change) while HWR capital cost was varied to encompass a range from 20% more expensive than the LWR to 5% less expensive than the LWR. As expected, the HWR power cost is quite sensitive

to the assumed HWR capital cost penalty. As noted previously, HWR power costs are higher than those of the LWR with an evaluated HWR capital cost penalty of 8.2% (reference value) and a U_3O_8 price of \$40/lb. Equal power costs would be obtained for this U_3O_8 price if the HWR capital cost would prove to be about 2.5% higher than that of the LWR. For a uranium price of \$100/lb, the HWR capital cost could be about 20% higher than that of the LWR before HWR capital costs would exceed those of the LWR.

The effect of capacity factor variations is illustrated in Figure 5.4.2-5. As the capacity factor is reduced, the power costs for the HWR increase at a somewhat more rapid rate than those of the LWR because of the relatively larger fraction of the total power cost which is attributable to plant capital costs; however, the relative competition between the LWR and HWR is not altered significantly for either U_3O_8 price assumption for reasonable values of capacity factor (assuming the capacity factor is the same in both units). The effect of differences in capacity factor between the two units is illustrated in Figure 5.4.2-6, where the capacity factor of the LWR is held constant at 75% and the capacity factor of the HWR varied between 70 and 80%. This figure illustrates that the capacity factor of the HWR would have to be about 5 percentage points higher than that of the LWR (or 80%) before HWR power costs would be as low as those of the LWR for \$40/lb U_3O_8 . At \$100/lb U_3O_8 , the capacity factor of the HWR could decrease by 5 percentage points below that of the LWR (or to 70%) before HWR power costs would rise to equal those of the LWR.

The sensitivity to D_2O price is illustrated in Figure 5.4.2-7. The reference value for the D_2O cost, \$213/kg, is representative of the current world market price. This price might be expected to increase, perhaps up to \$300/kg, when new D_2O production facilities are constructed to ever more increasing stringent

environmental criteria. On the other hand, alternate technologies for D_2O production are under development, and should they prove feasible, some reduction in D_2O cost might be anticipated. As this figure illustrates, a moderate reduction in the price of D_2O , to about \$175/kg, would reduce HWR power costs to those of the LWR at a U_3O_8 price of \$40/lb. At a U_3O_8 price of \$100/lb, the cost of D_2O could increase significantly, to more than \$300/kg, before the HWR power cost would increase to that of the LWR.

The effect of separative work (SWU) price on power costs is illustrated in Figure 5.4.2-8. The HWR is less sensitive to the SWU price because of the lower SWU requirements, as compared to the LWR. Costs higher than the \$100/unit reference value might be anticipated in the future for gaseous diffusion plants as the cost of energy increases, while some reduction in SWU cost might be anticipated with the development and deployment of advanced isotope enrichment technologies. As this figure illustrates, the economic performance of the HWR relative to the LWR is not particularly sensitive to reasonable variations in the price of separative work, and rather large changes in the price of separative work would be required to alter the conclusions on the relative competitiveness of the HWR under either the \$40 or \$100/lb U_3O_8 cost assumptions.

The sensitivity to fabrication costs is illustrated in Figure 5.4.2-9, where power costs for the HWR and LWR are plotted as a function of fabrication cost. As this figure illustrates, both the HWR and LWR show similar trends with fabrication cost. More significant is the effect of the difference in fabrication cost between that of the LWR and the HWR. The sensitivity of the power cost to this fabrication cost differential is illustrated in Figure 5.4.2-10 where the fabrication cost for the LWR has been held constant at its reference value of \$106/kg, and the fabrication cost for the HWR varied between \$30 and \$110/kg.

Increasing the LWR-HWR fabrication cost differential reduces HWR power cost, but as the figure illustrates fabrication costs below \$30/kg would be required to achieve equal power costs with that of the LWR at a uranium price of \$40/lb, while the HWR would still achieve lower power costs at \$100/lb should the fabrication cost of HWR fuel equal that of the LWR. The relative insensitivity of the HWR power cost to this fabrication cost differential is due to the relatively small contribution to the total fuel cycle cost from the fabrication component (cf. Figure 5.4.2-2). The sensitivity of HWR power costs to the fabrication costs would be significantly greater for natural uranium fueling, because of the much larger annual throughput of fuel.

Figure 5.4.2-11 provides power costs as a function of spent fuel disposal cost. A fairly wide range of spent fuel disposal costs were included in this analysis because of the great uncertainty in this parameter, particularly with respect to HWR fuel disposal. A reference value of \$145/kg (\$150/kg for the LWR) has been used in the reference economic analysis, but more recent DOE estimates for spent fuel disposal costs of LWR fuel have ranged to somewhat above \$200/kg. On the other hand, Canadian assumptions for the disposal of natural uranium fuel typically indicate spent fuel disposal costs of less than \$20/kg; because of the higher discharge exposure for slightly enriched fuel in the HWR, somewhat higher spent fuel disposal costs should be anticipated, but these should, nevertheless, be considerably lower than the reference value used in NASAP analyses. As this figure illustrates, the economic competition between the HWR and LWR is not significantly altered for either the \$40 or \$100/lb U_3O_8 assumptions, provided the same spent fuel disposal cost is utilized in both reactor types. However, any significant differential in the spent fuel disposal cost between the two reactor types could significantly impact perceptions of the economic competitiveness between the two units. For example, for the \$40/lb U_3O_8 cost assumption, reducing

the HWR spent fuel disposal cost to about \$70/kg (while retaining the \$150/kg cost for the LWR) would result in approximately equal busbar power costs.

A parameter which conveniently characterizes the economic performance of the HWR with respect to the LWR is the break-even U_3O_8 price at which both reactors will achieve equal busbar power costs. For U_3O_8 prices below this break-even value, the LWR will prove to be more economic, while prices higher than this break-even value will result in superior economic performance by the HWR. This break-even U_3O_8 price is shown as a function of various economic assumptions in Figures 5.4.2-12 through 5.4.2-15. It should be noted that although this break-even U_3O_8 price is a convenient indicator of the relative performance of the HWR with respect to the LWR, it is dependent upon the assumed economic performance of the LWR as well as that of the HWR. In this analysis, the "standard" LWR has been utilized. Any improvements to the LWR which serve to reduce LWR power cost will result in somewhat higher break-even U_3O_8 prices.

The effect of the HWR capital cost penalty and of the relative capacity factor of the HWR relative to the LWR on the break-even U_3O_8 price is illustrated in Figure 5.4.2-12. For the reference assumptions (an 8.3% capital cost penalty and a 2.5 percentage point advantage in capacity factor for the HWR) the break-even U_3O_8 cost is about \$55/lb. Should the HWR capital cost equal that of the LWR (zero penalty) the HWR would achieve lower power costs than the LWR for uranium prices above \$27/lb, while if HWR capital cost proves to be 20% greater than that of the LWR, uranium would have to sell for more than about \$70/lb before superior economic performance was obtained from the HWR. If HWR capacity factor should prove to be 5 or 8 percentage points higher (80 or 83%) than that of the LWR, HWR power costs would be lower than the LWR should the price of uranium be \$43 or \$27/lb, respectively.

As mentioned previously, a number of options for improving the uranium utilization of the LWR are under development, and it is anticipated that reductions in uranium requirement of 10-15% might be realized in the near term. As a result of the reduced uranium requirement, fuel cycle costs should decrease by about 5%. Such a reduction in LWR fuel cycle cost would increase the break-even U_3O_8 price from the \$55/lb (reference assumptions) to about \$90/lb.

Break-even U_3O_8 prices as a function of the cost of D_2O and separative work are illustrated in Figure 5.4.2-13. As the D_2O price is varied between \$100 and \$300/kg, the break-even U_3O_8 price varies between \$20 and \$80/lb. Similarly, varying the cost of separative work between about \$170 and \$50 per unit SWU results in a similar variation in the break-even price of U_3O_8 . The effect of HWR fabrication cost and of HWR and LWR spent fuel disposal costs on the break-even U_3O_8 price is shown in Figure 5.4.2-14. The break-even U_3O_8 price is observed to be relatively insensitive to the variation in these parameters, since spent fuel disposal and fuel fabrication represent relatively small contributions to the total power cost. Lastly, Figure 5.4.2-15 shows the sensitivity of the break-even U_3O_8 cost to the cost of money. If the cost of money (deflated effective cost of money) is reduced from the 4.525% assumed as the reference value to 3%, the break-even U_3O_8 cost decreases from \$55/lb to about \$35/lb, while increases in the cost of money will result in higher break-even U_3O_8 costs. The break-even cost increases as the cost of money is increased because of the relatively larger fraction of the total HWR power cost attributable to plant and D_2O capital cost.

5.4.2.3 Effect of Batch Fueling

One often suggested modification to the operating mode of the HWR is to employ batch fueling in place of the on-line refueling predicated in the reference conceptual HWR. The use of batch refueling might potentially simplify safeguards since it would limit fuel handling operations to well defined intervals. However, it should be noted that the batch-fueling mode as employed in LWRs provides little improvement in safeguards, per se. Rather, it is the difficulty of the refueling operation (i.e., the necessity to shut the reactor down, remove the reactor head and internals, etc.) which provides a significant barrier against the clandestine diversion of irradiated fuel. Consequently, simply employing a batch fueling mode in the HWR will do little to strengthen safeguards unless extensive modifications are made to the reactor plant to make such clandestine removal of fuel more difficult. In particular, design modifications to preclude easy access to individual fuel channels would probably be required. The purpose of this section is to summarize the effect of batch fueling on the uranium utilization performance of the HWR, if such modifications to preclude rapid or on-line refueling were to be made.

On-line refueling maximizes the uranium utilization of the HWR for two reasons:

1. On-line refueling maximizes the achievable discharge burnup obtainable from a given initial enrichment, and
2. On-line refueling maximizes the conversion ratio by eliminating neutron losses to control poisons.

If one ignores for a moment the effect of batch fueling on conversion ratio (i.e., Item 2 above), the core average burnup, E_a , achievable in the HWR will be the same whether online refueling or batch fueling is employed. This is because the core average fissile content and core average fission product burden which are characterized by the core average burnup defines the neutron balance necessary to maintain the reactor critical at end of cycle. The discharge burnup which can be achieved, E_d , is approximated by the formula:

$$E_d = \frac{2Z}{Z+1} E_a$$

where Z is equal to the number of batches. This equation is plotted as a function of the number of batches, Z , in Figure 5.4.2-16. For on-line refueling, where Z equals infinity, the achievable discharge exposure is twice the core average exposure. The achievable discharge exposure decreases as the number of batches present in the reactor is decreased, and equals the core average exposure for a single-batch refueling ($Z = 1$).

The interval between refuelings is also dependent upon the number of refueling batches employed (through the discharge exposure when other parameters such as core loading and capacity factor are held constant). The relationship between discharge exposure, batch number, and the interval between refuelings is given by the following equation:

$$T = \frac{E_d}{Z} \frac{L}{P\epsilon} = \frac{2}{Z+1} \frac{L}{P\epsilon} E_a,$$

where L is the fuel loading, P is the reactor thermal power, and ϵ is the capacity factor.

In the case of on-line refueling (i.e., where the number of refueling batches is very large), the interval between refueling becomes very small; with on-line refueling several refueling visits per day are made. This short interval between refueling is, of course, of no great concern with on-line refueling since the reactor is not shut down for the refueling operation. However, in the batch-fueling mode, the interval between refueling outages must be long (preferably one-year or longer, with six months as a minimum) in order not to adversely affect reactor availability. Consequently, the number of batches and the batch-fueling mode must be selected to be sufficiently low (i.e., the fraction of fuel replaced at each visit must be sufficiently large) to allow satisfactorily long periods of operation between refuelings. The effect of batch size on refueling interval is shown in Figure 5.4.2-17, where the refueling interval as a function of batch size is shown for a number of postulated core average discharge exposures. The implication of this figure is that for a given core average discharge exposure, the number of refueling batches is limited to rather small numbers, with a concomitant decrease in achievable discharge burnup and hence U_3O_8 utilization. In order to utilize a larger number of in-core batches, it is necessary to raise the core average discharge exposure (i.e., to increase initial enrichment).

The effect of batch fueling on uranium requirements is shown more explicitly in Table 5.4.3-6. This table compares the 30-year U_3O_8 requirements and the achievable discharge exposure for the batch fuel natural uranium and 1.2 w/o enriched HWR with that of on-line refueling, assuming either annual or semiannual refueling intervals. It should be noted that only the effect of batch fueling on the achievable discharge exposure is included in this tabulation, and that the effect of batch refueling on conversion ratio has been neglected. As a result of neglecting the effect of batch refueling on conversion ratio, the 30-year U_3O_8 requirements

shown in this table are somewhat optimistic, but nevertheless are illustrative of the degradation of uranium utilization which occurs with batch refueling. For natural uranium fueling, an annual refueling interval is impossible. Loading the core with fresh natural uranium fuel simply does not provide sufficient reactivity to allow the core to be operated for a full year; refueling is required after about 7 months of operation. However, as Table 5.4.3-6 illustrates, operation with natural uranium fueling is possible on a semiannual refueling schedule. On a semiannual refueling schedule, the core can be operated with approximately 1.26 batches resident in core, i.e., approximately 80% of the fuel must be discharged and loaded with fresh natural uranium fuel assemblies at each semiannual refueling outage. The achievable discharge burnup, 4,180 MWd/t is significantly lower than the 7,500 MWd/t achieved with on-line refueling. As a result, 30-year U_3O_8 requirements increase significantly, from the 5,000 ST U_3O_8 to about 8,800 ST U_3O_8 , a value significantly higher than that of the LWR. Consequently, one can conclude that batch fueling in the natural uranium HWR, even with semiannual refueling, is impractical. Of course, improved uranium utilization could be obtained by more frequent refuelings. This would, however, seriously compromise plant availability unless provisions were made for easy and rapid refueling; such easy and rapid fueling would, however, negate the safeguards incentive which provides the motivation for considering batch fueling in the first place.

For slightly-enriched fueling (1.2 w/o), two-batch fuel management, in which one-half the core is replaced at each refueling interval, is required to attain an annual refueling interval. In this case, the achievable discharge exposure is 13,100 MWd/t (compared to 19,750 for on-line refueling) and 30-year U_3O_8 requirements increase from the 3,545 of on-line refueling to 5,242 ST/GWe. Better U_3O_8 utilization is achieved with semiannual refueling,

where the 16,430 MWd/t discharge exposure results in 30-year U_3O_8 requirements of 4,222 ST/GWe.

Improved U_3O_8 utilization can be obtained with batch fueling by allowing the discharge exposure to increase. In the case of on-line refueling, the discharge exposure is 19,750 MWd/t, and so it appears reasonable to increase initial enrichment to achieve this discharge exposure with batch fueling; limiting the discharge exposure to this value is consistent with the selection of the 1.2 w/o value for on-line refueling, and hence should not necessitate fuel design modifications or core changes to accommodate increased peaking. Thirty-year U_3O_8 requirements for a batch-fueling mode which attains the discharge exposure of 19,750 MWd/t are given in Table 5.4.2-7. For an annual refueling interval, the initial enrichment must increase from 1.2 to 1.5 w/o U-235 in order to achieve this discharge exposure (i.e., the same discharge exposure as on-line refueling). The resulting 30-year U_3O_8 requirements are about 4600 ST, a value about 30% higher than that attained with on-line refueling and a value roughly comparable to that thought possible in improved LWRs. Thirty-year U_3O_8 requirements decrease to about 4100 ST U_3O_8 if semiannual refueling is employed; an initial enrichment of 1.35 w/o is required to obtain the same cycle length as is possible with an initial enrichment of 1.2 w/o with on-line refueling. Here, 30-year U_3O_8 requirements are about 16% higher than when on-line refueling is employed.

In summary, batch refueling is impractical for the natural uranium fueled HWR. With slightly enriched fueling, batch refueling might be considered, but it results in a very significant degradation of uranium utilization. When consideration is limited to annual refueling (which is highly desirable from a reactor availability standpoint), the batch-fueled slightly enriched HWR achieves a uranium utilization roughly comparable to that which should be possible with the implementation of fuel performance improvements

in the LWR; resource utilization is approximately equal to that of the natural uranium fueled HWR with on-line refueling. When semiannual refueling is considered, the uranium utilization of the batch-fueled slightly-enriched HWR is improved significantly, but U_3O_8 requirements are still about 16% higher than when on-line refueling is employed. Some improvement in U_3O_8 utilization might be obtained by further increasing the discharge exposure (that is the initial enrichment), but this would necessitate fuel design modifications to accommodate the higher burnup and some further reductions in power density to accommodate higher peaking. Furthermore, some additional degradation in uranium utilization should be anticipated when the effects of batch refueling on conversion ratio are considered. The design modifications which are required in order to perform batch fueling in a fashion which would provide a significant increase in security against diversion as compared to on-line refueling are also thought to be quite significant. In conclusion, batch fueling of the HWR does not appear attractive.

FIGURE 5.4.2-1

FUEL CYCLE FACILITIES FOR THE SLIGHTLY ENRICHED ONCE THROUGH CYCLE

Figure 5.4.2-2
POWER COST COMPONENTS

Figure 5.4.2-3

POWER COSTS AS A FUNCTION OF U_3O_8 PRICE
FOR THE ONCE-THROUGH URANIUM CYCLE

Figure 5.4.2-4
EFFECT OF HWR CAPITAL COST ON POWER COST

Figure 5.4.2-5
POWER COSTS AS A FUNCTION OF CAPACITY FACTOR

Figure 5.4.2-6

POWER COSTS AS A FUNCTION OF HWR
CAPACITY FACTOR (CONSTANT LWR CAPACITY FACTOR)

Figure 5.4.2-7

POWER COSTS AS A FUNCTION OF D₂O PRICE

Figure 5.4.2-8
POWER COST AS A FUNCTION OF
SEPARATIVE WORK PRICE

Figure 5.4.2-9

POWER COSTS AS A FUNCTION OF FABRICATION COST

Figure 5.4.2-10

POWER COSTS AS A FUNCTION OF HWR
FABRICATION COST (CONSTANT LWR
FABRICATION COST)

Figure 5.4.2-11
POWER COSTS AS A FUNCTION OF THE COST
OF SPENT FUEL DISPOSAL

Figure 5.4.2-12

BREAKEVEN U_3O_8 PRICE AS A FUNCTION OF HWR
CAPITAL COST PENALTY AND CAPACITY FACTOR

Figure 5.4.2-13

BREAKEVEN U_3O_8 PRICE AS A FUNCTION
OF SWU AND D_2O PRICE

Figure 5.4.2-14

BREAKEVEN U_3O_8 PRICE AS A FUNCTION OF
SPENT FUEL DISPOSAL AND HWR
FABRICATION COST

Figure 5.4.2-15

BREAKEVEN U_3O_8 PRICE AS A FUNCTION
OF THE COST OF MONEY

Figure 5.4.2-16

RATIO OF DISCHARGE TO CORE AVERAGE EXPOSURE AS A FUNCTION OF BATCH NUMBER

Figure 5.4.2-17

DAYS BETWEEN REFUELINGS AS A FUNCTION OF THE NUMBER OF BATCHES

TABLE 5.4.2-1
GENERAL REACTOR PERFORMANCE SPECIFICATIONS

A. Power Plant Performance

Thermal Power (MW)	
Total Nuclear	4029
Coolant	3800
Electrical Power (MWe)	
Gross	1343
Net	1260
Thermal Efficiency, %	31.3

B. Reactor Parameters

Core Volume (liters)	321,000
Core Dimension (m)	
Equivalent Diameter	8.8
Active Length	5.9
Core Power Density (Mw/l)	0.012
Coolant Flow Rate (Mg/sec)	14.2
Coolant Inlet Temperature (°C)	299
Coolant Outlet Temperature (°C)	337
Primary System Pressure (psia)	2250
Core Fuel Loading (MT)	
Uranium Fueling	166,056
Thorium Fueling	151,131
Number of Fuel Channels	740
Fuel Assemblies/Fuel Channel	12
Number of Fuel Assemblies	8880

TABLE 5.4.2-2
FUEL ASSEMBLY SPECIFICATIONS

Fuel Pellets

Form	Compacted and Sintered UO ₂ or ThO ₂ Pellets
Number per fuel Rod	~ 30
Diameter, mm	12.15
Average Stack Length, mm	472
Dish and Chamfer Volume, %	2.45
Pellet Density, gm/cc	10.6

Clad

Material	Zr-4
Outside Diameter, mm	13.1
Thickness, mm	0.42
Diametral Gap, mm	0.09
Clad Temperature, °C	~ 350

Fuel Assembly

Fuel Rods/Assembly	37
UO ₂ per Assembly, kg	~ 21.2
ThO ₂ per Assembly, kg	~ 19.3
Average Fuel Temperature, °C	~ 936
Maximum Fuel Temperature, °C	~ 1900

TABLE 5.4.2-3
FUEL MANAGEMENT SUMMARY
SLIGHTLY ENRICHED ONCE-THROUGH CYCLE

Initial Core

Enrichment, w/o	0.711 (Nat. u)
Core Loading, MT	166.1
Number of Fuel Assemblies	8880
Fabrication Loss, %	1.0

Reload Fuel

Enrichment, w/o	1.20
Discharge Exposure, MWD/MT	19,750
Fuel Throughput, MT/yr	55.85
Fraction of Core Replaced, %/yr	33.6
Fuel Residency Time, yr	2.97
Fabrication Loss, %	1.0
Conversion Loss, %	0.5

U₃O₈ Requirements (ST/GWe)*

Initial Core	172.6
Annual Equilibrium	114.2
30-year Cumulative	3546
30-year Cumulative, Net**	3426

SWU Requirements (MT SWU/GWe)*

Initial Core	0
Annual Equilibrium	31.4
30-year Cumulative	927.3
30-year Cumulative, Net**	942.0

Pu Discharge

Annual Discharge, kg/GWe	149
Concentration, kg fissile Pu/MT fuel	3.4

*0.2 w/o tails, 75% capacity factor

**30-year Cumulative, Net requirements are 30-year Cumulative requirements less credit for incore and excore fissile inventories at the end of the assumed 30-year reactor lifetime

TABLE 5.4.2-4
 BUSBAR POWER COST
 (mills/kw-hr)

U ₃ O ₈ Price, \$/lb	1.2 w/o HWR		Standard LWR	
	<u>40</u>	<u>100</u>	<u>40</u>	<u>100</u>
Fabrication	0.52	0.52	0.46	0.46
Fuel Disposal	0.99	0.99	0.58	0.58
Uranium ¹	1.45	3.54	2.50	6.12
SWU	0.43	0.43	1.79	1.79
Carrying Charges	0.34	0.76	0.86	1.64
D ₂ O	<u>2.04</u>	<u>2.04</u>	<u>-</u>	<u>-</u>
Total Fuel Cycle Cost	5.77	8.28	6.20	10.59
O&M	2.13	2.13	1.76	1.76
Capital Cost	<u>11.27</u>	<u>11.27</u>	<u>10.75</u>	<u>10.75</u>
Total Busbar Power Cost	19.17	21.68	18.71	23.10

1) Includes conversion

TABLE 5.4.2-5
ECONOMIC PARAMETERS
FOR THE ONCE-THROUGH URANIUM CYCLE

Conversion to UF ₆ , \$/kg	4
Separative Work, \$/SWU	100
Fabrication, \$/kg	
Natural UO ₂	64
1.2 Enriched UO ₂	72
Spent Fuel Disposal, \$/kg	145
Present Worth Rate, %	4.525
Fixed Charge Rate, %	10.0
Carrying Charge Rate, %	8.0
Tail Assay, w/o	0.2
Capacity Factor, %	77.5

TABLE 5.4.2-6
URANIUM REQUIREMENTS IN THE BATCH FUELING MODE

	Discharge Burnup (Mwd/MT)	Number of Batches	30-Year Cumulative U_{30} Requirements (SI/GWe)
<u>On-Line Refueling</u>			
Natural Uranium	7500	∞	4997
1.2 w/o Slightly Enriched	19750	∞	3545
<u>Annual Batch Refueling</u>			
Natural Uranium	*	*	*
1.2 w/o Slightly Enriched	13110	2.0	5240
<u>Semi-Annual Batch Refueling</u>			
Natural Uranium	4179	1.26	8810
1.2 w/o Slightly Enriched	16430	5.0	4220

*Not possible

TABLE 5.4.2-7
 URANIUM REQUIREMENTS IN THE BATCH FUELING MODE
 FOR A DISCHARGE EXPOSURE OF 19,750 Mwd/MT

Initial Enrichment w/o	30-Year Cumulative U_{30}	
	<u>Annual Refueling</u>	<u>Semi-Annual Refueling</u>
1.5	4597	-
1.35	-	4101

5.4.3 Plutonium Recycle

One method for improving the uranium utilization of the HWR is to recycle the fissile plutonium discharged in spent fuel. Although there are a number of ways in which this plutonium might be recycled, the most commonly envisioned method is to use the recovered plutonium to "top" natural uranium, i.e., to increase the fissile content of natural uranium fuel Ref. 5.4-4. This mode of recycle has some similarities with the slightly enriched uranium cycle discussed above, where enrichment facilities are used to increase the fissile content of natural uranium; here the recycle of fissile plutonium into natural uranium serves a similar function.

The fuel cycle facilities required for the plutonium recycle fueling mode are illustrated schematically in Figure 5.4.3-1; this figure also shows the mass flows for the equilibrium cycle of operation. Facilities for the reprocessing of spent fuel, for disposal of reprocessing wastes, and for the fabrication of mixed-oxide fuels must be added to the fuel cycle facilities required for once-through fueling, while facilities for isotopic enrichment and spent fuel disposal are eliminated. As in the case of once-through fueling, the initial core would be comprised of natural uranium fuel elements. This initial core would operate for approximately 0.5 years before refueling would be required. After the first half year of operation, on-line refueling would be initiated in which fresh uranium assemblies are loaded into the reactor and spent fuel discharged. Refueling with uranium fuel continues for a two-year period until recycle can begin (a two-year time period is assumed between fuel discharge and the initiation of recycle to allow sufficient time for fuel reprocessing and refabrication). During this period before recycle is initiated, either natural uranium or slightly enriched uranium fuel can be utilized; for the purpose of this analysis natural uranium fueling in the interim before recycle begins has been assumed.

The plutonium content of the spent fuel is recovered by reprocessing and used to supplement the fissile content of natural uranium fuel; the natural uranium fuel is assumed to be enriched to the same concentration of fissile plutonium as was present in spent fuel. Uranium recovered from the spent fuel is assumed to be treated as a waste, since this uranium has little value because of the low U-235 content of the spent fuel.

Plutonium content in the spent fuel and the plutonium enrichment in the refabricated recycled fuel is shown in Figure 5.4.3-2 as a function of years. As illustrated in this figure, recycle begins approximately 2.5 years after reactor startup and the plutonium enrichment in the refabricated fuel is comparable (slightly lower because of reprocessing, refabrication and decay losses) than the plutonium content of spent fuel. Since it has been assumed that the recovered plutonium has been added to an equal volume of (natural) uranium as had been present in the spent fuel, the recycle of plutonium into natural uranium increases the fissile content of natural uranium to slightly more than 1 w/o fissile, and as a result the achievable discharge exposure increases. This increase in discharge exposure is shown in Figure 5.4.3-3. The natural uranium fuel charged between 0.5 and 2.5 years is capable of obtaining a discharge exposure of 6,300 MWd/t in the conceptual HWR design. Once plutonium recycle begins 2.5 years after reactor startup, the equilibrium cycle discharge exposure increases to about 14,700 MWd/t. As a result of the higher discharge exposure with recycle, the quantities of fuel which must be charged each year are reduced, as is illustrated in Figure 5.4.3-4. During the period of time when natural uranium fueling is employed (0.5 to 2.5 years) fresh fuel must be charged (and spent fuel discharged) at a rate of approximately 140 MT/GWe-yr; if the reactor were to be operated with natural uranium fueling on the once thru cycle (i.e., without plutonium recycle) this charge/discharge rate would have to be continued throughout

the life of the reactor. Once plutonium recycle begins (i.e., after 2.5 years) the charge/ discharge rate is decreased to about 60 MW/GWe-yr because of the higher discharge exposure. Figure 5.4.3-4 also shows the annual quantities of plutonium discharged in spent fuel, and as this figure illustrates the quantities of plutonium discharged annually actually decreases once plutonium recycle is initiated. This is because the concentration of plutonium (i.e., the enrichment) in spent recycled fuel assemblies is only slightly greater than that of spent natural uranium assemblies, while the higher discharge exposure possible with recycle reduces the quantity of spent fuel discharged per year.

Fuel management information for the plutonium recycle fueling mode is shown in Table 5.4.3-1. The fuel cycle information is tabulated assuming a 75% capacity factor, 1% losses for both reprocessing and refabrication, and two-year ex-core time period between spent fuel discharge and reinsertion for fuel reprocessing and fabrication. The annual U_3O_8 requirements are approximately 77.8 ST/GWe, a requirement approximately 32% lower than the requirements of the slightly enriched once-through fuel cycle. The 30-year cumulative U_3O_8 requirements (2,690 ST/GWe) are about 24% lower than that required for slightly-enriched uranium once-through fueling; the reduction in 30-year requirements is somewhat lower than the reduction in the annual requirements because of the effects of the initial core loading and the ex-core fuel inventory.

Fuel cycle costs are compared for the plutonium recycle and slightly-enriched once-through uranium fueling modes in Table 5.4.3-2 (only fuel cycle costs are compared here since the capital and O&M costs are assumed to be equal in both cases). These fuel cycle costs are tabulated for two sets of economic assumptions: "near term" and "long term". These economic assumptions are specified in Table 5.4.3-3. For the slightly-enriched uranium

once-through case, the near-and long-term cost assumptions differ only in the price of U_3O_8 , with \$40/lb U_3O_8 assumed for the near term and \$100/lb U_3O_8 assumed in the long term. For the plutonium recycle case, the cost of reprocessing and mixed-oxide fuel fabrication is also assumed to decrease in the long term, because of increased experience and fuel throughputs.

As Table 5.4.3-2 indicates, the fuel cycle cost for the plutonium recycle mode is higher than that for slightly-enriched once-through fueling for both the near-term and long-term cost assumptions. Inspection of this table indicates that the plutonium recycle mode is less economic because of the higher contributions from fuel fabrication and fuel reprocessing/disposal, which prove to outweigh the savings in uranium and separative work. The costs attributable to fabrication are higher in the plutonium recycle mode because of the higher assumed cost for mixed-oxide fabrication (\$361 and \$210/kg for the near- and long-term assumptions, respectively) compared to the \$72/kg assumed as a fabrication cost for slightly-enriched fuel. Fabrication costs in the recycle mode are also slightly higher because of the somewhat greater fuel throughput which results from the slightly lower burnup in the recycle mode compared to slightly-enriched fueling (14,700 vs. 19,750 MWd/t for slightly-enriched fueling). Similarly, the recycle mode is penalized by the cost of fuel reprocessing (\$280 and \$197/kg for the near-and long-term assumptions, respectively) compared to the cost of spent fuel disposal on the once-through cycle (\$145/kg).

The sensitivity of the fuel cycle cost for the plutonium recycle and slightly-enriched once-through fueling modes to the cost of U_3O_8 is illustrated in Figure 5.4.3-5. The price of U_3O_8 would have to rise above approximately \$200/lb before the plutonium recycle mode would become more economic than slightly-enriched once-through fueling.

As noted at the beginning of this Section, the self-generated recycle mode discussed above is only one of a number of ways of

recycling plutonium. The economic evaluations performed for this recycle mode indicate that fuel cycle costs are higher than for once-through fueling because of higher cost contributions from fuel fabrication and reprocessing/disposal. This suggests that more favorable economics can be obtained in recycle modes which employ higher initial plutonium enrichment and which consequently achieve increased discharge burnup, so as to reduce the cost contribution from fuel fabrication and reprocessing/disposal. The efficiency of uranium utilization is not expected to change significantly. Unfortunately, time did not permit the quantitative evaluation of these various alternative modes of plutonium recycle.

Figure 5.4.3-1

FUEL CYCLE FACILITIES FOR THE URANIUM CYCLE WITH Pu RECYCLE

Figure 5.4.3-2

PLUTONIUM CONCENTRATIONS IN SPENT AND RECYCLE
FUEL ASSEMBLIES

Figure 5.4.3-3

DISCHARGE EXPOSURE AS A FUNCTION OF TIME
IN THE URANIUM CYCLE WITH Pu RECYCLE

Figure 5.4.3-4

FUEL THROUGHPUTS AND FISSILE PLUTONIUM DISCHARGE
FOR THE URANIUM CYCLE WITH Pu RECYCLE

Figure 5.4.3-5

FUEL CYCLE COST AS A FUNCTION OF U_3O_8 PRICE
FOR THE ONCE-THROUGH CYCLE
AND WITH PLUTONIUM RECYCLE

TABLE 5.4.3-1
FUEL CYCLE INFORMATION FOR THE
URANIUM CYCLE WITH PLUTONIUM RECYCLE

Initial Core

Enrichment, w/o	0.711 (Nat. U)
Core Loading, MT	166.1
Fabrication Loss, %	1.0

Reload Fuel (First 2.0 yrs)

Enrichment, w/o	0.711 (Nat. U)
Discharge Exposure, MWD/MT	6300
Fuel Throughput, MT/yr	175.4
Fabrication Loss, %	1.0

Reload Fuel (Remaining Years)

Enrichment, w/o	Nat. U + Self Generated Pu Recycle
Discharge Exposure, MWD/MT	14740
Fuel Throughput, MT/yr	74.8
Fissile Losses, %	
Fabrication	1.0
Reprocessing	1.0
Ex-Core Time, yrs	2

U₃O₈ Requirements (ST/GWe)*

Initial Core	170.9
Annual Equilibrium	77.2
30-Year Cumulative	2671
30-Year Cumulative, Net**	2315

SWU Requirements 0

*75% Capacity factor

**30-year cumulative, Net requirements are 30-year cumulative requirements less credit for incore and excore fissile inventories at the end of the assumed 30-year reactor lifetime

TABLE 5.4.3-2
FUEL CYCLE COST FOR PLUTONIUM RECYCLE FUEL MODE
(Mills/kw-hr)

	Plutonium Recycle Mode		1.2 w/o Once Through Cycle	
	<u>Near Term</u>	<u>Long Term</u>	<u>Near Term</u>	<u>Long Term</u>
Fabrication	2.59	2.25	0.52	0.52
Fuel Reprocessing/ Disposal	2.58	1.81	0.99	0.99
Uranium	1.05	2.63	1.45	3.54
SWU	0	0	0.43	0.43
Carrying Charges	0.40	0.72	0.34	0.76
D ₂ O	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>
Total Fuel Cycle Cost	8.66	9.45	5.77	8.28

TABLE 5.4.3-3
ECONOMIC ASSUMPTIONS FOR THE
RECYCLE MODE

	<u>Near Term</u>	<u>Long Term</u>
U ₃ O ₈ , \$/lb	40	100
Pu/UO ₂ Fabrication, \$/kg	361	310
Reprocessing, \$/kg	280	197
Present Worth Rate, %	4.525	4.525
Fixed Charge Rate, %	10.0	10.0
Carrying Charge Rate, %	8.0	8.0
Capacity Factor, %	75	75

5.4.4 Denatured Thorium Fuel Cycles

As supplies of reasonably-priced U_3O_8 are consumed, it may be desirable to shift to thorium fueling in the HWR since significant further reductions in uranium requirements can be realized with this fueling mode. The "conventional" thorium cycle, in which essentially pure U-235 or U-233 is used to enrich thorium, has been extensively evaluated in Canada (for example, see Reference 5.4-5). This section is limited to a discussion of a denatured thorium cycle, which is thought to offer greater assurance against proliferation. Denatured thorium fuel consists of a mixture of thorium and uranium oxides. The uranium component is enriched to equal to or less than 20 w/o U-235 or equal to or less than 12 w/o U-233 (or to some intermediate value if both U-233 and U-235 are present). At these enrichments, the uranium component is unsuitable for use in nuclear weapons without isotopic upgrading, even if chemically separated from the thorium. The uranium-to-thorium ratio is selected to achieve the desired cycle length.

There are two denatured thorium fuel cycles of interest. The first, the denatured U-235/thorium cycle utilizes U-235 to supply the fissile requirements for initial loadings and for annual makeup. The U-235 is, of course, obtained by isotopic enrichment of natural uranium. Spent fuel would be reprocessed to recover the bred U-233 and residual quantities of U-235, and this recovered uranium would be recycled to reduce the demand for U-235. This fuel cycle is discussed in Section 5.4.4.1.

The second denatured thorium fuel cycle of interest is the U-233/thorium cycle. This fuel cycle requires an exogenous supply of U-233 for initial loadings and for annual makeup. U-233 requirements could be obtained from a number of sources, such as either thorium-fueled converter reactors or thorium-blanketed breeder reactors, and the bred U-233 would subsequently be denatured with U-238 (tails assay uranium) for subsequent use in the denatured U-233/thorium-fueled HWR. Here also, the spent fuel would be reprocessed to recover

the residual quantities of fissile uranium and recycled to reduce the demand for U-233. This fuel cycle is discussed in Section 5.4.4.2.

Since U-238 is present in limited quantities in denatured thorium fuel, some plutonium is produced. One option for utilizing this plutonium is as enrichment for a thorium-fueled HWR. Since plutonium is chemically separable from thorium, fresh Pu/thorium fuel would have to be subject to the same safeguards as Pu/uranium fuel. However, the bred fissile material, U-233, could be denatured with U-238 (tails uranium) and subsequently used at dispersed-sited reactors as part of the exogenous U-233 requirements of the denatured U-233/thorium fuel cycle. This fueling option in the HWR is discussed in Section 5.4.4.3.

As will become evident after reviewing Sections 5.4.4.1 through 5.4.4.3, the performance of the thorium-fueled HWR is dependent upon the achievable discharge exposure (i.e., the initial enrichment). Since the costs assumed in this analysis for refabricating U-233-bearing fuels are many times higher than the cost of fabricating slightly-enriched uranium fuel and since the cost of reprocessing is higher than the assumed cost for spent fuel disposal, high discharge exposures, which minimize the quantities of fuel refabricated and reprocessed each year, are favored from an economic standpoint.

In general, discharge exposures in excess of 30,000 MWd/t are necessary to attain economic parity with the slightly-enriched once-through fuel cycle for reasonable U_3O_8 prices. Consequently, some caution should be exercised in reviewing the thorium cycle fuel management and economic performance information, since the conceptual HWR design has been predicated on somewhat lower discharge exposures, i.e., the 19,750 MWd/t discharge exposure achieved in the once-through uranium cycle with an initial fuel enrichment of 1.2 w/o. Some modification to the fuel design will undoubtedly be necessary to obtain discharge exposures much in excess of 20,000 MWd/t, and such high discharge exposures may

also prove to have some ramifications to the HWR design. In particular, the precursor isotope to the U-233 bred in the thorium fuel cycle, Pa-233, has a much longer half life (27 days) than does the corresponding precursor isotope in the uranium cycle, Np-239 with a 2.3 day half life. The longer half life of protactinium results in transients in the quantities of U-233 present after reactor shutdown, during ascents to power, and during maneuvers when the power level in the reactor is being changed. These transient U-233 concentrations may place more severe demands on the control system than imposed by the uranium fuel cycles, and may also lead to some limitations in the capability of the reactor to execute load follow maneuvers. More detailed evaluations of thorium utilization which address in a more complete fashion the design and operational performance considerations of thorium fueling should be performed prior to any commitment to deploy thorium fuel in the HWR.

5.4.4.1 Denatured U-235/Thorium Fuel Cycle

In the denatured U-235/thorium fuel cycle, U-235, extracted from natural uranium by isotopic enrichment facilities, is utilized to supplement recycled fissile uranium recovered from the spent fuel of previous years of operation. In this fuel cycle, the initial core and early reload batches (before recycle can begin) utilizes fuel containing a mixture of thorium and uranium, with the uranium component enriched to 20 w/o U-235. After discharge from the reactor, spent fuel is reprocessed to recover the residual quantities of uranium (containing U-235, U-233, and small quantities of other uranium isotopes) which is recycled to reduce the requirements for U-235.

Since the fissile content of this recycled material is less than that required to achieve the specified cycle length (because of depletion of fissile material during the prior cycle of irradiation), U-235 must be added. There are several options for adding the required makeup quantities of U-235. One option is to add denatured uranium (20 w/o U-235 in uranium) in sufficient quantities to increase the fissile content to that required to achieve the specified cycle length. This option has the advantage that only denatured makeup uranium is required. However, because of the U-238 present in the denatured uranium makeup, the total U-238 content will increase and the thorium content will decrease from recycle pass to recycle pass; as a result, increasingly large quantities of plutonium would be generated at the expense of diminished U-233 production.

A second option would be to "top" the recycled fuel with more highly enriched uranium (greater than 20 w/o) to provide the additional U-235 requirements. The enrichment of this topping material is selected to meet the denaturing requirements (between 12 and 20 w/o, depending upon the relative concentrations of U-233 and U-235) when mixed with the recycled uranium recovered from the spent fuel; "topping" enrichments between 20 and 70 w/o U-235 are required, depending upon the recycle pass. This option has the disadvantage of requiring more highly-enriched uranium but avoids the dilution of the recycle uranium with excess quantities of U-238 with a concomitant increase in plutonium production and reduction in U-233 production. Safeguard concerns associated with the use of such highly-enriched uranium can be largely avoided by either colocating the enrichment facility with the safeguarded reprocessing plant, or by shipping the recycle uranium to the enrichment facility for topping. Even if the highly enriched uranium were shipped from an enrichment facility to the reprocessing facility, safeguards would be much more easily applied than would be the case where weapons usable material were

contained in fresh fuel shipped to a large number of dispersed-sited reactors. This option of "topping" with more highly enriched uranium was assumed in developing the fuel management information discussed in subsequent sections of this chapter.

Fuel cycle facilities required for the denatured U-235/thorium fuel cycle are shown in Figure 5.4.4-1(a). This figure also illustrates the equilibrium cycle mass flows for a specific fuel management scheme in which a discharge fuel exposure of about 14,000 MWd/t is achieved. As indicated in the figure, the uranium content (U-235, U-233 and small quantities of other uranium isotopes) of the spent fuel discharged from the HWR is recovered by reprocessing (a modified Thorex process would be employed). Since plutonium will be bred from the U-238 denaturant, the reprocessing facility must be safeguarded, as must facilities to store the recovered plutonium. Although existing fabrication facilities could probably be utilized to fabricate fresh denatured U-235/thorium fuel, new facilities will be required for recycle fuel fabrication. Recycled fuel will have to be fabricated in specially designed hot-fuel fabrication facilities because of the gamma activity of the daughters of the U-232 which is produced in trace quantities along with U-233 in thorium-bearing fuels. This facility will probably have to employ both remote fabrication and remote maintenance.

Because of the on-power refueling feature of the HWR, the discharge fuel exposure achieved can be readily selected by adjusting the initial enrichment of the fuel. In the case of the LWR, the choice of initial enrichment/discharge burnup is limited because of the need to refuel at discrete intervals; this consideration generally precludes operating with low discharge exposures. Thus, there is no single denatured thorium fuel cycle in the HWR, but rather a spectrum of fuel cycles is possible, each having a different discharge fuel exposure.

As the discharge exposure is decreased, the conversion ratio will be improved because the fission product burden is reduced and hence fewer neutrons are lost to parasitic captures in fission products. Consequently, less U-235 will be required for makeup when a low discharge burnup is employed. However, when such low burnup fuel management schemes are utilized, fuel must be frequently reprocessed and refabricated and the cost for these services will adversely affect fuel cycle cost economics. Fissile material losses during reprocessing and refabrication, and ex-core fissile inventories, will also increase as the cycle length is reduced, partially negating the benefits of the improved conversion ratio.

As the discharge burnup is increased, larger quantities of U-235 makeup will be required because of the decrease in conversion ratio which results from increased neutron losses to the larger quantities of fission products produced as a result of the higher burnup. In this case, additional costs will be incurred for the purchase of makeup U-235, but the cost of reprocessing and fuel fabrication will be reduced since smaller quantities of fuel would be reprocessed and fabricated each year. The optimal performance of the denatured thorium fuel cycle with respect to U-235 utilization consequently occurs at very low discharge exposures, while the optimum from the standpoint of economic performance typically occurs at somewhat higher fuel discharge exposures and occurs as a result of the economic tradeoff between the increased costs of makeup U-235 and the decreasing cost of fuel fabrication and reprocessing services.

The effect of initial U-235 enrichment on the achievable burnup of fresh denatured U-235/thorium fuel (prior to uranium recycle) is illustrated in Figure 5.4.4-2. The initial enrichment is defined as the percentage of the total heavy metal content (uranium plus thorium) which is U-235. This enrichment is obtained by varying the ratio of uranium to thorium in the fuel rod (the

enrichment of U-235 in the uranium component always remains constant at 20 w/o). The uranium-to-thorium ratio required to obtain the specified initial U-235 enrichment is also shown on Figure 5.4.4-2. As this figure illustrates, the achievable discharge burnup can be varied over a rather large range by relatively small changes in the initial enrichment; increasing the initial enrichment from about 1.9 w/o to 2.5 w/o changes the achievable discharge burnup from about 8,000 to 37,000 MWd/t.

Figure 5.4.4-2 also shows the quantity of fissile plutonium discharged each year. The quantity of discharged fissile plutonium decreases from about 22.3 to 8.8 kg/GWe as the initial enrichment is increased from 1.9 to 2.8 w/o. The decrease in the fissile plutonium discharge results from the reduced fuel throughput as initial enrichment and discharge exposure is increased. The enrichment of fissile plutonium in the discharged fuel increases as the initial enrichment is increased from (0.02 to 0.05 w/o for initial enrichments of 1.9 to 2.8), but the lower annual fuel throughput results in smaller quantities of fissile plutonium discharged per year for the higher burnup fuel.

In order to characterize the fuel cycle and economic performance of the HWR operating on the denatured U-235/thorium cycle, two fuel management schemes have been explored in some detail. These two fuel management schemes differ in the discharge exposure which is attained: the first fuel management scheme employs an initial enrichment of 2.0 w/o U-235 and attains a relatively low discharge exposure (13,700 MWd/t), while the second employs an initial enrichment of 2.5 w/o to attain a discharge exposure of about 37,000 MWd/t. As noted above, superior fissile utilization is achieved when lower discharge exposures are employed, since the conversion ratio is highest at low exposures where losses of neutrons to fission products is small; on the other hand, fuel cycle cost economics tend to favor somewhat higher discharge

exposures where the throughput of spent fuel is lower. The low-discharge burnup case is thus representative of fuel management options which seek to minimize U-235 requirements, while the high burnup case is representative of fuel management schemes which seek to minimize fuel cycle costs.

Fuel cycle information is summarized in Table 5.4.4-1. Information is provided for both the 13,700 and 37,000 MWd/t denatured thorium fuel cycles, as well as for the slightly enriched uranium fuel cycle, for comparison. The reactor design is unmodified from that developed for slightly-enriched uranium fueling. The core loading is slightly lower for the denatured thorium cycle because of the lower density of thorium fuel. Comparing the U_3O_8 requirements given in this table with those of the slightly enriched once-through uranium cycle indicates that annual (and 30-year net) U_3O_8 requirements are reduced by about 53 and 63% for the high and low burnup cases, respectively. The 30-year cumulative requirements are reduced by a somewhat smaller factor, by about 30%, because of the higher initial loadings required for thorium-fueled systems and because of the ex-core fissile inventory (the fissile material which has been discharged from the reactor and which resides some place in the fuel reprocessing-refabrication stream awaiting recycle). The 30-year cumulative separative work requirements for the denatured thorium fuel cycle are about twice that required for slightly-enriched uranium fueling. As Table 5.4.4-1 illustrates, separative work requirements are higher because of the initial core loading and because of ex-core inventories, since the annual separative work requirements are roughly comparable with those of uranium fueling. The annual fissile plutonium discharge is only a small fraction of that of the uranium cycle (6% to 12%, for the high and low burnup denatured thorium cases, respectively).

It is interesting to note that although the annual reload U_3O_8 requirements are about 21% lower for the low burnup fuel management schedule as compared to the high burnup schedule, the 30-year cumulative requirements are only slightly (about 2%) lower. In the low-burnup case, the annual throughput of fuel is much higher (80.4 vs. 29.8 metric tons per year). As a result, larger quantities of fuel are reprocessed and refabricated each year in the case of a low burnup schedule. Because of the more frequent reprocessing and refabrication, fissile losses during these operations are increased. The ex-core fuel inventory, which under the assumptions of this analysis is twice the annual fuel throughput (because of the assumption of a two-year ex-core time) is also much larger for the low burnup schedule. The impact of reprocessing and fabrication losses and of the ex-core inventory on the 30-year cumulative U_3O_8 requirements is illustrated more clearly in Table 5.4.4-2 where the U_3O_8 equivalent of losses and ex-core inventory has been tabulated. As this table illustrates, U_3O_8 losses and ex-core inventory requirements under the reference fuel cycle assumptions are about 2-1/2 times larger (681 vs. 259 ST/GWe) for the low-burnup schedule. If there were no fissile losses due to refabrication and reprocessing and if the ex-core inventory could be reduced to zero, the 30-year cumulative U_3O_8 requirements for the low burnup schedule would be about 20% lower than those of the high burnup schedule. Thus, if one is to realize the resource conservation benefits of the low burnup schedule, fuel cycle facilities must be designed to reduce fabrication and reprocessing losses, and to minimize the time between spent fuel discharge and the reloading of recycle fuel assemblies (ex-core time). The last entry of Table 5.4.4-2 tabulates the 30-year cumulative U_3O_8 requirements for an alternate set of loss and ex-core time assumptions which are thought to be reasonably achievable targets. In particular, reprocessing and fabrication losses are assumed to be 0.5% each, and the ex-core time has been reduced to one year. Under these alternate fuel cycle assumptions, the 30-year cumulative U_3O_8

requirements are about 10% lower for the low-burnup schedule as compared to the high burnup schedule.

Estimates of the fuel cycle cost for the denatured U-235/thorium fuel cycle are compared with the slightly-enriched once-through uranium cycle in Table 5.4.4-4, while pertinent economic parameters are given in Table 5.4.4-3. Fuel cycle costs are tabulated under two sets of assumptions: near-term cost assumptions and long-term cost assumptions. The near and the long-term cost assumptions differ primarily with respect to the price of U_3O_8 , with 40 and 100 \$/lb prices assumed, respectively. The cost for fabrication and reprocessing are also somewhat different under the near and long-term assumptions: the prices are assumed to decrease somewhat with time, due to higher throughputs and increased experience. Of particular note are the rather high fabrication costs for U-233-bearing fuel as compared to the fabrication cost of slightly enriched uranium fuel, and the higher cost of reprocessing as compared to the cost of spent fuel throwaway. The higher refabrication and reprocessing cost will tend to favor higher discharge exposures, because of the reduced fuel throughput.

As Table 5.4.4-4 illustrates, fuel cycle cost for the denatured U-235/thorium cycle are highly dependent on the discharge burnup employed, primarily because of the influence of this parameter on the contributions to the total fuel cycle cost attributable to fabrication and reprocessing. For the low-discharge-burnup case, the higher cost attributable to fabrication and reprocessing, as well as the higher separative work and carrying charges which characterize thorium-fueled systems, result in significantly higher fuel cycle costs than are obtained on the slightly enriched uranium once-through cycle. The economic competitiveness of the denatured thorium fuel cycle is improved for the high burnup case, since the lower annual throughput of fuel results in decreased fabrication and reprocessing costs. Fuel cycle costs for denatured

thorium and slightly-enriched uranium fueling are shown in Figure 5.4.4-3 as a function of U_3O_8 price (the long-term assumptions have been used for other economic parameters). Compared to uranium fueling, fuel cycle costs are observed to increase less rapidly for the thorium cycle as the price of U_3O_8 increases, because of the reduced U_3O_8 requirements. As indicated in this figure, the price of uranium would have to increase to about \$180/lb before the denatured thorium fuel cycle would be competitive with slightly-enriched once-through uranium fueling, even for the high burnup case. Uranium price would have to be quite high before the low burnup case becomes economically competitive under the economic assumptions of this analysis.

One major reason that the denatured thorium cycle does not compete with the slightly-enriched uranium fueling until the price of uranium becomes rather high is the high cost of refabricating U-233-bearing fuel (567 and 487 \$/kg for the near-term and long-term assumptions, respectively). The sensitivity of the fuel cycle cost to the cost of refabrication is illustrated in Figure 5.4.4-4. The fuel cycle cost component attributable to refabrication is particularly important for the low discharge burnup case, because of the high throughput of fuel which must be refabricated each year. It would appear that if the denatured thorium fuel cycle is to be economically competitive with once-through uranium fueling for reasonable U_3O_8 prices, less expensive methods of refabricating U-233-bearing fuel must be developed.

5.4.4.2 Denatured U-233/Th Cycle

The denatured U-233/thorium fuel cycle is similar to the U-235/thorium cycle discussed above, except that U-233 is used in place of U-235 to provide the initial core and makeup fissile requirements. The required fuel cycle facilities are shown in Figure 5.4.4-1(b). These facilities are identical to those

required for the U-235/thorium cycle, except that the enrichment facility is replaced by an exogenous source of U-233, and all fuel which is fabricated bears U-233. Spent fuel discharged from the HWR is reprocessed to recover the residual quantities of uranium (containing U-233 and small quantities of other uranium isotopes) which is recycled to reduce the requirements for U-233. The fuel contains a mixture of uranium and thorium, with the uranium content limited to an enrichment of 12 w/o U-233, so that even if chemically separated from the thorium, it would not constitute weapons usable material. Since some plutonium will be present in the spent fuel, because of the U-238 denaturant, the reprocessing facility must be safeguarded. Plutonium recovered from spent fuel could be stored, as shown in this figure, or utilized to enrich a thorium-fueled HWR plutonium burner (as discussed in Section 5.4.4.3) in order to breed additional quantities of U-233.

The performance of the HWR operating on the denatured U-233/thorium fuel cycle is illustrated in Figure 5.4.4-5 which shows the annual U-233 makeup requirement as a function of initial enrichment. Also shown in this figure is the discharge burnup achievable for each value of the initial U-233 enrichment. As in the case of the U-235/thorium-fueled system, makeup U-233 requirements decrease as the cycle length is reduced, while economic considerations may favor higher discharge exposures at the expense of increased U-233 consumption, depending upon the price of U-233 and of fuel fabrication and reprocessing services. As Figure 5.4.4-5 illustrates, the denatured U-233/thorium fuel cycle has a net consumption of fissile material (even when credit is taken for the fissile plutonium produced). Thus, self-sufficient operation, where no U-233 is required for makeup in the equilibrium cycle, is impossible with denatured fueling. This mode of operation is possible with highly-enriched fuel (Ref. 5.4-5), but the reduction in the quantity of U-233 produced in denatured fuels precludes this mode of operation.

The quantity of fissile plutonium discharged annually is shown in Figure 5.4.4-6 as a function of initial enrichment. Although the enrichment of fissile plutonium in the spent fuel increases from about 0.01 to 0.06 w/o heavy metal as the initial U-233 enrichment is increased from 1.4 to 1.9%, the annual quantities of fissile plutonium discharged in spent fuel decreases from about 52 to 15 kg/GWe-yr over the same enrichment range, because of the decrease in fuel throughput which occurs as the initial enrichment and hence the discharge burnup is increased.

The 30-year U-233 requirements are summarized in Figure 5.4.4-7 as a function of initial enrichment (discharge fuel burnup). The total U-233 requirements have been divided up into four components: the U-233 inventor of the initial core, U-233 consumed by the HWR, U-233 lost due to reprocessing and refabrication losses (a 1% reprocessing and a 1% refabrication loss is assumed), and the ex-core U-233 inventory (two-year ex-core time assumed). As Figure 5.4.4-7 illustrates, increasing the discharge burnup results in a greater consumption of U-233; this is because the conversion ratio is decreased due to neutron losses to the increased fission product burden. However, as this figure also illustrates, both reprocessing and fabrication losses and the ex-core inventory increase as the burnup is reduced. Fabrication and reprocessing losses increase because of the larger throughput of fuel at low discharge exposures, while the ex-core inventory increases since larger quantities of fresh fuel must be loaded into the reactor before recycle can begin. As Figure 5.4.4-7 illustrates, the net effect is that a minimum in U-233 requirements is reached at a burnup of about 15,000 MWd/t; for higher burnups, the degradation in conversion ratio results in higher U-233 requirements, while for lower burnups reprocessing and fabrication losses and the higher ex-core fissile inventory requirements result in increased U-233 requirements.

In the calculations described above, the effect on U-233 requirements of the buildup of higher uranium isotopes during repeated recycle passes was not included. More detailed cycle-by-cycle calculations were performed for the near optimal case having an initial enrichment of 1.55 w/o U-233 and a discharge burnup of 14,000 MWd/t. Fuel management information for this case is summarized in Table 5.4.4-5. As this table illustrates, the 30-year U-233 requirement is about 6,850 kg/GWe, while the annual U-233 requirement is 110 kg/GWe.

Estimates of the fuel cycle cost for the denatured U-233/thorium fuel cycle are shown in Table 5.4.4-6. The economic parameters which were used in this evaluation were identical to those for the denatured U-235/thorium cycle (cf. Table 5.4.4-3). Fuel cycle costs for the denatured U-233/thorium cycle will, of course, depend upon the price for which U-233 can be purchased. Since knowledge of the U-233 producing system would be required in order to establish the U-233 price, the fuel cycle cost evaluation was performed parametric in this quantity, with U-233 prices varying between \$10 and \$30 per gram.

As Table 5.4.4-6 indicates, the denatured U-233/thorium cycle is unlikely to be economically competitive with slightly-enriched uranium fueling in the near term while the price of uranium is relatively low (\$40/lb). Under the near-term cost assumptions, the fuel cycle cost for U-233/thorium fueling is about 5.6 mill/kW-hr greater than for the slightly-enriched uranium cycle, even under the assumption of a U-233 price of \$10/gm. Table 5.4.4-6 illustrates that the main reason for the higher denatured U-233/Th fuel cycle costs is the higher cost components attributable to fabrication and spent fuel reprocessing. These increased costs result from the higher cost of fabricating denatured U-233-bearing fuel (\$567/kg vs. \$72/kg for SEU fuel), the higher cost of fuel reprocessing as compared to spent fuel disposal (\$291/kg

vs. \$145/kg), and the somewhat higher fuel throughputs which result from the lower discharge exposure of the reference denatured thorium fuel cycle (14,000 MWd/t vs. 19,750 MWd/t for the slightly-enriched case). In the longer term, as the price of U_3O_8 increases (and the assumed price of refabrication and reprocessing decreases) the U-233/thorium cycle becomes more economically competitive, but as Table 5.4.4-6 illustrates, the denatured thorium cycle is still at a disadvantage with respect to slightly-enriched uranium fueling even at a U-233 price of \$10/gm.

In Figure 5.4.4-8, the fuel cycle costs for slightly enriched uranium fueling as a function of U_3O_8 price are compared with the fuel cycle costs for the denatured U-233/thorium cycle as a function of U-233 price (all other economic assumptions are long term). This information was utilized in Figure 5.4.4-9 to determine the breakeven U-233 price as a function of the price of U_3O_8 ; that is, this figure provides the price one could afford to pay for U-233 as a function of U_3O_8 price in order to have equal fuel cycle costs between the denatured U-233/thorium and once-through uranium fuel cycles. As this figure illustrates, if U-233 could be obtained free, the denatured U-233/thorium fuel cycle would be more economic than slightly enriched uranium fueling only if the price of U_3O_8 exceeded \$120/lb. As the price of U-233 increases, the price of U_3O_8 would have to rise still further in order to obtain economic parity with slightly-enriched fueling. For example, if the price of U-233 were \$30/gm, the denatured U-233/Th cycle would be more economic than slightly-enriched fueling if the price of U_3O_8 were about \$165/lb or greater, while slightly-enriched fueling would be more economic for lower U_3O_8 prices.

Fuel cycle costs for the denatured U-233/thorium cycle would, however, be lower if a higher discharge exposure were employed. With a higher discharge exposure, the quantities of fuel which must be refabricated and reprocessed each year are lower, so that

the fuel cycle cost contribution attributable to these services would be lower. Makeup U-233 costs would increase, but as Figure 5.4.4-7 indicated, the increase in U-233 requirements does not increase rapidly with increased burnup. The effect of discharge exposure (initial enrichment) on the fuel cycle cost for the denatured U-233/Th cycle is illustrated in Figure 5.4.4-10. The fuel cycle costs illustrated in this figure are only approximate, since the detailed cycle-by-cycle mass balances have not been developed for burnup schedules other than the 14,000 MWd/t reference fuel cycle. These calculations also employ a U-233 price of \$30/gm and the long-term economic assumptions, including a U_3O_8 price of \$100/lb. As this figure illustrates, a discharge exposure of approximately 25,000 MWd/t is required in order to achieve economic parity with the slightly-enriched once-through fuel cycle in the HWR. Should the price of U_3O_8 prove higher than the \$100/lb value assumed in the analysis, or should the price of U-233 prove less than the \$30/gm assumption, economic parity with the slightly-enriched once-through fuel cycle would occur at somewhat lower burnups, while still higher discharge exposures would be necessary for lower U_3O_8 or higher U-233 prices.

5.4.4.3 Pu/Thorium Fuel Cycle

In the denatured U-235/Th and U-233/Th fuel cycles discussed previously, it has been assumed that the plutonium produced from the U-238 denaturant has been stored. An alternate would be to utilize this plutonium to fuel additional reactors--either breeder reactors or converter reactors such as the HWR. Because of the presence of plutonium, such reactors would have to be subject to stringent safeguards, perhaps collocated with the reprocessing facility at secured energy centers. Although this plutonium could be used to enrich uranium, the use of this plutonium to

enrich thorium is preferred since the bred fissile material, U-233, can be denatured and used subsequently in the denatured U-233/thorium cycle at dispersed-sited reactors. Since the bred U-233 could be denatured and used at dispersed sites, the number of Pu/Th reactors at the secured energy centers would be quite small, in contrast to the Pu/U case where the bred fissile material (plutonium) would have to be recycled within the secured energy center, thus increasing the number of reactors required to fully utilize the plutonium. The basic fuel cycle facilities required for the Pu/Th cycle are shown in Figure 5.4.4-1(c). These fuel cycle facilities are similar to those required for the previously discussed denatured U-235 and U-233 thorium fuel cycles, except that a facility must be added to fabricate plutonium-bearing thorium fuel.

The basic characteristics of the Pu/Th cycle are illustrated in Figure 5.4.4-11 where the achievable discharge burnup, net plutonium requirements, and net U-233 production are shown as a function of initial enrichment. This figure was generated from calculations of the first recycle pass, i.e., the effect of continual recycle of plutonium with the resulting buildup of higher plutonium isotopes has not been included in these calculations. As this figure illustrates, both the net requirements for fissile plutonium and the production of U-233 decreases as the discharge exposure (i.e., initial fissile plutonium enrichment) is increased. The total fissile consumption (i.e., net plutonium requirements less U-233 production) proves to be relatively insensitive to burnup, although a minimum at about 25,000 MWd/t is observed. The net fissile consumption appears to be insensitive to discharge exposure because the greater in situ utilization of U-233 (with a concomitant improvement in conversion ratio) largely compensates for the degradation in conversion ratio due to neutron losses in fission products as the discharge exposure is increased.

Figure 5.4.4-11 indicates that low discharge exposures should be employed if the objective is to maximize U-233 production and plutonium consumption. When low discharge exposures are employed, the number of reactors operating on the Pu/Th cycle is minimized, while the quantities of U-233 produced and hence available to fuel dispersed-sited reactors is maximized. On the other hand, economic considerations will favor the use of somewhat higher discharge exposures in order to minimize fuel throughput and hence the costs associated with reprocessing and fabrication.

As mentioned previously, the calculations summarized in Figure 5.4.4-11 did not consider the effects of the changing of plutonium isotopic composition with continued recycle. To establish the effects of plutonium recovery and recycle in the Pu/Th cycle, cycle-by-cycle inventories were generated for the 2.3 w/o initial fissile plutonium enrichment, which achieves a discharge burnup of 23,400 Mwd/T. The results of these calculations are summarized in Figure 5.4.4-12 where fissile plutonium requirements and U-233 production is plotted as a function of years of operation. Equilibrium cycle fissile plutonium makeup requirements are about 578 kg/GWe, in contrast to the 650 kg/GWe required if plutonium is not recycled; although the fraction of the plutonium which is fissile is rather low after the first pass through the reactor, these calculations indicate that it is still worthwhile to recycle the plutonium, since plutonium makeup requirements are reduced by about 11% with recycle. Initial core, annual, and 30-year cumulative plutonium fissile requirements are tabulated for this case in Table 5.4.4-7. Also shown in this table are the equilibrium cycle U-233 production and 30-year cumulative U-233 production. Averaged over the 30-year assumed plant lifetime, about .45 kg of U-233 is produced per kg of fissile plutonium consumed. This 30-year average U-233 production is somewhat lower than the .53 kg U-233 produced per kg fissile plutonium consumed in the equilibrium cycle because of the required initial core and excore fissile

plutonium inventory and of the delay time between spent fuel discharge and the availability of U-233 for recycle.

Fuel cycle costs for the Pu/Th cycle are compared to those of once-through fueling in Table 5.4.4-8, under both near-term and long-term economic assumptions (c.f. Table 5.4.4-3). Pu/Th fuel cycle costs will, of course, depend both upon the price at which fissile plutonium can be purchased and the price for which U-233 can be sold; hence, the fuel cycle costs shown in Table 5.4.4-8 are tabulated parametric in these parameters. With the near-term economic assumptions, Pu/Th fuel cycle costs are higher than those of the slightly-enriched once-through fuel cycle for all values of Pu and U-233 price explored. Fuel cycle costs are higher than those of slightly-enriched fueling because of the increased fuel fabrication and fuel reprocessing costs, which prove to outweigh the savings in U_3O_8 and separative work. Fuel cycle costs for the Pu/Th cycle would, of course, be lower than those for once-through uranium fueling if plutonium could be obtained for less than \$10/gm and/or the sale price of U-233 were greater than \$30/gm. Under the long-term cost assumptions, the Pu/Th fuel cycle is more economically competitive partially due to the reduction in unit fabrication and reprocessing costs, and partially due to the increase in the once through uranium fuel cycle costs as a result of the increase in U_3O_8 price to \$100/lb. Under the long-term economic assumptions, lower fuel cycle costs are realized for Pu/Th fueling for all cases except where the price of plutonium is assumed to be high (\$30/gm) and the price of U-233 low (\$10/gm).

Some further insight into the economic performance of the Pu/Th cycle can be obtained by evaluating the fuel cycle cost based upon the cost of recovered plutonium from the HWR. It has been assumed that the cost of reprocessing is \$291 or \$204/kg, for the near- and long-term economic assumptions, respectively. Since it

has been assumed that spent fuel disposal (for the once-through fuel cycle) costs \$145/kg, reprocessing will add \$146 or \$59/kg as compared to the cost of spent fuel disposal for the near-term and long-term economic assumptions, respectively. Since 187 kg/GWe-yr of fissile plutonium is produced in the slightly-enriched uranium-fueled HWR, one would have to recover a cost of 43.6 or 17.6 \$/gm fissile plutonium, for the near- or long-term economic assumptions, in order to obtain the same fuel cycle cost with reprocessing as would be obtained with spent fuel disposal. These two values for the plutonium recovery cost from the slightly-enriched HWR were used to construct the break-even U_3O_8 price as a function of U-233 sale value shown in Figure 5.4.4-13. For the near term plutonium recovery cost, the break-even U_3O_8 cost varies between \$172 and \$120/lb as the assumed sale price of U-233 varies between zero and \$50/kg. Since it is unlikely that the price of U_3O_8 will rise to above \$120/lb in the foreseeable future, the Pu-Th cycle is unlikely to be economically competitive to once-through fueling, even if U-233 could be sold for as much as \$50/gm. The break-even U_3O_8 price is reduced for the long-term economic assumptions because the plutonium price is lower (because the cost of reprocessing is decreased). With the long-term economic assumptions reprocessing and the use of plutonium in the Pu/Th cycle becomes more economic than once-through operation when the price of U_3O_8 reaches \$35-\$85/lb, depending upon whether U-233 can be sold for \$50 or \$0/gm. One can then conclude that the Pu/Th cycle is likely to become economic in the long term, provided the cost of fabrication and reprocessing can be reduced to the levels assumed in this evaluation.

Fuel cycle information for the Pu/Th cycle in the HWR can be combined with similar information for the denatured U-233 and 235 thorium fuel cycles to establish the effect of plutonium utilization on either U-233 or U_3O_8 requirements. Thirty-year cumulative U_3O_8 requirements for the denatured U-235/thorium fuel cycle are

compared in Table 5.4.4-9 for the case of plutonium storage and the case where the plutonium is utilized to fuel the Pu/Th HWR. As this table indicates, the use of plutonium reduces U_{308} requirements by about 7.0 and 3.5% for the denatured U-235/thorium cycles employing discharge exposures of 13,700 and 37,000 MWd/t, respectively. The benefits from plutonium utilization are smaller for the high burnup case since the annual discharge fissile plutonium is lower (cf. Table 5.4.4-1). Table 5.4.4-9 also gives the number of Pu/Th-fueled reactors required to utilize the plutonium produced from 50 denatured U-235/thorium dispersed-sited reactors; about two plutonium-fueled reactors are required to consume the plutonium produced by 50 denatured thorium reactors operating with a 13,700 MWd/t discharge exposure, while one plutonium-fueled HWR will serve 50 denatured thorium reactors operating with a 37,000 MWd/t discharge exposure.

Table 5.4.4-10 provides similar information for the denatured U-233/thorium cycle. Use of the plutonium in a Pu/Th-fueled HWR would reduce U-233 requirements by about 7.8%. Approximately 2 Pu/Th reactors would be required to utilize the plutonium produced by 50 denatured U-233/thorium-fueled reactors.

Figure 5.4.4-1

FUEL CYCLE FACILITIES FOR THORIUM FUELING

A. U-235/Th CYCLE (13,700 MWD/T DISCHARGE EXPOSURE)

B. U-233/Th CYCLE (14,000 MWD/T DISCHARGE EXPOSURE)

Figure 5.4.4-1

FUEL CYCLE FACILITIES FOR THORIUM FUELING

(Cont'd)

C. Pu/Th CYCLE (23,400 MWD/T DISCHARGE EXPOSURE)

Figure 5.4.4-2

DISCHARGE BURNUP AS A FUNCTION OF INITIAL ENRICHMENT IN THE DENATURED U-235/Th CYCLE

Figure 5.4.4-3
FUEL CYCLE COST AS A FUNCTION
OF U_3O_8 PRICE

Figure 5.4.4.4

EFFECT OF REFABRICATION COST
ON FUEL CYCLE COST

Figure 5.4.4-5
ANNUAL U-233 AND NET FISSILE REQUIREMENTS
AS A FUNCTION OF INITIAL ENRICHMENT

Figure 5.4.4-6
FISSILE Pu CONTENT OF SPENT DENATURED
U-233/Th FUEL

Figure 5.4.4-7

U-233 REQUIREMENTS (30-YEAR) AS A FUNCTION
OF INITIAL ENRICHMENT

Figure 5.4.4-8

FUEL CYCLE COST FOR THE DENATURED
U-233/Th CYCLE AS A FUNCTION
OF U-233 PRICE

Figure 5.4.4-9

BREAKEVEN U_3O_8 PRICE AS A FUNCTION
OF U-233 COST

Figure 5.4.4-10

FUEL CYCLE COST AS A FUNCTION OF DISCHARGE EXPOSURE
IN THE DENATURED U-233/Th CYCLE

Figure 5.4.4-11

FISSILE Pu REQUIREMENTS AND U-233 PRODUCTION
AS A FUNCTION OF BURNUP IN THE Pu/Th CYCLE
(EQUILIBRIUM CYCLE, FIRST GENERATION Pu)

Figure 5.4.4-12

FISSILE CHARGE/DISCHARGE FOR THE Pu/Th CYCLE

Figure 5.4.4-13

BREAKEVEN U_3O_8 PRICE AS A FUNCTION
OF U-233 PRICE FOR THE Pu/Th CYCLE

TABLE 5.4.4-1
FUEL CYCLE INFORMATION
DENATURED U-235/THORIUM CYCLE

	<u>Denatured U-235/Thorium</u>		<u>1.2 w/o UO₂ Cycle</u>
Discharge Exposure, Mwd/MT	13,700	37,000	19,750
Fraction of Core Replaced, %/yr	53.2	19.7	33.6
Core Loading, MT	151.1	151.1	166.1
Fuel Throughput, MT/yr	80.44	29.81	55.85
Fissile Losses, %			
Conversion	0.5	0.5	0.5
Fabrication	1.0	1.0	1.0
Reprocessing	1.0	1.0	NA
Ex-core Time, yrs	2.0	2.0	NA
Requirements for U ₃ O ₈ (ST/GWe)*			
Initial Core	611.8	764.7	172.6
Annual Equilibrium Reload	41.7	52.9	114.2
30-Year Cumulative	2464	2503	3546
30-Year Cumulative, Net**	1250	1587	3426
SWU Requirement (MT SWU/GWe)*			
Initial Core	556.9	696.2	0
Annual Equilibrium Reload	39.9	51.8	31.4
30-Year Cumulative	2273	2358	927.3
30-Year Cumulative, Net**	1198	1555	942.0
Annual Fissile Pu discharge, kg/GWe	22.6	12.0	149

*0.2 w/o tails, 75% capacity factor

**30-year cumulative, Net requirements are 30-year cumulative requirements less credit for incore and excore fissile inventories at the end of the assumed 30-year reactor lifetime

TABLE 5.4.4-2
EFFECT OF EXCORE TIME AND LOSSES ON U_3O_8 REQUIREMENT

	30-Year Cumulative U_3O_8 (ST/GWe)	
Burnup, Mwd/MT	<u>13,700</u>	<u>37,000</u>
U_3O_8 Requirements, Standard Loss Assumptions	2464	2503
Losses		
Reprocessing (1%)	78.0	24.5
Fabrication and Conversion (1.5%)	<u>35.1</u>	<u>38.6</u>
Subtotal	113.1	63.1
Ex-core Inventory (2 yr ex-core time)	<u>568.0</u>	<u>195.9</u>
Total	681.1	259.0
U_3O_8 Requirements, No losses, zero ex-core time	1783	2244
U_3O_8 Requirements, Alternate Assumptions:	2129	2380
Reprocessing loss - 0.5%		
Fabrication loss - 0.5%		
Conversion loss - 0.5%		
Excore Time - 1 yr.		

TABLE 5.4.4-3
ECONOMIC ASSUMPTIONS FOR THORIUM
FUELING

	<u>Near-Term</u>	<u>Long-Term</u>
U ₃ O ₈ , \$/lb	40	100
Conversion, \$/kg	4	4
Separative Work, \$/SWU	100	100
Fabrication, \$/kg		
Denatured U-235/ThO ₂	99	80
Denatured U-233/ThO ₂	567	487
Pu/ThO ₂	370	318
Reprocessing of ThO ₂ , \$/kg	291	204
Thorium, \$/kg	33	33
Present Worth (Discount) Rate, %	4.525	4.525
Fixed Charge Rate, %	10.0	10.0
Carrying Charge Rate, %	8.0	8.0
Tail Assay, w/o	0.2	0.2
Capacity Factor, %	75	75

TABLE 5.4.4-4

**FUEL CYCLE COSTS FOR THE DENATURED U-235/Th FUEL CYCLE
(MILLS/KW-HR)**

DISCHARGE BURNUP, Mwd/kg	DENATURED U-235/Th				1.2 W/O ONCE-THROUGH URANIUM CYCLE	
	13.7		37.0		19.75	
	NEAR TERM	LONG TERM	NEAR TERM	LONG TERM	NEAR TERM	LONG TERM
FABRICATION	4.57	3.91	1.50	1.28	0.52	0.52
FUEL REPROCESSING/DISPOSAL	2.80	1.97	1.02	0.72	0.99	0.99
URANIUM	1.15	2.81	1.06	2.59	1.45	3.54
SWU	1.30	1.30	1.19	1.19	0.43	0.43
THORIUM	0.38	0.38	0.12	0.12	—	—
CARRYING CHARGES	1.04	1.69	0.94	1.55	0.34	0.76
D ₂ O	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>
TOTAL FUEL CYCLE COST	13.28	14.10	7.87	9.49	5.77	8.28

TABLE 5.4.4-5
FUEL CYCLE INFORMATION FOR THE DENATURED U-233/THORIUM CYCLE

	<u>Denatured U-233/Thorium</u>
Discharge Exposure, Mwd/MT	14,000
Fraction of Core Replaced, %/yr	52.1
Core Loading, MT	151.1
Fuel Throughput, MT/yr	78.7
Fissile Losses, %	
Conversion	NA
Fabrication	1.0
Reprocessing	1.0
Ex-core Time, yrs	2.0
Requirements for U-233, kg/GWe*	
Initial Core	1877.7
Annual	109.7
30-Year Cumulative	6849
30-Year Cumulative, Net**	3291
Fissile Pu Production, kg/GWe*	
Annual	24.7
30-Year Cumulative	667

*75% Capacity Factor

**30-year cumulative, net requirements are 30-year cumulative requirements less credit for incore and excore fissile inventories at the end of the assumed 30-year reactor lifetime

TABLE 5.4.4-6
FUEL CYCLE COSTS FOR THE DENATURED (14,000 MWD/TEM DISCHARGE EXPOSURE)
U-233/Th CYCLE (Mills/kw-hr)

	Denatured U-233/Th Cycle		1.2 w/o Once-Through Uranium Cycle	
	<u>Near Term</u>	<u>Long Term</u>	<u>Near Term</u>	<u>Long Term</u>
Fabrication	5.37	4.60	0.52	0.52
Fuel Reprocessing/Disposal	2.73	1.92	0.99	0.99
Uranium	-	-	1.45	3.54
SWU	-	-	0.43	0.43
Thorium	0.37	0.37	-	-
Carrying Charges	0.24	0.26	0.34	0.76
D ₂ O	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>	<u>2.04</u>
Subtotal	10.75	9.19	5.77	8.28
U-233 at 10 \$/gm	0.64	0.64	-	-
U-233 at 30 \$/gm	1.93	1.93	-	-
Total Fuel Cycle Cost*				
U-233 @ 10 \$/gm	11.39	9.83	5.77	8.28
U-233 @ 30 \$/gm	12.68	11.12	5.77	8.28

*No credit has been taken for the value of the fissile plutonium produced. Fuel cycle costs would be reduced by 0.04 or 0.11 mills/kw-hr if fissile plutonium could be sold at 10 or 30 \$/gm respectively.

TABLE 5.4.4-7
FUEL CYCLE INFORMATION FOR THE Pu/Th CYCLE

Discharge Exposure, Mwd/MT	23,400
Fraction of Core Replaced, %/yr	31.2
Core Loading, MT	151.1
Fuel Throughput, MT/yr	47.1
Fissile Losses, %	
Fabrication	1.0
Reprocessing	1.0
Ex-core Time, yrs	2.0
Fissile Pu Requirements	
Initial Core, kg/GWe	2827
Initial Core, w/o	2.3
Annual (Equilibrium), kg/GWe	578
Annual (Equilibrium), w/o	2.8
30-Year Cumulative, kg/GWe	21291
U-233 Production, kg/GWe	
Annual (Equilibrium)	308
30-Year Cumulative	9209

TABLE 5.4.4-8
FUEL CYCLE COSTS FOR THE Pu/Th CYCLE
(MILLS/KW-HR)

	Pu/Th CYCLE								1.2 W/O ONCE-THROUGH URANIUM CYCLE	
	NEAR TERM				LONG TERM				NEAR TERM	LONG TERM
Pu PRICE, \$/gm FISSILE U-233 PRICE, \$/gm	10 10	10 30	30 10	30 30	10 10	10 30	30 10	30 30	NA	
FABRICATION	2.04	2.04	2.04	2.04	1.74	1.74	1.74	1.74	0.52	0.52
FUEL REPROCESSING/ DISPOSAL	1.55	1.55	1.55	1.55	1.09	1.09	1.09	1.09	0.99	0.99
URANIUM	—	—	—	—	—	—	—	—	1.45	3.54
SWU	—	—	—	—	—	—	—	—	0.43	0.43
THORIUM	0.27	0.27	0.27	0.27	0.27	0.27	0.27	0.27	—	—
PLUTONIUM	0.92	0.92	2.77	2.77	0.92	0.92	2.77	2.77	—	—
U-233 CREDIT	-0.53	-1.57	-0.53	-1.57	-0.53	-1.57	-0.53	-1.57	—	—
CARRYING CHARGES	0.65	0.83	1.58	1.76	0.67	0.86	1.60	1.79	0.34	0.76
D ₂ O	2.04	2.04	2.04	2.04	2.04	2.04	2.04	2.04	2.04	2.04
TOTAL FUEL CYCLE COST	6.94	6.08	9.72	8.86	6.20	5.35	8.98	8.13	5.77	8.28

TABLE 5.4.4-9
EFFECT OF Pu UTILIZATION ON U_3O_8 REQUIREMENTS
OF THE DENATURED U-235/Th CYCLE

Burnup, Mwd/kg	<u>13.7</u>	<u>37.0</u>
30-Year Cumulative U_3O_8 Requirements, ST/GWe ^{1,2}		
- Pu Storage	2464	2503
- Pu use in Pu/Th fueled HWR ³	2296	2414
Pu/Th reactors per U-235/Th reactors	1.7/50	0.9/50

TABLE 5.5.4-10
EFFECT OF Pu UTILIZATION OF U-233 REQUIREMENTS
OF THE DENATURED U-233/Th CYCLE

Burnup, Mwd/kg	<u>14.0</u>
30-Year Cumulative U-233 Requirements, ST/GWe ¹	
- Pu Storage	6849
- Pu use in Pu/Th fueled HWR ³	6312
Pu/Th reactors per U-235/Th reactors	1.9/50

-
1. 75% Capacity factor
 2. 0.2% w/o Tails composition
 3. 2.0 w/o fissile Pu initial enrichment, 23.4 Mwd/kg discharge exposure

Fuel management calculations were performed to characterize the U_3O_8 and/or other fissile material requirements, separative work requirements, and fuel cycle and net power cost economic performance for the 1.2 w/o enriched uranium fuel cycle, the uranium fuel cycle with plutonium recycle, the denatured U-235/thorium cycle, the denatured U-233/thorium cycle, and the plutonium-enriched thorium fuel cycle. The more significant findings of this evaluation are summarized below.

The slightly-enriched uranium once-through fuel cycle, the uranium fuel cycle with plutonium recycle, and the U-235/denatured thorium fuel cycle all require the mining of U_3O_8 to provide the initial and makeup fissile requirements. The uranium utilization of each of these fuel cycles is summarized in Table 5.4.5-1. Also included in this table are values for the standard natural uranium fueled HWR and for the standard PWR for comparison. It is noted that evaluations of the natural uranium fueled HWR and of the LWR have not been part of this evaluation; values for these two reactor types may vary somewhat, and in particular, improvements are currently under development which should serve to reduce the U_3O_8 requirements of the LWR from that shown in the table. As the table indicates, approximately 3,550 ST U_3O_8 are required to fuel the slightly-enriched HWR for 30 years of operation, compared to about 5,000 ST for the natural uranium HWR and 6,000 ST for the PWR; thus, the U_3O_8 requirements for the natural uranium fueled HWR are about 40% greater than that of the slightly enriched HWR, while the U_3O_8 requirements for the LWR are approximately 70% greater. Further reductions in U_3O_8 requirements can be realized through the recycle of plutonium (the U_3O_8 requirements are about 75% of that of the slightly-enriched HWR), or by the use of the denatured U-235/thorium cycle where U_3O_8 requirements are about 70% of those of the slightly-enriched HWR. The uranium requirements

for the U-235/thorium cycle can be reduced slightly (4% additional reduction) through the use of the plutonium recovered from the denatured spent fuel in the Pu/Th fuel cycle.

Separative work requirements for the slightly-enriched HWR are about 1/4 those of the LWR. The use of denatured thorium fueling increases separative work requirements by about a factor of 2.5. The annual discharge of fissile plutonium in spent fuel from the slightly-enriched HWR is somewhat lower than that discharged from the light-water reactor, and is more than a factor of two smaller than that discharged from the natural uranium-fueled HWR. The quantity of fissile plutonium discharged in the U-235/thorium fuel cycle is about 10% that of the slightly-enriched once-through fuel cycle.

Power costs are summarized in Table 5.4.5-2. Power costs are provided for near-term economic assumptions, which are characterized primarily by a U_3O_8 price of \$40/lb, and for long-term economic assumptions, which are characterized by a U_3O_8 price of \$100/lb. The cost of reprocessing and of recycle fuel fabrication are also somewhat lower under the long-term economic assumptions. The slightly-enriched HWR is seen to have somewhat higher power costs than that of the standard LWR under the near-term economic assumptions. However, the power costs of the slightly-enriched HWR are less sensitive to the price of U_3O_8 than are those of the LWR, and for the long-term economic assumptions HWR power costs are lower than those of the standard LWR. Once-through fueling is seen to be more economic than either the uranium cycle with plutonium recycle or the denatured U-235/Th fuel cycle for both the near and long-term economic assumptions. The same conclusion appears to hold for the denatured U-233/thorium fuel cycle, although there is some potential for this fuel cycle to be economically competitive with once-through fueling if higher discharge burnups were employed.

Power costs for this fueling mode are also dependent upon the price for which U-233 can be purchased; in this tabulation, a U-233 price of \$30/gm has been assumed. The last fuel cycle listed on Table 5.4.5-2 cannot be viewed in isolation, since it utilizes plutonium produced from the slightly-enriched HWR, and hence is dependent upon the cost of plutonium recovery from this system. These calculations do, however, indicate that it is more expensive to recover plutonium and utilize it in the Pu/Th fuel cycle under the near-term economic assumptions, but that under the long-term economic assumptions reprocessing of spent slightly enriched HWR fuel to recover plutonium and the subsequent use of plutonium in the Pu/Th fuel cycle is more economic than operating the slightly-enriched HWR in the once-through mode.

Under the reference economic assumptions employed in the power cost analysis, slightly enriched fueling in the HWR does not become economically competitive with the once-through fuel cycle in the LWR until the price of uranium rises to about \$55/lb. This break-even U_3O_8 price, however, is very sensitive to a number of economic parameters. It is most sensitive to the capital cost differential between the HWR and LWR (an evaluated cost differential of 8.3% was used in the analysis) to the difference in capacity factor assumed for the HWR and LWR (HWR and LWR capacity factors of 77.5 and 75%, respectively, were used), and to the cost of heavy water (\$213/kg was assumed).

The possibility of employing batch refueling, in place of on-line refueling, was briefly evaluated and found to be unattractive. With an annual refueling interval, the uranium requirements of the slightly-enriched HWR are increased by about 30%, or to a value roughly comparable to that which may be possible in the LWR with the implementation of various methods for improving uranium utilization. For the natural uranium-fueled HWR, annual refueling is impossible. If semiannual refueling is employed, U_3O_8 requirements

are increased by about 16% for the slightly-enriched case, while the uranium requirements for the natural uranium fueled system are nearly doubled. In addition, major design modifications would be required in order to perform batch fueling in a fashion which would provide a significant increase in security against diversion as compared to on-line refueling.

The use of plutonium recycle in the HWR results in a significant reduction in U_3O_8 requirements; U_3O_8 requirements are approximately 25% lower than those of the slightly enriched once-through HWR. Although it is anticipated that plutonium recycle could be performed in the conceptual HWR design without modifications, this refueling mode does not appear to be economically attractive because of the high fuel cycle cost contributions from mixed-oxide fuel fabrication and fuel reprocessing. The economic analyses indicate that the price of U_3O_8 would have to rise to approximately \$200/lb before the plutonium recycle mode could become more economic than slightly-enriched once-through fueling.

Significant reductions in U_3O_8 requirements can also be realized by employing the denatured U-235/Th fuel cycle; U_3O_8 requirements are about 70% of that required for the slightly-enriched once-through fuel cycle. Under the reference fuel cycle assumptions (a total of 2.5% fissile material losses during conversion, fabrication and reprocessing, and a two-year ex-core time), uranium utilization is insensitive to discharge exposure, although lower discharge exposures will result in improved uranium utilization if losses are minimized and the ex-core time reduced to minimize the ex-core inventory of fissile materials. The economic performance of the denatured U-235/thorium cycle is, however, very sensitive to discharge exposure. For low discharge exposures, the denatured U-235/thorium cycle is unlikely to be economically competitive with slightly-enriched once-through fueling for any reasonable U_3O_8 price assumption. The economic competitiveness of the

denatured thorium fuel cycle is improved for the high burnup case, since the lower annual throughput of fuel results in decreased fabrication and reprocessing costs. However, even for a discharge burnup of 37,000 MWd/t, the price of uranium would have to increase to about \$180/lb before the denatured thorium fuel cycle would be competitive with slightly-enriched once-through uranium fueling. One can conclude, therefore, that the denatured U-235/thorium fuel cycle is unlikely to prove economic unless it is possible to develop less expensive refabrication and/or reprocessing methods.

It also appears that the denatured U-233/thorium fuel cycle (with a 14,000 MWd/t discharge exposure) is unlikely to prove economically attractive, since the fuel cycle cost for this refueling mode is about 1.5 mills/kw-hr greater than that of the uranium once-through fuel cycle (\$100/lb U_3O_8), even assuming a rather low, \$10/gm, U-233 price. In order to be economically competitive with the once-through fuel cycle, high discharge exposures, in excess of 25,000 MWd/t, would have to be employed.

The use of fissile plutonium in the Pu/Th fuel cycle appears to be more economically attractive than the recycle of plutonium into uranium fuel. The Pu/Th fueling mode appears economically preferable to the once-through fuel cycle if the price of uranium is greater than \$140-172/lb for near-term economic assumptions, and \$53-84/lb for the long-term economic assumptions. The ranges here depend upon the price at which the recovered U-233 can be sold, with the higher U_3O_8 prices corresponding to zero U-233 value and the lower U_3O_8 prices to a U-233 value of \$30/gm.

As mentioned above, the thorium fuel cycles do not appear to be economically competitive unless rather high discharge exposures are employed (30,000 MWd/t or greater). Modifications to the fuel design, and perhaps some further modifications to the reactor design, will be required to employ such high discharge exposures.

The use of thorium fueling may also place increased demands on the reactor control system, and may result in some restrictions in the ability to maneuver. It is recommended that more detailed evaluations of the design changes which may be necessitated by the higher discharge exposure and of the implications of thorium fueling on reactor and control system design and upon reactor operational performance be undertaken before this mode of fueling is seriously considered.

TABLE 5.4.5-2
POWER COST SUMMARY

	Mills/kw-hr ¹⁾	
	<u>Near Term²⁾</u>	<u>Long Term²⁾</u>
Standard LWR	18.71	23.10
SEU HWR - 1.2 w/o (Once Through)	19.17	21.68
HWR with Pu Recycle	22.06	22.85
HWR with Denatured U-235/Th Fueling		
13,700 Mwd/MT	26.68	27.50
37,000 Mwd/MT	21.27	22.89
HWR with Denatured U-233/Th Fueling ³⁾		
14,000 Mwd/MT	26.08	24.52
HWR with Pu/Th Fueling ⁶⁾		
23,400 Mwd/MT	24.15 ^{3,4)}	19.75 ^{3,5)}

1) 75% capacity factor for LWR, 77.5% for HWR

2) Near and long term economic assumption characterized by 40 and 100 \$/lb U₃O₈, respectively; reprocessing and refabrication costs are also reduced in the long term cost assumption

3) Assumes 30 \$/gm U-233

4) Assumes Pu recovery cost of 44 \$/gm

5) Assumes Pu recovery cost of 17 \$/gm

6) Fuel cycle costs are based upon cost of Pu recovery from SEU HWR

TABLE 5.4.5-1
RESOURCE UTILIZATION SUMMARY

	U ₃ O ₈ Requirements		SWR Requirements		Annual	
	30-Year (ST/GWe)	Relative to SEU HWR	30-Year (MT/GWe)	Relative to SEU HWR	Pu Production kg/GWe	Relative to SEU HWR
Standard PWR	6000	1.71	3600	3.88	176	0.94
HWR - Natural U Fueling (Once-Through)	5000	1.41	0	0	350	1.87
SEU HWR - 1.2 w/o (Once-Through)	3550	1.00	927	1.0	187	1.0
HWR With Pu Recycle	2670	0.75	0	0	-	-
HWR, Denatured U-235/Th, Pu Storage						
13,700 MWD/MT	2460	0.69	2270	2.45	22.6	0.12
37,000 MWD/MT	2500	0.71	2360	2.55	12.0	0.06
HWR, Denatured U-235/Th with Pu use in the Pu/Th cycle						
13,000 MWD/MT	2300	0.65	2200	2.37	-	-
34,000 MWD/MT	2410	0.68	2280	2.46	-	-

References

- 5.4-1 "NASAP Provisional Data Base for the Second Rolling Report", DOE-FCE, July 21, 1978.
- 5.4-2 D. R. Haffner, et. al., "Thorium Assessment Program System Studies", Hanford Engineering Development Laboratory, TC-1064, February 1978.
- 5.4-3 "Improvement of Availability of PWR Nuclear Plants Through the Reduction of the Time Required for Refueling/Maintenance Outages", COO-2962-5/WCAP-9215, April 1978.
- 5.4-5 M. F. Duret, "The Use of Plutonium in CANDU Reactors", AECL, June 1976.
- 5.4-6 E. Critoph et. al., "Prospects for Self-Sufficient Equilibrium Thorium Cycles in CANDU Reactors", ANS 1975 Winter Meeting, San Francisco, November 16-21, 1975.
- 5.4-7 "Comparison of Ontario Hydro with World Power Reactors 1978," Ontario Hydro.

5.5 Safety and Licensing Assessment of the HWR

5.5.1 Introduction

A particular reactor plant can be truly said to be "licensable" in a particular country only after an extensive review of reactor design and safety has been performed by the appropriate licensing agency, and approval for construction has been obtained. This review process requires the preparation of safety analysis reports (SAR) by the applicant in which the reactor, its containment system, and other associated systems are described in detail and in which the consequences of various postulated transients and accidents are detailed and compared against appropriate acceptance criteria.

The preparation of such a detailed SAR for a large capacity U.S. sited HWR has yet to be pursued, nor has the NRC had the opportunity to review HWR design in any great detail. Although the preparation and subsequent NRC review of an SAR for the HWR would be instrumental in establishing the licensability of the concept and of identifying design changes required for U.S. siting, the effort to prepare this document is large and beyond the scope of present evaluation. This section is limited to a qualitative discussion of safety and licensing issues with the intent of establishing the prospects for HWR licensing in the U.S. and of identifying areas where additional research and development is likely to be required to resolve potential safety and licensing issues.

The conceptual HWR design described in previous sections of this chapter is based upon the CANDU-PHWR. A number of changes, however, have been made to the basic CANDU-PHWR design, the most notable of which are the use of slightly-enriched fueling, the increase in primary and secondary system pressures, and the increase in net electrical generating capacity. In viewing the

safety and licensability of the HWR, it is useful to separate the question of licensability of the CANDU reactor as currently deployed in Canada from the question of the licensability of the various design changes to the CANDU-PHWR which have been incorporated into the reference conceptual design.

Reactors of the standard CANDU-PHWR design are, of course, licensed and operating in Canada as well as in several other nations. The extensive analyses performed for the CANDU-PHWR by the Canadians, the review and acceptance of these analyses by Canadian licensing authorities, and the highly satisfactory performance of CANDU reactors is a testimony to the overall inherent safety and licensability of the concept. However, it must be recognized that Canadian licensing requirements to a large degree have developed independently from those in the U.S. for the LWR, and hence some changes to HWR design may be necessary to conform to U.S. licensing traditions and construction practices.

In viewing safety and accident considerations, it is useful to distinguish between inherent safety features and the engineered features of plant design. The inherent safety features include such aspects as core transient performance (e.g., coefficients of reactivity) and fundamental design characteristics, such as the use of pressure tubes, which are very difficult and/or costly to alter by design changes. The engineered features of the plant are more readily altered to meet various licensing and construction criteria.

A review of the engineered features of the CANDU pressurized heavy water reactor was performed by United Engineers and Constructors for the Department of Energy (Ref. 5.5-1). This evaluation identified a number of changes to the engineered design of the CANDU reactor which were suggested to conform to U.S. construction practices and the strict application of appropriate U.S. licensing

criteria. These engineering changes were subsequently incorporated into a conceptual large plant design, and a capital cost for a plant incorporating these changes was developed (Ref. 5.5-2). These evaluations indicated no fundamental problem with modifying the engineered features of the CANDU system to conform to U.S. criteria, although the recommended design changes do have a significant impact on plant capital cost. The reference conceptual design, discussed in detail in Section 5.1, used the UE&C studies as a point of departure. The reference conceptual design therefore includes those changes in engineered design features which are thought necessary to attain conformance with U.S. licensing criteria, and the capital cost of the conceptual HWR developed in Section 5.3 fully reflects the cost of these design modifications.

When considering the inherent safety features of the HWR concept, it should be recognized that in most instances a particular concept will have certain features which are more satisfactory and certain features which are less satisfactory than other alternatives. Of prime concern is the overall implications of safety when all the unique attributes of the system, both positive and negative, are considered. It is also appropriate to note that the acceptability of various inherent features of a reactor concept is typically viewed with respect to licensing criteria and design codes. In the U.S., these licensing criteria and design codes have evolved about the light water reactor, and consequently it should be expected that these licensing criteria and design codes are not fully appropriate to the HWR or sufficiently comprehensive to encompass all aspects of HWR design. A review of licensing issues associated with a U.S.-sited CANDU plant, emphasizing the inherent safety features of the concept, was performed in Reference 5.5-3. This report concluded that relatively few modifications to the CANDU reactor would be required to obtain conformance with the intent of U.S. licensing criteria and design codes. Emphasis in this report was placed on evaluating the CANDU design with respect to the intent of U.S. criteria and

design codes, rather than upon a strict interpretation of these codes, in recognition of the aforementioned fact that these criteria and codes have evolved for the light water reactor and hence are not totally appropriate to the HWR concept. Lastly, the HWR concept was the subject of a brief review by the U.S. NRC (Reference 5.5-4). This review, while identifying a number of areas which will require additional and more detailed evaluation, concluded that it was likely that a suitably designed HWR could be licensed in the U.S.

The following sections provide a qualitative assessment of the safety and licensing impact of the conceptual HWR design. Since detailed safety analysis has not been performed for this evaluation the description and assessment of events are based primarily on the CANDU-PHWR, with emphasis placed on the particular design changes incorporated into the conceptual HWR design. The core physics parameters for slightly enriched and thorium fueling are reviewed to establish whether or not these fueling modes could lead to significant changes in core transient performance, as compared to natural uranium fueling which constitutes the design basis for the conventional CANDU-PHWR. Also, presented in the following is a containment evaluation, a discussion of environmental impact, and an assessment of proliferation risks and safeguards.

5.5.2 Core Related Safety/Licensing Aspects of CANDU

The purpose of this section is to address the CANDU safety/licensing issues involved with fuel thermal limits and core power ratings. In order to provide background information, a general overview of the Canadian licensing approach is provided along with a brief listing of the significant inherent characteristics of CANDU which are relevant. Next, the safety related design basis events to be considered are presented along with the categorization, acceptance criteria, protective actions, and initial conditions. A comparison of the U.S. and Canadian approaches is provided.

5.5.2.1 General Approach

The general approach to licensing in Canada is quite different from that used in the U.S. The Canadian approach, summarized in Table 5.5.2-1, requires that each system in the plant be categorized as either a process or a safety system. Process systems are those required for normal operation. Safety systems are provided to limit the release of radioactivity if a failure should occur in a process system. These process and safety systems are separate and independent as far as practicable. Two types of failures must be considered - single and dual failures. A single failure is a failure in the process system and a dual failure is a failure in the process system combined with the failure of one of the following safety systems to perform as designed: Shutdown System 1, Shutdown System 2, Emergency Core Cooling (ECC) System, or Containment System (various degrees of containment impairment are postulated, based upon the loss of containment subsystems). In evaluating single failures, credit is taken for the operation of only one of the two shutdown systems. It is also a requirement that single failures which necessitate safety system action occur no more frequently than once every three years, and that dual failures occur no more often than once every thousand years. Separate radiation release acceptance criteria are specified for each of these types of failures.

In comparison, the U.S. approach involves the generation of a list of safety related design basis events categorized on the basis of probability of occurrence. Detailed acceptance criteria are provided for each category with the more probable events having the more restrictive acceptance criteria. In general, only those events which the Canadians classify as single failures are addressed by the U.S. licensing approach.

5.5.2.2 Inherent Characteristics

There are several significant differences in the inherent characteristics between a PWR and a HWR which must be considered when evaluating the core-related safety/licensability. The inherent characteristics of a reactor are the design features and core transient performance which are fundamental to the design of the reactor and which are very difficult and/or costly to alter by design changes. When considering the inherent safety characteristics of a concept, it should be recognized that in most instances a particular concept will have features which are both more and less satisfactory than other alternatives. Of prime concern is the overall implication to safety and licensability when all the unique attributes of the system, both positive and negative, are considered. Some of the more important inherent characteristics of the HWR are summarized below:

- ° Pressure Tube Configuration

In contrast to pressurized light-water reactors which employ a massive steel vessel to contain the primary system pressure, zirconium alloy pressure tubes (4-6 mm, depending on system pressure) serve this function in HWRs of the CANDU type. The use of the pressure tube configuration in the HWR has both positive and negative safety implications.

The pressure vessels employed in pressurized light water reactors are external to the active core region of the reactor, and hence the thickness of this vessel can be chosen without affecting core nuclear performance. The thickness of these vessels consequently has been selected to provide wide margins to failure, sufficiently large that the probability of failure of these massive vessels is almost universally viewed to be insignificant. Like the pressure

vessel of the PWR, the pressure tubes of the HWR are subject to the full stress of the primary coolant pressure. Since these pressure tubes reside in the active region of the core, they affect core nuclear performance, and hence their thickness is carefully selected to minimize parasitic neutron losses while providing adequate strength to contain the primary system pressure and to accommodate changes in pressure tube material properties which occur as a result of irradiation. Minor "failures" of pressure tubes have occurred (i.e., leaking of pressure tubes has occurred). The possibility and consequences of pressure-tube failure must consequently be considered; to be addressed is the degree of failure which can credibly be postulated (cracks which are limited to minor coolant leakage or catastrophic failure), the consequences of a pressure-tube failure, and the probability that a pressure tube failure will lead to tube-to-tube failure propagation.

In assessing the potential for pressure tube failure, it is important to recognize that the pressure tube wall thickness is much smaller than the critical crack size which can result in catastrophic failure of the pressure tube. Prior to pressure tube installation and core operation, pressure tubes would be 100% inspected to ensure that no detectable faults are present in the pressure tubes; faults which may be postulated to still be present but too small to be detected by inspection will consequently be much smaller than the critical crack size. In-service inspection at appropriate intervals will be performed to ensure that larger size faults do not develop.

Under these circumstances, crack propagation, should it occur, would result in a penetration of the pressure tube wall and coolant leakage, rather than a catastrophic pressure-tube failure ("leak before break"). A leak of a pressure tube is readily detectable by means of the surveillance

system provided to sample the gas contained in the annular space between pressure tubes and calandria tubes, providing warning of more serious pressure-tube failure and providing ample time for corrective action. Thus, by ensuring that the pressure tubes are free of faults larger than the critical crack size, the risk of catastrophic pressure tube failure, along with the possibility of tube-to-tube failure propagation which might be argued could accompany such catastrophic pressure tube failure is reduced to insignificant levels.

Extensive development programs have been performed in Canada to establish the fracture mechanics of thin-walled tubing, and the critical crack length under reactor service conditions has been experimentally determined as has the effect of irradiation on the size of the critical crack (Ref. 5.5-13). The fact that pressure tubes will exhibit small leaks, rather than catastrophic failures, and that these leaks can be detected by the annulus gas monitoring system was demonstrated in Pickering units where cracking occurred in the vicinity of the rolled joints in a number of pressure tubes (Ref. 5.5-14). Coolant leakage from these cracks was detected by the gas monitoring system, allowing leaking pressure tubes to be replaced without compromising the safety of the unit. Experiments have also been performed which demonstrate that pressure tube failure propagation will not occur even in the highly unlikely event of a catastrophic pressure tube failure (Ref. 5.5-9, 5.5-15).

In summary, significant breaches of the pressure boundary as a result of pressure tube failure are not anticipated since pressure tubes will leak before a break occurs and such leaks are detectable by the surveillance system, as has been demonstrated on operating units. Furthermore, even in the unlikely event of a major pressure tube rupture (or even the rupture of two pressure tubes) consequences are acceptable (the major rupture of a pressure tube is addressed in Canadian

Safety evaluations). Experiments have demonstrated that pressure tube failure propagation (i.e. the sequential failure of a number of pressure tubes) will not occur even in the unlikely event of a major pressure tube rupture.

A second pressure-tube-related feature of HWRs of the CANDU type is the use of a rolled joint to connect the Zr-Nb pressure tube to the steel end fittings. The use of these rolled joints have proven successful in operating CANDU reactors, and incipient failure of these rolled joints would also be detectable by the annulus gas monitoring system mentioned above. Although the pressure tubes are designed to serve for the full lifetime of the plant, they can be replaced relatively easy if necessary. Thus, the safety of the pressure tube reactor is enhanced by the ability to replace tubes should in-service inspection show signs of developing pressure tube or rolled joint faults. It might also be noted that there are currently no U.S. codes (ASME) which allow the use of Zr-Nb and rolled joints as the pressure boundary (since this material and the rolled joint are not employed in U.S. reactors, there has been no need to develop such codes). There appears to be no reason why such codes cannot be developed and approved, although it is recognized that considerable time and effort will be necessary to obtain such code changes.

The pressure tube configuration employed in CANDU-type reactors also has important implications to the loss-of-coolant accident. The use of the pressure tube configuration, coupled with the injection of emergency core cooling water into both the inlet and outlet headers, facilitates the supply of coolant water to the fuel channels in the event of a loss-of-coolant accident. The configuration of individual fuel channels tends to facilitate coolant delivery to all core locations. As discussed in greater detail in Section 5.5.5, the ability to deliver coolant to all core locations

enhances the performance of the ECC system and results in peak clad temperatures during the reflood phase which are significantly lower than those which occur earlier in the transient. As a result, consequences of a loss of coolant accident are much less dependent on ECCS performance than is the case for PWRs. The pressure tube configuration also allows the fuel channels to be separated into two independent loops, so that in the event of a primary coolant pipe rupture only one-half of the core would be subject to loss of coolant. Based on these characteristics, the performance of the HWR during the loss-of-coolant accident could be judged to be inherently superior to that of the light water reactor.

° Independent Moderator System

Reactors of the Candu type employ a separate heavy water moderator which is independent of the heavy water coolant which flows through the fuel channels. The moderator operates at relatively low temperatures (150-200°F) and at near atmospheric pressures.

The presence of large quantities of relatively cool moderator surrounding the fuel channels provides a backup heat sink in the event of failure of the ECC system. Canadian analyses indicate that heat conduction to the moderator through the walls of the pressure and calandria tubes is sufficient to prevent fuel melting. As discussed in subsequent sections of this chapter, the loss of coolant accident coupled with the failure of the ECC system, is an event which must be addressed in Canadian licensing analyses.

The HWR is also equipped with an alternate shutdown system in the form of gadolinium poison injection into the moderator. The performance of this system is enhanced, and its design simplified, by the near atmospheric pressure of the moderator circuit. The ability to inject soluble poison into the low

pressure moderator provides an alternate shutdown system of comparable speed to that of the shutdown rods. Likewise, all reactor control and shutdown devices (e.g., zonal control system and shutdown rods are inserted into the moderator, avoiding the necessity of penetrating the pressure boundary as required in LWRs.

The use of a moderator system separate from the reactor coolant system also has implications to overpower transients. In the pressurized water reactor, where light water serves as both coolant and moderator, increases in core power will cause voiding and hence the expulsion of moderator from the active region of the core. The removal of moderator from the active region of the core will decrease reactivity, and hence limit the magnitude of overpower transients. In contrast, overpower transients in pressure tube reactors do not affect the degree of moderation present in the core because of the separation of moderator and coolant circuits. The use of separate moderator and coolant circuits is consequently an inherent characteristic which can contribute to positive power and coolant void coefficients of reactivity.

° Coefficients of Reactivity

The power coefficient of reactivity at operating power levels is near zero, and may even be slightly positive, in the HWR. As a result, transients which increase core power can increase core reactivity, and hence tend not to be self limiting. The void coefficient of reactivity in the HWR is positive. As a result, depressurization events (such as the loss of-coolant accident) and undercooling transients which lead to boiling in the fuel channels will result in an increase in core reactivity, and hence further increases in channel power and voiding. The void and power coefficients of reactivity are thus viewed as inherent characteristics of the HWR which are less satisfactory than those of light

water reactors. Partially mitigating the effect of these positive coefficients is the longer mean neutron lifetime (approximately 30 times larger than for LWRs), which result in much slower prompt critical power transients than would occur in an LWR for an equal amount of reactivity insertion.

There are, of course, a number of accidents in the LWR which can insert positive reactivity, so that the positive reactivity introduced in the HWR by coolant voiding is by no means a unique characteristic. For example, the collapse of voids in a BWR as a result of pressure increases and the ejection of control rods in PWRs and BWRs would also increase reactivity. LWRs have also operated with a positive moderator temperature coefficient, where increases in moderator temperature would result in a reactivity insertion. However, it should be noted that these reactivity transients in light water reactors are self limiting, since the resulting increase in core power will expel moderator from the core (cause voiding) and thus reduce core reactivity terminating the power transient; the operation of engineered systems may nevertheless be required to limit the consequences of transients to acceptable levels. Because of the separate moderator and coolant systems of the HWR, increases in core power caused by coolant voiding are inherently nonlimiting, and consequently must be terminated by reactor trip.

As mentioned previously, the rate of increase of core power during depressurization or undercooling events is reduced by the longer mean neutron lifetime (should the amount of voiding be sufficient to cause the reactor to approach prompt critical). Canadian analyses indicate that the power excursions which result from these events can be easily terminated by the trip system before fuel damage occurs. Plant safety is also enhanced by providing two fast acting and independent shutdown systems: the shutdown rods and the soluble poison injection system.

As a result of the positive coolant void coefficient of reactivity, cold water accidents (as steam line break, excess load) are less severe in the HWR, since they result in a decrease in core reactivity. In the PWR, cold water accidents increase reactivity because of highly negative moderator temperature coefficient of reactivity.

° On-Line Refueling

The on-line refueling capability of the HWR facilitates the detection and removal of damaged fuel. As a result, the activity levels in the reactor coolant can be maintained at lower values than in an LWR. In addition, the economic impact of the small amount of fuel damage which would release activity from the fuel is much less because the plant would not have to be shut down for two or more weeks for refueling to remove damaged fuel, as would be the case for the LWR.

The amount of excess reactivity held up in control poisons is considerably less in the HWR because of on-line refueling. In light water reactors, the excess reactivity required to operate between refueling intervals must be held up in control poisons, and is potentially available for release during accidents which might expel these poisons from the core. In the PWR, soluble boron and burnable poison shims are employed for this purpose; since these poisons either cannot be removed from the core during accidents, or can be removed only slowly, the excess reactivity of the PWR does not introduce licensing issues materially different from those of the HWR. In the case of the BWR, control rod insertion is used to partially hold up this excess reactivity, with the potential for significant reactivity additions during postulated accidents in which these control rods are expelled from the core.

° Tritium

Larger quantities of tritium are produced in the HWR because of the interaction of neutrons with the heavy water moderator and coolant. Although the higher tritium inventories could have an environmental impact if released, economic considerations have dictated that the HWR be designed to prevent the loss of significant amounts of heavy water. The majority of the tritium inventory is contained in the moderator region and the isolation of the tritium-bearing moderator from the environment is simplified by the low pressure (near atmospheric) at which the moderator operates.

° Xenon Stability

HWRs of the 600 MW(e) size are unstable to xenon oscillations. As a result, a control system must be provided to stabilize the core against xenon transients, and the protection system must be designed so as to preclude excessively high local power densities.

In order to control xenon transients, reactors of the Candu type are provided with an automatic control system. This control system utilizes in-core detectors to establish the local power density in regions of the reactor core, and a zonal control system operated by the plant computer to adjust the local power density and keep it within acceptable limits. The plant is also provided with power setback and stepback control actions, which serve to reduce reactor power (gradually or in a stepwise fashion, respectively), should the power density in a region of the core become excessive. These control system actions are fully backed by reactor trips (two independent shutdown systems) which utilize information from separate sets of in-core detectors

and which will shut the reactor down upon detecting high local power densities, should control actions fail.

5.5.2.3 Safety Related Design Basis Events

This section intercompares U.S. and Canadian procedures for determining and categorizing design basis events, the assumptions used in evaluating these events, and the criteria used in the U.S. and Canada to judge their acceptability. The purpose of this inter-comparison is to provide some insight into whether the consequences of these design basis events, as predicted by Canadian safety analyses, will also prove acceptable in the U.S. To the extent that these design basis events are determined and analyzed in a manner consistent with U.S. practice and evaluated against acceptance criteria which are as, or more, demanding than the corresponding criteria employed in the U.S., it can readily be concluded that they satisfy U.S. licensing requirements. It is more difficult to draw conclusions in those instances where U.S. licensing imposes additional requirements or where U.S. acceptance criteria is more demanding or differs significantly from the criteria employed in Canada. It is incorrect to assume in this instance that the Canadian design does not meet U.S. licensing requirements; rather, one can only conclude that more detailed and quantitative evaluations are necessary to definitively establish the degree of conformance to U.S. requirements.

In particular, there are several requirements imposed by U.S. regulation which do not have to be explicitly addressed under Canadian licensing rules. However, this does not mean that these aspects are not addressed in Candu design or that the Candu design would necessarily fail to meet U.S. requirements on this account. An example, which is discussed in greater detail below, is the question of fuel damage during moderate frequency events. It is a requirement of U.S. licensing that no fuel damage occur

as a result of these moderate frequency events, while Canadian licensing imposes no explicit limitations on fuel damage during such events. Nevertheless, the question of fuel damage is addressed in Candu design and sufficient margins are provided in order to avoid the economic consequences of fuel failure during events which may be anticipated to occur; limitations are also placed upon the activity in the primary coolant. As a result, U.S. requirements on fuel damage may well be met, but detailed assessments would be required in order to definitively demonstrate this supposition or to establish the additional margins, if any, required to fully meet U.S. requirements. It should also be noted that conservatisms are often introduced in somewhat different fashions in U.S. and Canadian analyses. In particular, Canadian licensing rules require that accidents (single failures) be evaluated under the assumption that only one of the two shutdown systems is available, and further that the first (fastest) trip signal to the safety system is also unavailable. These assumptions introduce conservatisms which would not be required under U.S. licensing. Events which do not appear to meet U.S. acceptance criteria or which do not include conservatisms explicitly imposed by U.S. regulation may nevertheless prove to meet U.S. requirements; here again detailed quantitative analyses would be required to establish the tradeoffs between Canadian and U.S. mandated conservatisms.

° Determination of Design Basis Events

Safety related design basis events are determined in the U.S. as described in Reference 5.5-5. Briefly, the design basis events are generated by concern growing from a variety of sources such as (a) operating experience, (b) the design process, (c) commercial developments, and (d) NRC regulatory developments. There is no unambiguous process for determining design basis events in the U.S., and consequently, the list of design basis events may differ from plant to plant.

Once a concern has arisen, identified by the NRC or design organization, it must be addressed. The first (and in many instances, the most important) step is to define or bound the design basis event to be examined. The nature of the initiating event must be clearly specified as to whether it is mechanistic or arbitrary in nature or some combination of the two. The LOCA Large Guillotine Break is the classic example of an arbitrary initiating event, not supportable by normal engineering experience or predictive analysis. Spurious closure of valves is mechanistic, at least given the initiating signal. The anticipated transient without scram event is a mixed event. Next, the spectrum of events to be considered must be specified. This includes defining the initial conditions for the event and whether or not the event is to be considered alone or in combination with other events. Finally, one must carefully describe the response of associated systems to the event.

In Canada, the safety related design basis events are limited to failures in the process system which could result in unacceptable releases in radioactivity without action from the safety systems. Based on discussions with the Canadians, it appears that the process of defining the safety related design basis events for the CANDU design is similar to the process used in the United States. In both cases, engineering judgment is the main factor used to develop the list. No definitive criteria (such as IEEE 379 for defining single failures in safety systems) exist to guide the implementation of this process.

One subtle difference between the approach used in the CANDU design and the U.S. approach is that the U.S. approach explicitly requires the investigation of events which could be initiated by failures of the protective system; in practice,

significant events of this type are also addressed in CANDU design.

° Categorization of Design Basis Events

The safety related design basis events are categorized in the U.S. on the basis of probability of occurrence. The current NRC regulatory guides recommend the use of the categories shown in Tables 5.5.2-2 thru 5.5.2-6. These are normal operation, moderate frequency, infrequent and limiting fault. Normal operation is all planned operational modes for events as the name implies. Moderate frequency events are those unplanned events which may occur once a year per plant (anticipated operational occurrences). The infrequent events may occur once during the plant lifetime. The limiting faults are incidents which are not expected to occur; however, the probability of occurrence is greater than 10^{-7} per reactor year.

Other classification systems with greater resolution are being proposed to and considered by the NRC at this time. The purpose of these systems is to more precisely obtain a constant product of occurrence probability and consequences. The U.S. industry is also moving towards factoring more operating reactor data on failures into the determination and categorization of safety related design basis events. It is hoped that this effort coupled with the increased resolution being proposed for the categorization systems will permit justifiable relaxation in the acceptance criteria for some of the safety related design basis events which the NRC requires to be addressed.

The Canadian licensing approach does not involve the same type of categorization of safety related design basis events

as in the United States. In Canada, all failures of process systems must meet the same acceptance criteria when all of the safety systems operate. For example, acceptance criteria for a large break LOCA are the same as the acceptance criteria for a failure in the reactor regulating system. However, every process system failure which is postulated must have acceptable consequences in the presence of a failure of one of the safety systems. These acceptance criteria are more relaxed than for single failures because the probability for such a dual failure is much lower than for the case where the safety systems function properly. This approach is based on allowing greater consequences for the lower probability events as in the United States.

° Acceptance Criteria

Acceptance Criteria for the safety-related design basis events are specified for each category of event. The criteria employed in the United States are summarized in Tables 5.5.2-3 through 5.5.2-6 for each category (normal operation, moderate frequency faults, infrequent faults, and limiting faults). Note that no breach of fission product barriers, including the fuel cladding, are permitted as a result of moderate frequency events. In addition, only a small amount of fuel failure is permitted as a result of infrequent faults. Finally, the acceptance criteria contain off-site radiation exposure limits which must not be exceeded; different exposure limits are specified for each category of event.

In Canada, acceptance criteria for safety-related design basis events are based solely on off-site radiation exposure. The off-site exposure limits placed on each category of event (single or dual failure) is specified in Table 5.5.2-1. Unlike U.S. acceptance criteria, there is no requirement

to show that the fuel remains intact during moderate frequency events or of limiting the amount of fuel damage during infrequent events. The question of fuel failures is not viewed as a licensing issue per se, since the off-site radiation exposure limitations provide definitive protection to the health and safety of the public. Limitations are placed on coolant activity during normal operation ($\sim 10^3$ curies I-131) to ensure that site boundary dose limits are not exceeded during abnormal occurrences.

Although the question of fuel failures is not viewed as a licensing issue in Canada, in practice trip set points are typically established to avoid fuel damage during moderate frequency events because of the economic consequences of such failures. Thus in practice, Candu reactors might be considered to operate in accordance with U.S. licensing criteria. However, since limitations on fuel failure are an acceptance criteria under U.S. licensing, it is obligatory in the U.S. to show that the extent of such fuel damage is conservatively predicted, and that margins to predicted fuel damage are provided to accommodate limiting conditions for operation and analysis uncertainties. Since fuel damage is not an acceptance criteria in Canada, there is no reason or obligation to perform such conservative evaluations. A detailed quantitative assessment would be required to establish whether the margins provided to avoid the economic consequences of fuel damage are sufficient to meet the intent of U.S. licensing requirements.

As mentioned previously, separate acceptance criteria are specified in Canada for single and dual failures. It is Canadian licensing practice to analyze the accidents (process failure) on the assumption that the safety systems function properly (except that credit is taken for the operation of

only one shutdown system), and then to reanalyze the events on the assumption that any one of the following safety systems was in a failed state when the process failure occurred:

1. Shutdown System 1
2. Shutdown System 2
3. ECC System
4. Containment System

It should be emphasized here that, except for the containment system, failure of a safety system in this context means unavailability of the entire system (failure of a component in a redundant system that still leaves the particular safety function intact is not counted as a safety system failure). For the containment system, different degrees of impairment are postulated. Consequently, each safety-related design basis event has to be analyzed a number of times, as shown in Table 5.5.2-7, once for the single failure with all protective systems functioning properly, and additional times for the unavailability of each of the independent safety systems.

The requirement to analyze both single and dual failures appears to be unique to the Canadian licensing practice. In the U.S., only events which the Canadians categorize as single failures are analyzed (ATWS might be considered as an exception), i.e., dual failures are generally considered "Class 9" events in the U.S. and are not normally analyzed as part of a license application. As noted previously, Canadian acceptance criteria are less restrictive than those of the U.S. in that they allow fuel failure during moderate frequency events. On the other hand, as Table 5.5.2-1 illustrates, Canadian acceptance criteria allows only small

off-site exposures for all single-failure events, including limiting faults. Thus, unexpected events (for example, a loss-of-coolant accident) must meet the same acceptance criteria as do moderate frequency events. In the case of dual failures, the acceptance criteria is approximately equivalent to the U.S. acceptance criteria for single failure events, (limiting faults) although the demands placed upon reactor design by the assumption of unavailability of a safety system are certainly greater.

There are other Canadian conditions which must be met. One of these requirements is to demonstrate that the frequency of serious process failures is less than one per three years. The serious process failures are defined as failures in the process system which necessitate protective system action to protect the plant. Process failures which are dealt with by process systems without necessitating protective system action are not counted in the tabulation of serious process failures. Compliance with this frequency criteria is demonstrated by early plant operating experience.

The requirement that the frequency of serious process failures be less than one in three years is achieved where possible by designing the process system for less frequent failure probability and where infeasible (for example, the loss of load) by designing the control system as the first line of defense to negate the consequences of process failures such that protective system action is not required. Thus credit is taken for the control system in reducing the frequency of serious process failures (i.e., failures where the protective system must act); however, in performing single and dual failure accident analyses, no credit is taken for the action of the control system (unless that action is adverse).

Another requirement is that the designer must demonstrate the unavailability of each safety system is less than 1×10^{-3} . This means that the safety system must be available to perform its intended function at least 99.9% of the time. The unavailability of the safety system refers to unavailability of the entire system. (Failure of the component in a redundant safety system could leave the particular safety function intact.)

In U.S. licensing practice, the principal thermal hydraulic design basis is avoidance of thermally induced fuel damage during normal steady state operation and during moderate frequency events. The following design limits are established to meet this criterion:

- a. A minimum allowable limit is set on the departure from nucleate boiling ratio (DNBR) during normal operation and any moderate frequency event to assure that there is a 95% probability with a 95% confidence that departure from nucleate boiling will not occur, e.g., a DNBR equals 1.3 for the W-3 correlation (see Table 5.5.2-8).
- b. The peak temperature of the fuel will be less than the melting point during normal operation and any moderate frequency event. (see Table 5.5.2-9)

The Canadian fuel limits for accidents are shown in Table 5.5.2-10. In the CANDU design, there are no explicit safety limits applied to the fuel; the only limit is on activity release. A sufficient condition to preclude activity releases in excess of the limits is to maintain primary heat transport system integrity. If this primary heat transport system integrity is maintained, there will be no way that the activity can get out of the primary heat transport system.

As a result, releases of activity from the fuel could not migrate to the site boundary.

In order to maintain the integrity of the primary heat transport system sufficient conditions have been defined which, if met, preserve the primary heat transport system integrity. These conditions are the following:

1. Primary heat transport system pressure below an acceptable limit.
2. Avoidance of centerline fuel melting.

For safety related design basis events it is demonstrated that these sufficient conditions are met and therefore that the primary heat transport system integrity is maintained, thus preventing release of any activity to the site boundary. (Some fuel defects may be caused by these events without violating the sufficiency conditions, however they are considered of economic concern only.)

For the loss of coolant event where the primary heat transport system's integrity cannot be maintained, the fuel damage is assessed and appropriate activity releases are calculated. These releases are then compared with the limits at the site boundary to demonstrate acceptable consequences.

Fuel limits for normal operation are specified in Table 5.5.2-11. For normal operation, an economic limit of no fuel sheath defects has been established. In order to preclude these defects, it has been determined that a sufficient condition is the avoidance of dryout. The primary method of accomplishing this is to use the control systems to limit the consequences and the probability of occurrence of any

excursion. However, appropriate margins are factored into the design such that many of the safety related design basis events are terminated by the protective systems prior to the onset of dryout, in order to avoid the economic consequence of potential fuel failures, even if the control systems failed to perform this task. Thus, the amount of or need for margin to the dryout limit during normal operation is based on economic considerations, rather than safety considerations.

A comparison of the Canadian activity release limits with those in the United States is provided in Table 5.5.2-12.

° Protective Actions

The core shutdown systems for the CANDU design are summarized in Table 5.5.2-13. Two shutdown systems are provided: shutdown system 1 inserts mechanical control rods into the core, while shutdown system 2 injects soluble poison (gadolinium) into the moderator. Each shutdown system typically employs two out of three logic and is designed to have a total unreliability of less than 10^{-3} . Trip setpoints for shutdown system 2 are set somewhat above those of shutdown system 1, so that the second shutdown system is typically activated only if shutdown system 1 fails. Once activated, the poison injection system is quite fast, discharging the soluble gadolinium poison into the moderator within about 0.5 seconds.

The high neutron power (local) trip uses in-core detectors. Each shutdown system is completely independent, with shutdown system 1 incorporating 34 vertically mounted detectors, while shutdown system 2 employs 23 in-core flux detectors which enter the core horizontally. The high rate log neutron power trip employs ex-core ion chambers, with three chambers being provided for each shutdown system. The check marks in

Table 5.5.2-13 indicate the other trip parameters provided on each shutdown system.

Several of the shutdown system trip parameters play a role in preventing centerline fuel melt and dryout. These trips include the high neutron power (local), high rate log neutron power, and low flow trips. In addition, the setbacks and stepbacks in the control system (described in Section 5.5.2.3) act as additional line of defense against exceeding the fuel limits.

Other protective functions include the emergency core cooling system and the containment system as shown in Table 5.5.2-14. Furthermore, various support systems play a role in mitigating accident consequences. One example of such a system is the auxiliary steam generator feedwater pump which provides an emergency source of water to the steam generators if the main feedwater pumps are unavailable.

Operator action is required in several cases for plant protection in PWRs. Typical actions include isolation of the steam generator after a tube rupture, emergency feedwater flow control during small breaks, and secondary steam dump control to cool the plant down to shutdown cooling after various events.

In the CANDU design, it appears that there are more operator actions required to mitigate the consequences of safety related design basis events than in the PWRs. Many of the supplementary safety systems are not automatically controlled but require operator actuation. The minimum time at which operator action is assumed in the safety analysis for CANDU is 15 minutes; shorter times may be justified and are handled on a case by case basis. Other typical necessary operator

actions in the CANDU design involve maintaining an acceptable post accident configuration after safety related design basis events have arrived at a stable configuration.

° Initial Conditions

For the PWR, operating limits on DNB and peak linear heat rate are imposed to ensure that the fuel design limits are not exceeded during moderate frequency events. These limits must be maintained in order to be consistent with the assumed initial conditions in the safety analysis. These operating limits serve one of two functions for the moderate frequency events: (a) they either provide adequate margin for the event to proceed to completion without exceeding the fuel design limits or, (b) they provide adequate margin for the protection system to be able to shut down the plant in time to keep the fuel design limits from being exceeded. In establishing these limits, the uncertainties in the process parameter measurements and in the processing to determine the limit parameters are considered. The safety analysis is done on the basis that all parameters are initially at their operating limits when the event occurs. These operating limits can be placed either on individual parameters or on combinations of parameters.

Because fuel damage is permitted from a licensing standpoint during safety related design basis events in the CANDU design, there is no licensing reason for imposing operating limits on parameters related to dryout (although limits are in fact imposed to limit the probability of occurrence of dryout because of the economic consequences of fuel damage). In addition, the LOCA is evaluated to be acceptable at the peak linear heat value permitted by the shutdown systems, so that no additional operating limits must be imposed due to

consideration of this accident. As a result, the plant output which can be achieved is limited only by 1) margin which the operator must maintain to the shutdown system limits to avoid spurious shutdowns, 2) the margin which the operator must maintain to the control system setback and stepback limits to avoid spurious power reductions, and 3) the power ceiling established by the plant license.

Although there are no operating limits on fuel related parameters for licensing reasons in the CANDU design, equipment is provided to enforce limits which have been specified to avoid fuel damage with its associated economic consequence. An on-line flux mapping system is used to monitor the core power distribution to aid in maintaining that distribution within the appropriate limits, and a more refined off-line flux mapping system is also available to the operator or control systems. In addition, setback and stepback functions are incorporated within the control systems to automatically regulate the plant within pre-established limits.

A setback is a power rampdown at a pre-established rate for as long as the setback condition is in effect. A stepback is a power step down achieved by dropping a dedicated group of four control rods into the core. This stepback proceeds until the rods are either dropped completely into the core or until the condition clears. When the condition clears, the rods are caught in the process of falling into the core so that a vernier control of the stepback magnitude is achieved. The rates and magnitudes of the setbacks and stepbacks and the initiating parameters are contained in Table 5.5.2-15 while Table 5.5.2-16 summarizes the key features of reactor control and Figure 5.5.2-1 provides a block diagram of the reactor regulating system. The relationship

of the actuation setpoints for the setbacks and stepbacks relative to normal operation and the setpoints for the two shutdown systems is shown in Table 5.5.2-17.

Implementation of these setbacks and stepbacks is carried out via the digital control computer which is in service at the time. (Two redundant digital computers are utilized for the control functions in the CANDU design, including the on-line flux mapping. One of these two computers is in use at all times while the other is in a ready mode waiting to be switched on if the operator or automatic switchover circuit so dictates.)

In U.S. licensing practice, the equipment used to monitor operating limits within the plant is not part of the protective system. Instead, it is designed to essentially the same criteria as are used for the control systems. The basis for this design approach is that a single failure in the equipment used to monitor the operating limits will not result in a violation of the safety criteria because the reactor protective system provides an automatic reactor trip to prevent the fuel design limits from being exceeded during any excursion which results from an inadvertent violation of the operating limits.

In the CANDU design, the control system setback and stepback functions operate to maintain the plant within a set of pre-established limits during normal operation. These systems combined with some administrative controls are used to maintain the plant parameters within appropriate limits. These limits are adequate to avoid equipment damage and to avoid a reactor trip.

In U.S. licensing practice, technical specifications are issued by the NRC to mandate the appropriate administrative controls for the maintaining the plant within acceptable operating limits during normal operation and for assuring the appropriate operation of the monitoring and protection systems. The definition and purpose of the technical specifications are discussed in Tables 5.5.2-18 through 5.5.2-21. In summary, the technical specifications are the part of the operating license which defines the limitations on nuclear plant operation which are necessary to validate assumptions made in the safety analysis of the plant. The monitoring systems which are provided facilitate the implementation of these administrative controls; however, the prime responsibility for operating the plant within the technical specification limits rests with the operator.

5.5.2.4 Factors in Determining Power Rating

In the CANDU design, the plant power output which can be achieved is limited by the action points of the control system setback and stepback functions. Since one of the purposes of the setbacks and stepbacks is to limit plant power as necessary to avoid reactor trips without utilizing information from the trip systems, additional margin must be added to the setpoints to provide assurance that the setback or stepback will come into action before the trip is initiated. In addition, margin has been added to the trip system limits so that the economic consequences of fuel failure are avoided during many of the design basis events. The net result is that the protective systems and control actions which have been provided to avoid the economic consequences of fuel failure during many of the design basis events determine the power rating of the CANDU reactor. It appears that the CANDU power rating could be increased by the use of more accurate systems for monitoring the approach to dryout in the fuel and by

using the protective system information directly as input to the setback and stepback functions for the purpose of avoiding reactor trips.

An indication of the thermal margin available in the Candu design is given by the following:

- ° Best estimate of the channel power at which dry-out would occur with reactor coolant system inlet temperature, pressure, and header-to-header pressure drop fixed at rated values is 150% of the nominal power level of an average channel in the flattened (central) zone of the reactor. The best estimate of the channel power at which dryout would occur is 136% of the maximum anticipated power (108% of region average power) of channels with minimum flow (98% of average flow) in the flattened (central) region.
- ° The channel power level at which dryout has any significant probability of occurring with reactor coolant system inlet temperature, pressure, and head-to-header pressure drop fixed at rated values is 122% of the maximum anticipated power of channels with minimum flow in the flattened (central) zone.
- ° The reactor power level at which dryout could conceivably occur with core power distribution and reactor coolant pump speed fixed at their rated values is greater than 122%.
- ° The local power level (as interpolated from two or more in-core detector measurements) at which dryout could conceivably occur with localized core peaking, resulting from various single failures and allowing for measurement uncertainty, is 112% of the value at rated power.

With the numbers given above, high neutron flux power trip setpoints of 112% of the normal detector value at rated power will preclude dryout unless the uncertainties happen to stack up more adversely than assumed or severe localized peaking occurred which is not fully measured by the nearest detectors. If it were to become necessary in the United States to avoid fuel damage with a higher probability, these setpoints for the high neutron flux power trips would need to be reduced. However, if these setpoints were reduced, there would be less margin available between the setbacks, stepbacks, and trips and also less margin between the normal full power operating point and the actuation points for the setback and stepback control actions. The net effect is that the power rating would need to be reduced if the high neutron flux power trip setpoints had to be reduced substantially.

TABLE 5.5.2-1

LICENSING IN CANADA

AECB SITING PRINCIPLES/GUIDELINES

1) PLANT - PROCESS SYSTEMS

SAFETY SYSTEMS

SHUTDOWN
ECC
CONTAINMENT

2) SYSTEMS SEPARATE AND INDEPENDENT

- AS FAR AS PRATICABLE

3) "SINGLE" AND "DUAL" FAILURES CONSIDERED

SINGLE < 1 EVENT IN 3 YEARS

DUAL < 1 EVENT IN 1000 YEARS

REFERENCE (ACCIDENT) DOSE LIMITS

		SINGLE	DUAL
INDIVIDUAL	EXT. WHOLE BODY DOSE	0.5 rem	25 rem
	THYROID DOSE	3 rad	250 rad
POPULATION	EXT. WHOLE BODY DOSE	10^4 m-r*	10^6 m-r
	THYROID DOSE	10^4 m-r	10^6 m-r

* man-rem

TABLE 5.5.2-2

U.S. EVENT CLASSIFICATION

CONDITION I - NORMAL OPERATION

CONDITION II - MODERATE FREQUENCY

CONDITION III - INFREQUENT

CONDITION IV - LIMITING FAULT

TABLE 5.5.2-3

U.S. EVENT CALSSIFICATION

CONDITION I NORMAL OPERATION

- A) ALL PLANNED OPERATIONAL MODES OR EVENTS
- B) EXAMPLES: STARTUP, STEADY STATE POWER OPERATION, LOAD CHANGES

TABLE 5.5.2-4

U.S. EVENT CLASSIFICATION

CONDITION II - MODERATE FREQUENCY

- A) EVENTS WHICH MAY OCCUR ONCE A YEAR PER PLANT
- B) REQUIREMENT - RETURN TO POWER AFTER CORRECTIVE ACTION, FISSION PRODUCT RELEASE AS LOW AS REASONABLY POSSIBLE, (LESS THAN 0.5 REM/YR.)
- C) EXAMPLES - PARTIAL LOSS OF FEEDWATER FLOW, CEA DROPS.

TABLE 5.5.2-5

U.S. EVENT CLASSIFICATION

CONDITION III - INFREQUENT

- A) AN EVENT WHICH MAY OCCUR ONCE DURING THE PLANT LIFETIME.
- B) REQUIREMENT - LIMITED FUEL DAMAGE IS PERMITTED. FISSION PRODUCE RELEASE SHOULD NOT AFFECT PUBLIC USE OF PROPERTY OUTSIDE THE EXCLUSION AREA.
(TWO HOUR IODINE DOSE LESS THAN 0.3 REM TO THE THYROID).
- C) EXAMPLES - LOSS OF ALL NORMAL A.C. POWER.

TABLE 5.5.2-6

U.S. EVENT CLASSIFICATION

CONDITION IV - LIMITING FAULT

- A) AN INCIDENT WHICH IS NOT EXPECTED TO OCCUR, BUT THE PROBABILITY OF OCCURRENCE IS GREATER THAN 10^{-7} PER REACTOR YEAR.
- B) REQUIREMENTS - SHOULD NOT CAUSE UNDUE RISK TO PUBLIC HEALTH AND SAFETY. FISSION PRODUCT RELEASE LESS THAN THAT RESULTING IN A TWO HOUR IODINES THYROID DOSE OF 300 REM.
- C) EXAMPLES - MAIN STEAM PIPE BREAKS.

TABLE 5.5.2-7

ACCIDENT MATRIX

FUEL COOLING: 1) POWER INCREASE
 2) FLOW DECREASE
 3) COOLANT TEMP. INCREASE

EXAMPLES: 1) LOSS OF REGULATION (LOR)
 2) LOSS OF COOLANT (LOC)
 3) LOSS OF HEAT SINK (LOHS)

SINGLE/DUAL APPROACH

	SINGLE	DUALS			
		SDS1	SDS2	ECC	CONT.
LOR	✓	✓	✓	—	—
LOC	✓	✓	✓	✓	✓
LOHS	✓	✓	✓	—	—

ASSESSMENT OF ALTERNATE
HEAT SINKS

TABLE 5.5.2-8

U.S. CRITERIA

DEPARTURE FROM NUCLEATE BOILING RATIO

SAFETY LIMIT: THE HEAT FLUX AT WHICH THERE IS A 95% PROBABILITY, AT THE 95% CONFIDENCE LEVEL, THAT NO DEPARTURE FROM NUCLEATE BOILING (DNB) OCCURS.

THE DNB RATIO IS DEFINED AS: $\text{CRITICAL HEAT FLUX} / \text{ACTUAL HEAT FLUX}$

OPERATING LIMIT: THE MAXIMUM POWER LEVEL THAT THE PLANT MAY BE OPERATED AT WHICH WILL NOT CAUSE THE DNB RATIO TO REACH A VALUE LESS THAN THE SAFETY LIMIT FOLLOWING EXPECTED EVENTS.

DEPENDENCE: POWER LEVEL, COOLANT FLOW RATE, PRESSURE, TEMPERATURE, POWER DISTRIBUTION

TABLE 5.5.2-9

U.S. CRITERIA

MAXIMUM FUEL POWER DENSITY

SAFETY LIMIT: THE POWER DENSITY AT WHICH THE FUEL BEGINS TO MELT AT THE PELLET CENTERLINE.

EXPRESSED AS THE LINEAR HEAT GENERATION RATE (KW/FT).

OPERATING LIMIT: THE MAXIMUM LINEAR HEAT GENERATION RATE THAT THE PLANT MAY BE OPERATED AT WHICH WILL NOT CAUSE CENTERLINE FUEL MELTING TO OCCUR DURING EXPECTED OPERATION OR THE MAXIMUM HEAT RATE ASSUMED IN LOCA ANALYSIS

WHY: CONTINUED OPERATION OF A PRESSURIZED WATER REACTOR BEYOND CENTERLINE MELTING HAS NOT BEEN PROVEN ACCEPTABLE.

DEPENDENCE: POWER LEVEL, POWER DISTRIBUTION

TABLE 5.5.2-10

CANADIAN FUEL LIMITS FOR ACCIDENTS

SAFETY LIMITS:

NO EXPLICIT FUEL LIMITS JUST ACTIVITY RELEASE
FOR SINGLE OR DUAL FAILURE

SUFFICIENT CONDITIONS:

A) ○ MAINTAIN PHT INTEGRITY

○ PREVENT FUEL MELTING ARBITRARY

-MAY CAUSE SOME FUEL DEFECTS

→ ECONOMIC CONCERN ONLY

-ELIMINATE USING REGULATING
SYSTEM IF ACCIDENT "FREQUENT"

B) ○ FOR LOC, ASSESS FUEL DAMAGE / RELEASE

→ ESTIMATES OF FUEL PERFORMANCE

METHOD:

-USE SAFETY SYSTEMS TO LIMIT CONSEQUENCES

TABLE 5.5.2-11

CANADIAN FUEL LIMITS FOR NORMAL OPERATION

ECONOMIC LIMIT:

- FUEL SHEATH DEFECTS

SUFFICIENT CONDITION:

- AVOID DRYOUT

METHOD:

- USE REGULATING SYSTEM TO LIMIT CONSEQUENCES AND PROBABILITY OF ANY EXCURSION FROM NORMAL OPERATING CONDITIONS

TABLE 5.5.2-12
REFERENCE DOSE LIMITS
CANADA

		SINGLE FAILURE	DUAL FAILURE
INDIVIDUAL	EST. WHOLE BODY DOSE	0.5 rem	25 rem
	THYROID DOSE	3 rads	250 rads
POPULATION	EST. WHOLE BODY DOSE	10⁴ man-rem	10⁶ man-rem
	THYROID DOSE	10⁴ man-rem	10⁶ man-rem

U.S. - NRC

10CFR20.105

PERMISSIBLE LEVELS OF RADIATION IN UNRESTRICTED AREAS (NOT INCLUDING EFFLUENT RELEASE LIMITS)

- A) FOR AVERAGE RADIATION LEVELS AND ANTICIPATED OCCUPANCY: 0.5 rem/yr WHOLE BODY**
- B) RADIATION LEVELS CAUSING DOSE OF: 2 mrem IN ONE HOUR OR 100 mrem IN 7 CONSECUTIVE DAYS**

10CFR100.11

REACTOR SITE CRITERIA (FOR MAJOR ACCIDENTS)

	<u>REM WHOLE BODY</u>	<u>REM THYROID</u>
A) SITE BOUNDARY IN 1ST 2 HOURS	25	300
B) LOW POPULATION ZONE DURING CLOUD PASSAGE	25	300

TABLE 5.5.2-13

SHUTDOWN SYSTEM TRIP SIGNALS

	<u>SDS#1</u>	<u>SDS#2</u>
HIGH NEUTRON POWER (LOCAL)	√1	√3
HIGH RATE LOG NEUTRON POWER	√2	√4
HIGH PHT PRESSURE	√	√
LOW GROSS FLOW	√	√
LOW SURGE TANK LEVEL	√	√
LOW STEAM GENERATOR LEVEL	√	√
HIGH BUILDING PRESSURE/TEMPERATURE	√5	√6

SHUTDOWN SYSTEM #1 INSERTS SHUTDOWN RODS INTO THE CORE WHEN A TRIP LIMIT IS VIOLATED.

SHUTDOWN SYSTEM #2 INSERTS SOLUBLE POISON INTO THE MODERATOR WHEN A TRIP LIMIT IS VIOLATED.

1. 34 IN-CORE DETECTORS WHICH ENTER FROM TOP OF CORE
2. 3 OUT OF CORE ION CHAMBERS
3. 23 IN-CORE FLUX DETECTORS WHICH ENTER THE CORE HORIZONTALLY
4. 3 OUT OF CORE ION CHAMBERS
5. BUILDING PRESSURE
6. BUILDING TEMPERATURE BEING CONSIDERED

TABLE 5.5.2-14

SAFETY SYSTEMS DESIGN BASIS

SHUTDOWN

- MUST OVERTAKE ANY POWER TRANSIENT
- POST-SHUTDOWN POWER TO DECAY LEVELS
- 2 SHUTDOWN SYSTEMS, INDEPENDENT, TESTABLE, NOT GREATER THAN 10^{-3} YRS/YR
- LARGE LOC SETS RATE + DEPTH REQ'T

ECC

- RE-ESTABLISHES FUEL COOLING
- MAINTAINS WATER LEVEL FOR LONG TERM
- SUPPLY FROM STORAGE TANK
- INJECTION INTO RIH AND ROH
- LONG TERM HEAT SINKS: MAIN HEAT EXCHANGERS, RECOVER COOLERS, BUILDING COOLERS

CONTAINMENT

- COOLS ESCAPING COOLANT/SUPPRESSES OVERPRESSURE
- CONTAINS ANY ACTIVITY
- PROVIDES LONG-TERM COOLING/SUBATMOS PRESSURE
- TWO TYPES

TABLE 5.5.2-15

SETBACK CONDITIONS

<u>VARIABLE</u>	<u>RATE</u>	<u>ENDPOINT (% FULL POWER)</u>
HIGH LOCAL NEUTRON FLUX	.1%/sec	40
SPATIAL CONTROL OFF NORMAL	.1%/sec	60
LOW BOILER LEVEL	1%/sec	20
LOW DEAERATOR LEVEL	1%/sec	2
HIGH BOILER PRESSURE	.5%/sec	8
HIGH SURGE TANK LEVEL	.1%/sec	2
MANUAL	.5%/sec	2
SETBACK ON HIGH MODERATOR TEMP	1%/sec	2
LOW DIFFERENTIAL PRESSURE ACROSS MODERATOR PUMPS	1%/sec	2

STEPBACK CONDITIONS

<u>CONDITION</u>	<u>ENDPOINT</u>
PRIMARY HEAT TRANSPORT PUMP FAILURE	60% OF CURRENT POWER
TURBINE TRIP, LOSS OF LINE, LOSS OF STATOR COOLING	60% FULL POWER
HIGH PRIMARY HEAT TRANSPORT PRESSURE	0%
LOW STEAM GENERATOR LEVEL	0%
HIGH NEUTRON FLUX	0%
HIGH LOG RATE	0%
REACTOR TRIP	0%

NOTE: THE ENDPOINT OF 0% IS NEVER ACTUALLY REACHED

TABLE 5.5.2-16

CANDU REACTOR

REACTOR CONTROL

- ON - POWER REFUELING DEPTH SMALL
- POSITIVE VOID COEFFICIENT AFFECTS RATE REQUIREMENT
- THERMAL "INERTIA" OF FUEL (approx. 10 sec. time const.) HELPS
 - LIQUID ZONES 0.1 mk/s., $\pm$ 7 mk in 14 UNITS
 - SOLID ABSORPTIONS 0.1 mk/s., - 10 mk in 4 UNITS
 - ADJUSTER RODS 0. ms/s., + 15 mk in 0.1 UNITS
 - SOLUABLE POISONS
- DEVICES OUTSIDE PRESSURE BOUNDARY OF COOLANT
- A DIGITAL CONTROL COMPUTER WITH A BACKUP COMPUTER
- FLUX MAPPING AND SPATIAL CONTROL
- SETBACK/STEPBACK FUNCTIONS

TABLE 5.5.2-17
LOSS OF REGULATION

LOR-HIGH POWER

FAST
INTERMEDIATE
SLOW

HIGH RATE LOG N, HIGH POWER
HIGH POWER, HIGH RATE LOG N
HIGH POWER (LOCAL), MANUAL

LOR-LOW POWER

FAST
INTERMEDIATE
SLOW

HIGH RATE LOG N, HIGH P
HIGH P, HIGH POWER (LOCAL)
HIGH POWER (LOCAL), MANUAL

TABLE 5.5.2-18

TECHNICAL SPECIFICATIONS - THAT PART OF THE OPERATING LICENSE WHICH DEFINES THE FOLLOWING LIMITATIONS ON NUCLEAR PLANT OPERATION:

1. SAFETY LIMITS
2. LIMITING SAFETY SYSTEM SETTINGS
3. LIMITING CONDITIONS FOR OPERATION
4. SURVEILLANCE REQUIREMENTS
5. DESIGN FEATURES
6. ADMINISTRATIVE CONTROLS

TABLE 5.5.2-19

DEFINITIONS

SAFETY LIMITS (SL) - LIMITS ON PARAMETERS RELATED TO CERTAIN BARRIERS TO FISSION PRODUCT RELEASE; CHOSEN TO PROVIDE ASSURANCE THAT THE BARRIER INTEGRITY IS PRESERVED.

LIMITING SAFETY SYSTEM SETTINGS (LSSS) - CONSERVATIVE ACTUATION LIMITS FOR SPECIFIED AUTOMATIC PROTECTION DEVICES; CHOSEN SUCH THAT THE CONSEQUENCES OF SPECIFIC PLANT PROCESS CONDITIONS ARE WITHIN THE REQUIRED LIMITS FOR THOSE CONDITIONS.

LIMITING CONDITIONS FOR OPERATION (LCO's) - THE LOWEST FUNCTIONAL CAPABILITY AND PERFORMANCE LEVELS OF EQUIPMENT, THE LIMITING VALUES OF PROCESS PARAMETERS, AND THE CONSERVATIVE ACTUATION LIMITS FOR SPECIFIED AUTOMATIC PROTECTIVE DEVICES REQUIRED FOR OPERATION CONSISTENT WITH THE SAFETY ANALYSIS OR WITH AN ACCEPTABLE LEVEL OF SAFETY.

TABLE 5.5.2-20

DEFINITIONS

SURVEILLANCE REQUIREMENTS - REQUIREMENTS RELATING TO TEST, CALIBRATION, MONITORING OR INSPECTION TO ASSURE THAT THE NECESSARY QUALITY AND OPERATIONAL STATUS OF SYSTEMS AND COMPONENTS ARE MAINTAINED OR TO ASSURE THAT PARAMETERS AND SETPOINTS ARE PERIODICALLY VERIFIED OR BE WITHIN THE LCO's.

DESIGN FEATURES - PHYSICAL CHARACTERISTICS OF A FACILITY NOT COVERED IN OTHER SPECIFICATIONS, WHICH, IF CHANGED, COULD SIGNIFICANTLY AFFECT THE RADIOLOGICAL SAFETY OF THE FACILITY.

ADMINISTRATIVE CONTROLS - REQUIREMENTS RELATING TO ORGANIZATION AND MANAGEMENT, PROCEDURES, RECORD KEEPING, REVIEW AND AUDIT, AND REPORTING NECESSARY TO ASSURE SAFE OPERATION OF THE FACILITY.

TABLE 5.5.2-21

THE PURPOSE OF TECH SPECS - TO DEFINE AND ASSURE THE FOLLOWING:

1. FACILITY PHYSICAL CHARACTERISTICS
(Number of Steam Generators Operable)
2. STATUS OF EQUIPMENT
(Minimum Number of Operable Reactor Protection System Channels)
3. OPERATING STATE OF EQUIPMENT
(Control Rod Insertion Limits)
4. VALUES OF PROCESS PARAMETERS
(Reactor Coolant System Pressure)
5. CONDITION OF EQUIPMENT AND STRUCTURES
(Maximum Permitted Reactor Coolant System Leak Rate)

ALL OF THE ABOVE ARE ASSUMED IN THE SAFETY ANALYSIS AND MUST BE PRESERVED
TO ASSURE THE VALIDITY OF THE SAFETY ANALYSIS THROUGHOUT THE OPERATING LIFE
OF THE PLANT

Figure 5.5.2-1
REACTOR REGULATING SYSTEM - BLOCK DIAGRAM

5.5.3

Core-Related Safety and Licensing Impact of Siting Candu in U.S.

The safety and licensing impact of siting a Candu-type reactor in the U.S. will be highly dependent on the degree to which the Canadian design approach is accepted by the NRC, and the degree to which U.S. licensing criteria are altered to reflect the inherent attributes of the HWR. Those aspects of the HWR design which could necessitate significant safety and licensing research and development (R&D) efforts for U.S. siting are reviewed in this section.

5.5.3.1 Adequacy of Zirconium Alloy Material and Rolled Joint as Pressure Boundary

The zirconium alloy pressure tube and rolled joint which connects the pressure tubes to their end fittings serve as the primary system pressure boundary in Candu-type systems. Since there is no comparable pressure boundary in the light water reactor, the experimental data base to justify the adequacy of the zirconium alloy pressure tube material and the rolled joint are not available in the U.S., nor have ASME design codes been developed.

Prior to licensing a Candu-type reactor in the U.S. (at the standard Candu operating pressure of 1600 psia), it will be necessary to develop the data base on pressure tube and rolled joint performance, or to obtain such a data base from Canadian experience. This experimental data base would address the properties of the irradiated zirconium alloy pressure tube at the operating temperatures and pressures of the HWR, and of information which demonstrates potential failure modes of the pressure tubes and that such failures do not lead to damage propagation.

Since the conceptual HWR operates at higher primary system pressures and temperatures than do present Candu reactors, it will be

necessary to extend the data base to cover these new operating conditions. Material development programs will be required to establish the strength of the zirconium alloy material and its creep and corrosion resistance at the elevated pressures and temperatures of the conceptual design.

R&D programs should also be performed to demonstrate that the rolled joint can be successfully fabricated using thicker pressure tube material and to demonstrate the integrity of this joint at the operating conditions of the conceptual HWR. No difficulty is anticipated in rolled joint fabrication since early Candu plants utilized pressure tubes of about the same thickness as is postulated for the conceptual design.

5.5.3.2 Limited In-Service Inspectability of Calandria and Calandria Tubes

The calandria and calandria tubes cannot be designed to fully meet U.S. requirements for Class 1 structures because of limitations on the in-service inspectability of these components. Since the calandria and calandria tubes are not viewed to serve a safety function, there is no real necessity for designing these components as Class 1 structures. However, it will be necessary to demonstrate that failures of the calandria or calandria tubes cannot compromise Class 1 structures (i.e., the pressure tubes). The demonstration that calandria or calandria tube failure will not compromise neighboring Class 1 structures is not anticipated to be difficult, since the calandria and calandria tubes operate at near atmospheric pressure.

5.5.3.3 Power and Void Coefficients of Reactivity

As noted in Section 5.5.2.2, the coolant void coefficient of reactivity is positive, while the power coefficient of reactivity under nominal operating conditions is close to zero, and may be

either slightly negative or slightly positive. The U.S. general design criteria No. 11 states, "The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity." Because the prompt inherent nuclear feedback characteristics of the HWR do not tend to compensate for rapid increases in reactivity, the HWR, strictly speaking, does not conform to this general design criteria.

Although the HWR may not conform explicitly to General Design Criteria - 11 it is believed that the HWR meets the general intent of this design criteria when the overall safety implications of the inherent characteristics are considered. In particular:

1. The total amount of positive reactivity addition, even upon complete voiding of the reactor coolant is relatively small, about \$1.7. Since the coolant system is subdivided into two loops, only one-half the core is subject to voiding at one time; this reduces the total reactivity resulting from voiding to slightly more than \$1.
2. Transients which result from the insertion of reactivity beyond prompt critical are relatively slow in the HWR, since the mean neutron lifetime is about 30 times longer than that of the LWR.
3. An alternate shutdown system, diverse and independent of the primary shutdown system, is provided.
4. The effects of the positive coefficients have, of course, been included in Canadian safety analyses. These analyses indicate that the positive void coefficient should not preclude limiting various transients and other postulated accidents to consequences which are viewed to be fully acceptable under U.S. licensing.

5.5.3.4 Relationship of Fuel Damage to Safety and Licensing

One of the most significant issues which is treated differently in Canada and the U.S. is the issue of the relationship of fuel damage to safety and licensability. In the Canadian approach, fuel damage is treated as an economic concern during typical single-failure events, with accident acceptance criteria based solely on site boundary dose limitations; for non-LOCA events, no centerline melt criterion is employed to insure pressure boundary integrity, thus preventing the release of coolant radioactivity and thereby assuring that site boundary limits are not exceeded. On the other hand, it is current U.S. practice to require that fuel damage not occur as a result of the more probable safety-related design basis events.

A considerable effort could be spared in licensing Candu-type reactors in the U.S. if the NRC were to agree that fuel damage is not in itself a licensing issue. There are a number of compelling reasons which support this contention:

1. By maintaining primary heat transport system integrity, any activity which may be released from the fuel would not be released outside of the primary heat transport system.
2. With on-line refueling, any fuel which has failed can be replaced while the reactor is at power, thus quickly reducing the activity level in the primary heat transport system without necessitating extensive reactor shutdown.

The primary impact of adopting the Canadian approach toward fuel failures in the United States would be a considerable reduction in the engineering effort involved with demonstrating the maintenance of fuel integrity during safety-related design basis events, and a significant reduction in the licensing effort would also result. In the Canadian approach, the acceptability of all single failures (the types of accidents evaluated in the U.S.) are evaluated with

respect to site-boundary limits, and all transients and postulated accidents must meet these site boundary limits independent of their anticipated frequency of occurrence (for non-LOCA events, the avoidance of fuel centerline melt is employed as a criteria to insure that these site boundary limits are met). As a result, only limiting events must be analyzed, i.e., only those events which have the maximum consequences with respect to the release of radioactivity. Since Canadian acceptance criteria (site boundary doses) are more restrictive than those of the U.S. (if fuel damage is not considered a safety and licensing issue), it would appear that the various postulated accidents (excluding LOCA which is subject to an independent set of rules in the U.S.) found acceptable by the Canadian safety analysis will also prove acceptable when judged against U.S. acceptance criteria.

The impact of adapting Candu to current NRC practice on fuel damage may be quite significant. The main difficulty with imposing fuel damage as a safety criteria is that it places secondary criteria (in addition to site boundary dose criteria) upon nonlimiting events. Thus, it becomes necessary to evaluate a large number of nonlimiting faults to determine whether fuel damage criteria are met, although it is clear that these events will not exceed site boundary dose limitations. To implement this approach on the Candu reactor would require, first of all, the development of a list of safety-related design basis events categorized by frequency of occurrence. The envelope of initial conditions acceptable for normal operation would then have to be specified on extensive parametric analysis of each safety-related design basis event initiated throughout the allowable range of initial conditions would have to be performed to determine the acceptable setpoints for the protective system. The impact of a wide range of environmental conditions surrounding the protective system components would need to be assessed to include the effects of instrumentation drift and other uncertainties.

As a result of considering fuel failure a safety-related issue, the effort required to perform the safety analysis would be significantly higher. Further, some derating of the power density may be required in order to demonstrate that fuel damage criteria are not violated during all possible operating modes and considering all aspects of uncertainty.

5.5.3.6 Recommendations

In view of the differences in design criteria and other regulations, it is recommended that a more thorough and quantitative safety and licensing evaluation be performed.

There are a large number of General Design Criteria, Code of Federal Regulations, Advisory Committee on Reactor Safeguards Generic Concerns, and NRC staff concerns that deal with LWR safety. In addition, the NRC has issued a large number of regulatory guides, standard review plans, branch technical positions, and other documents defining the detailed requirements that are imposed on the light water reactor. It is expected that the fundamental issues being addressed by each of these documents will need to be addressed also for the Candu design, or for any other commercial reactor operating in the United States. Therefore, it is recommended that a safety and licensing program be implemented for the purpose of performing this evaluation. The output of this evaluation would be a detailed compilation of (1) reasons why Candu complies with the intent of LWR requirements, (2) areas in which Candu would need to be modified to meet the intent of these requirements.

It is further recommended that a preliminary safety analysis report be prepared and submitted to the NRC for their review. This PSAR submittal should contain a quantitative evaluation of various transients and other postulated accidents, with the intent of providing sufficient detail to obtain NRC concurrence with the detailed design approaches taken in Candu to protect the

public health and safety. In order to effectively perform such a detailed safety and licensing evaluation, a joint U.S.-Canadian program is recommended. Although one objective of this preliminary safety analysis report would be to provide a quantitative assessment of Candu conformance to U.S. licensing criteria, the PSAR should also address the current light water reactor concerns that seem to be resolved by the Candu design. In particular, Reference 5.5-6 is a recent report to Congress on recommendations for research to improve the safety of light water power plants. Sixteen research topics were identified. As summarized below, a number of these topics appear to be adequately addressed by Candu design.

- ° Research topic 2 titled "Improved Plant Controls" is adequately addressed by the existing CANDU design. These controls are developed as the first line of defense against events which could result in the release of radioactivity. The planned use of control systems for this purpose provides the type of significant risk reduction being sought for light water reactors in this research topic.
- ° Research topic 3 on "Improved in Plant Accident Response" is partially addressed by the CANDU design. CANDU reactors are provided with a main control room and an auxiliary control room; the latter is installed in a separate location. Each control room by itself has full capability for reactor shutdown to cold conditions and for providing continuous state of the plant information. Light water reactors also incorporate this design basis provided that there is no equipment damage in either control room. In other words, the light water reactor design basis is for conditions which would make the control room uninhabitable without damaging any of the equipment. However, in the CANDU design, this capability is maintained even if total failure of all equipment in the other control room is assumed. This feature would improve the in plant accident response during an event such

as a fire in one of the control rooms that would damage the control room equipment.

- Research topic 5, "Alternate Emergency Core Cooling Concepts," is intended to decrease the risk associated with current light water reactor emergency core cooling systems. The CANDU design provides redundant means of cooling the core through the emergency core cooling system or through the moderator and its cooling systems. It is very likely that a value/impact analysis of the CANDU design in this area would show that the existing design is quite adequate and meets the intent of the research topic currently being addressed for light water reactors. The full primary system temperature and pressure shutdown cooling system employed in operating CANDU plants is also pertinent to this research topic.
- Research topic 7 is on "Alternate Containment Concepts". The dual failure design approach used in CANDU which demonstrates that the impact of losing the containment capability is acceptable, may meet the objective of this LWR research topic. Although the conceptual design described in this report employs a U.S. type containment, Canadian stations employing multiple units use an alternate containment concept (Vacuum Containment). This alternative might also be responsive to the intent of this research topic.
- Research topic 8 on "Improved Reactor Shutdown Systems" is more than adequately addressed by the CANDU design. In fact, the CANDU design approach goes even further than the intent of research topic 8 by considering the impact of the loss of each of the two independent shutdown systems separately, the independent loss of containment separately, and the independent loss of the emergency core cooling system separately.

- Research topic 10 addresses "Core Retention Measures". It is highly likely that the need for this capability would be removed by the CANDU design which contains several core cooling mechanisms that may very well remove the core melt event from the realm of possibility.

5.5.4 Performance Characteristics of Candu Relative to C-E PWR

5.5.4.1 General Aspects

In this section the performance characteristics of CANDU are discussed. The CANDU design philosophy for control has been to fully automate the plant as much as possible over the entire range of operation. This is done with a large digital computer which performs all of the control functions. There is a duplicate computer on standby ready to take over from the main computer in the event that a failure is detected in the controlling computer. The design philosophy of using the controls as a first line defense for safety related design basis events may also impact performance characteristics by improving plant availability through avoiding reactor trips.

5.5.4.2 Inherent Characteristics

There are several inherent characteristics of the CANDU reactor which impact its performance. First of all, it is necessary to quickly return the CANDU to power after a reactor trip to avoid a xenon poison out situation, because of the limited excess reactivity available to override xenon. The maximum amount of reactivity available for xenon override is $1.5\% \Delta p$ (adjuster rods) + $0.4\% \Delta p$ (liquid zone control) + $0.2\% \Delta p$ (power reduction) = $2.3\% \Delta p$.

As a result of the limited available excess reactivity, the plant must be returned to full power within 30 minutes of the time of a reactor shutdown from full load in order to prevent a xenon poison out occurrence. If the reactor is not returned to power within this time period, it becomes necessary to wait 36 hours before a return to criticality is possible. In addition, if the reactor is suddenly reduced to below approximately 55% power from an equilibrium full load condition, xenon poisoning will shut the

plant down and keep it down for many hours. As a result, the plant has been designed with the capability to rapidly return to full power after a reactor trip. Operating experience shows that whenever spurious reactor trips occur, the plant is normally returned to rated power in approximately 15 minutes. In addition, a poison prevent system is designed to keep the reactor above 60% power during a turbine trip or full load rejection so that xenon poison out will not occur. This system simply consists of a greater than 60% steam bypass to condenser capability, so that the steam generated at these conditions is bypassed to the condenser instead of going to the turbine.

The power coefficient of reactivity in the CANDU reactor is near zero or (in local regions of the core) slightly positive. As a result, it is much easier to change reactor power at a more rapid rate in CANDU than it is in a PWR which has a highly negative power coefficient of reactivity. To illustrate this point, the withdrawal of solid control absorbers by a small amount in the CANDU reactor will cause power to increase on a slightly positive period. To stop the power increase and hold power at the increased level, the solid control absorbers must be re-inserted slightly beyond their initial position. In the PWR, on the other hand, a withdrawal of a bank of control element assemblies will result in an initial power increase followed by a stabilization at a slightly higher power level where the moderator and doppler feedback effects counteract the positive reactivity added by the control rod withdrawal. To increase the power on a ramp, the control rods must be continually withdrawn. It is this maximum rate of reactivity addition which is possible in a PWR which presumably determines the maximum rate at which the power can be increased. For existing PWR designs, this rate is typically limited to a value on the order of 5% of rated power per minute. The design power increase rate for CANDU, which is achieved without imposing difficult control rod withdrawal rate requirements on the design, is 60% of rated power per minute.

Adjuster rods are provided in the CANDU design to shape the power distribution when inserted. When the adjuster rods are out, positive reactivity is added and the core peaking factor is increased. If power is returned toward full load when these rods are withdrawn, a thermal limit will be reached before full load is achieved. The power level which can be obtained is dependent upon the number of adjuster rods withdrawn. The on-line core flux mapping system provides the appropriate information on power distribution to the operator to allow him to monitor the preset limits. This system also provides information to the control system to automatically limit power increases as necessary. The protective (shutdown) systems also sense the power distributions which result when the adjuster rods are out. This is achieved by placing an individual overpower trip setpoint on each in-core detector signal which is supplied to the protective systems. As a result, the protective system automatically shuts down the reactor if any localized region in the core exceeds the appropriate limit for that region.

The frequency of occurrence of spurious reactor trips in the CANDU design is less than one per year on most operating CANDU plants. This very low number is achieved by making extensive use of control actions to cause power setbacks or stepbacks in the event that trip parameters are being approached. This control action steers the plant away from the trip which would otherwise occur, thus significantly improving plant availability.

Because of the sensitivity of the CANDU design to being shutdown for 36 to 48 hours on xenon poisoning after a shutdown, it is very important that the condenser be designed for high reliability. If the condenser system were to be out of service for 20 to 30 minutes or longer, the reactor would be shutdown for approximately 36 hours until the xenon poisoning decayed such that enough

reactivity existed in the core to return to power. A related aspect of this is that it is necessary that the condenser vacuum be maintained during upset conditions in order to avoid a xenon poison out condition. The task of re-establishing vacuum, even if the condenser system is operating properly, is a time consuming procedure which well may take longer than the 20 to 30 minutes available, and thus cause the reactor to be shutdown for a day and a half or so.

5.5.4.3 Performance Capabilities

For discussion of the performance capability of the CANDU design the power operation of the plant has been divided into three major categories. For the first major category, steady state power operation, the CANDU has the capability of operating continuously at any power level up to rated power. This includes operation at a reduced power level indefinitely which may necessitate operation for some hours with some steam being bypassed to the condenser so that the reactor would not be poisoned out on xenon during the period immediately following a reduction in power. In comparison, the xenon peak can always be accommodated by either boron dilution or by permitting the reactor coolant average temperature to drop in a PWR.

The next major category of performance capabilities addressed is the load change capability category. Load change capability is the capability of the unit to respond to external demands placed on the generating station. Implied within this category is the ability to block the imposition of a demand on the plant which the plant is not able to accommodate.

The first load change capability item in this category relates to the ability of the plant to accommodate ramp increases in turbine steam flow. As indicated, the CANDU design has significantly

greater capability to respond to steam flow increases over the power range when not limited by xenon. The main reason for this was explained previously in Section 5.5.4.2.

The CANDU design can also accommodate any type of load rejection from the turbine, up to and including full load rejection or a turbine trip, without a reactor trip. Of course, for the turbine trip case, it is necessary that outside power be available to supply the needed power to the primary heat transport pumps.

For a return-to-power after a reactor shutdown, it is essential that a return to about 60% of rated power be accomplished within approximately 30 minutes if an extended (approximately 36 hours) shutdown on xenon poisoning is to be avoided. As a result of this need, the capability has been provided and demonstrated in CANDU to return to high power in 30 minutes or less after a spurious shutdown.

The return-to-power capability after a turbine shutdown differs from that described above in that the reactor can be maintained at a significant power level thus avoiding the excessive xenon buildup associated with the reactor shutdown. The capability of leaving the reactor at a power level which can be accommodated by the steam bypass to condenser system, is available.

The CANDU system can be tied in to an automatic generation control system at a remotely located load control center. This permits the reactor to be used for load regulation duty by the load dispatchers. An important feature of the design permits this without the need for a operator intervention to keep the load demand within the capability of the reactor. This feature is the incorporation of automatic overrides on the load demand so that the load demand is always maintained within the capability of the reactor. This prevents reactor trips, operating limit violations,

equipment damage, or other adverse effects of inappropriate load demands on the turbine.

The CANDU can respond to turbine speed governor initiated load changes resulting from electrical frequency variations from the 60 Hertz setpoint. Frequency reductions cause rapid load increase and frequency increases cause rapid load decreases through the speed governor. These speed governor actions are necessary to maintain proper control and stability of the electrical grid.

Automatically controlled nuclear heatup is provided with the time required to bring the plant from a cold shutdown condition to rated power considerably reduced compared to a PWR. Controlled nuclear heat is used to supplement the electrical heat produced by the coolant pumps to bring the water temperature up from the cold value to the design condition for power operation. This heat source is particularly effective as the design condition is approached since the rate of heatup with a constant heat source decreases as the temperature rises. This occurs because the heat losses increase as the temperature increases. Furthermore, performing this function under automatic control maximizes the heatup rate because the automatic controls permit the rate to be maintained continually at its maximum permissible value.

Automatic turbine startup programs reduce the time required to bring the turbine from a cold condition to power operation. Automatic turbine startup is achieved by supervising all of the important limits which must be maintained during turbine startup so that the turbine can operate reliably closer to these limits. The net result is a decrease in the time required to bring the plant on-line and an increased level of automation which reduces the possibility of operator error.

The next category of performance capabilities is the high availability category. This category contains performance features of the design which increase the availability of the unit. The main distinction between this category and the load change capability category is that these features all improve the availability of the plant when the load demand on the plant is constant and there are no perturbations on the electrical grid. The features either permit the plant to ride through an internal fault without causing a shutdown, reduce the probability of such faults, or reduce forced outage time.

The first of these item involves the accommodation of expected and unexpected transients without a plant outage or equipment damage. It is a design basis of the CANDU control systems to take strong setback and stepback action as necessary to accommodate the expected transients and many unexpected transients without damaging any equipment or shutting down the plant.

The CANDU design can accommodate any magnitude of turbine load setback or runback without reactor trip. Such setbacks or runbacks are often incorporated into the turbine design for the purpose of keeping the turbine on line during various turbine, generator, or condenser problems. One such example is the loss of stator coolant in the generator. Another such example may be high turbine back pressure due to loss of a condenser circulating water pump. Also the design will permit operation at reduced power with one (of the two) primary heat transport system pump operating per loop. This feature permits the plant to remain available for power generation in the event that one or two pumps fail.

For comparison purposes a list by category of the performance capability of the CANDU design versus the C-E System 80 design is presented in Table 5.5.4-1.

Another important performance parameter is unit availability or capacity factor. The reported capacity factors (Reference 5.5-16) for the Pickering and Bruce units have averaged about 75%, compared to a capacity factor of about 65% for light water reactors operating in the U.S. The capacity factor of U.S. light water reactors, of course, has varied considerably from unit to unit, as has the average availability of light water reactors operated by a number of nations. The higher availability of Candu plants is thus thought to be due partially to inherent attributes of Candu design and partially to external influences (such as differences in licensing and inspection regulations, construction practices, utility load requirements, etc.) which vary from nation to nation. For a U.S. sited HWR, the difference in capacity factor with respect to U.S. LWRs should be attributable primarily to the inherent features of the Candu design, since the external influences on plant capacity factor should be essentially the same for both reactor types.

The major inherent attribute of Candu-type reactors which contributes to improved availability is the on-line refueling process. With on-line refueling, the periodic shutdown required in light water reactors to replace fuel is eliminated. However, it should be noted that maintenance and inspection is also performed during these refueling outages, and periodic outages in the HWR will also be required for this purpose. Consequently, it is difficult to determine the increase in availability which should properly be attributed to the on-line refueling feature of the HWR, since it is hard to assess how much refueling/maintenance outage times or frequency for the LWR would decrease if only maintenance were performed. In practice, some reduction in either the duration or frequency of the refueling/maintenance outage should be anticipated with on-line refueling. Based upon the discussion presented in Section 5.4, a capacity factor 1-3 percentage points higher than that of the LWR is anticipated for the U.S.-sited conceptual HWR.

Another advantage of on-line refueling is that any activity which may be released from the fuel due to fuel damage can be readily detected, and the damaged fuel can be replaced without decreasing the power output of the unit. In a light-water reactor replacing damaged fuel necessitates a refueling shutdown.

5.5.4.4 Impact of Enrichment and Increased Primary Pressure

The impact of enriching the fuel and increasing the primary pressure is not expected to be significant on the performance of the CANDU design. In making this statement, it is assumed that the power coefficient of reactivity and the xenon poisoning characteristics of the core will not be adversely affected by enrichment. It is also assumed that the margins between the nominal primary heat transport system pressure and the control action points, protective system trip points, and primary safety valve setpoints will be at least as large as they are in the present design. Furthermore, it is assumed that there will not be failures of materials or components resulting from the higher primary system pressure.

TABLE 5.5.4-1

PERFORMANCE CAPABILITY COMPARISON

<u>CANDU</u>	<u>C-E SYSTEM 80TM NSSS</u>
Steady State Power Operation	
1. Accommodate steady state operation at any turbine power level up to rated reactor power	Same
Load Change Capability	
2. Accommodate ramp increase in turbine steam flow at 60% per minute between 25% and 75% power 10% per minute between 70% and 100% power except when limited by xenon	Accommodate ramp increase in turbine steam flow at 5% per minute between 15% & 100% power except when limited by xenon
3. Accommodate step turbine steam flow increase of undefined magnitude (probably at least comparable to C-E PWR)	Accommodate step increase in turbine steam flow of 10% of rated between 15% and 100% power inclusive
4. Accommodate ramp decrease in turbine steam flow of any amount at any rate from any initial power level up to and including full load rejection or turbine trip from rated power	Same
5. Return to power after reactor shutdown within approximately 30 minutes or experience xenon poison out delay of approximately 36 hours	Return to power after reactor shutdown as shown in Figures 3.3-2 through 3.3-5
6. Return to power after turbine shutdown within 15 to 30 minutes using "poison prevent" mode of steam bypass to condenser system	Same
7. Daily load cycling capability undefined but probably similar to C-E PWR and not subject to core lifetime restriction	Daily load cycle shown in Figure 3.3-6 up to 90% of core life
8. Follow load demanded directly by remote load control center subject to automatically enforced reactor limits (permits station to be used for electrical grid load regulation duty)	Same when C-E Megawatt Demand Setter is used

TABLE 5.5.4-1 (Con't)

9.	Respond to turbine speed governor initiated load changes resulting from electrical grid frequency upsets	Same
10.	Accommodate automatically controlled nuclear heatup and automatically controlled cooldown	Not available with PWR
11.	Compatible with automatic turbine startup system	Same
High Availability		
12.	Accommodate expected and many unexpected transients without a) reactor trip, or b) equipment damage	Accommodate many expected and some unexpected transients without: a) reactor trip or b) equipment damage
13.	Initiate power stepback on low steam generator level	Accommodate loss of main feedwater pump without reactor trip from any load
14.	Accommodate turbine load setbacks or runbacks without reactor trip	Same
15.	Permit operation at reduced power with at least one primary heat transport system pump operating per loop	Permit operation at reduced power with at least one reactor coolant pump operating per loop
16.	Eliminate need for refueling shutdown via on-line refueling	Refueling shutdown required in LWR designs

5.5.5 Qualitative Description and Licensability Assessment of LOCA Events for HWR

5.5.5.1 Qualitative Description of the LOCA Event for A HWR

Unlike the PWR, where the loss-of-coolant accident imposes the most restrictive limitation on peak linear heat rate, the loss-of-coolant accident is found to be nonlimiting in the Candu reactor. It is the loss of regulation event which limits channel power and peak linear heat rate in Candu (see Section 5.5.6 for discussion of this event). The LOCA transient (leading to the highest fuel clad temperature) in Candu is an intermediate sized break in an inlet header. A typical flow path for the LOCA transient is shown in Figure 5.5.5-1. The reason the intermediate size break LOCA is limiting is that the leakage flow rate on one side of the core equals the pump flow rate on the other side of the core for a short period of time, causing the flow through the core to stagnate. During the stagnant flow period, clad temperatures are controlled by the energy stored in the fuel since heat transfer drops to very low values. The peak clad temperature is not sensitive to ECC injection pressure or to decay power because peak clad temperature occurs early in the transient.

Since the energy removal rate for very small breaks is very low, primary system cool-down via steam generator blowdown is required in the HWR to reduce the pressure of the primary system to the point where emergency core cooling water can be injected. This is accomplished by a safety-grade secondary side steam relief system. There is no significant impact from the positive void coefficient of reactivity during small breaks because the regulation system is capable of introducing negative reactivity at a rate faster than that introduced by voiding.

The clad temperatures due to larger breaks are sensitive to the break location. For large breaks in the HWR, the break discharge

controls primary system depressurization, so that steam generator cooling is not important in the short term. The rate of reactivity addition by voiding is greater than that which can be accommodated by the regulating system and consequently an overpower transient occurs which is quickly terminated by either SDS-1 or SDS-2 which are activated by either high neutron power or high rate neutron power signals provided to each system.

The limiting break size from the standpoint of peak clad temperature is a break of about 30% of the header cross-sectional area in the reactor inlet. The maximum clad temperature occurs early in the transient (6 to 7 seconds) and exceeds 2,200°F for a short period of time (not long enough to cause significant embrittlement). The post blow-down portion of the accident is less limiting (has lower clad temperatures than in the earlier phases of the transient).

5.5.5.2 Loss of Coolant Accident (LOCA) Transients

Two questions are considered in this section:

1. Is a LOCA more likely to occur in a HWR than in a PWR, and if so is the integrated risk (probability/consequence) comparable to that of the PWR?
2. If a LOCA does occur, can safety systems contain the accident within the criteria limits of the regulatory requirements?

The initiating event of a LOCA in a PWR for licensing applications (limiting events for licensing analysis) is arbitrary rather than mechanistic. That is, the plant is designed to withstand all potential causes (e.g., earthquakes) without having a major primary system pipe break. Thus, the design of a HWR should be such that a LOCA cannot occur as a result of a mechanistic initiating event. There are two aspects of the pressure tube HWR which have

no comparative PWR analogy where mechanistic initiating events might be postulated.

The first is the possibility of the occurrence of pressure tube ruptures in the Zircaloy pressure tubes which are a part of the pressure boundary. However, Canadian studies and experience has supported the position that an impending rupture will give adequate warning by leaking. Experiments have shown that several thousand full pressure cycles are required to grow a crack to a critical size and that it will leak before it reaches 1/5 of this size. Furthermore, the HWR design insures that the system pressure and temperature remain within safe limits and includes a surveillance system to recognize when and where a pressure tube leak has occurred. Ultrasonic testing of pressure tubes can also be performed at outages.

Thus, it is concluded that there is no mechanistic means of initiated a design basis LOCA as a result of pressure tube rupture. This of course, would not eliminate consideration of pressure tube ruptures from the LOCA analysis.

A second feature of the design of pressure tube HWRs which could contribute to the probability of initiating a small-scale LOCA is the on-line refueling system. The design basis for the on-line refueling machines is such that only a single power channel can be affected, i.e., the refueling machines are designed so that initiating events (for example, malfunction of the refueling machine control system) cannot result in refueling machine motions which could fail neighboring power channels. Because of the frequent refueling visits, the probability of small-scale loss-of-coolant accidents is increased, although operating experience shows this event to be unlikely. At any event, the consequence of such a small-scale LOCA will be rather limited, compared to the outcome of a postulated large-scale LOCA.

Thus, it is concluded that the use of Zr-Nb pressure tubes and on-line refueling results in postulated LOCA events of low consequence when mechanistic initiating events are considered, and that consequently the integrated risk (probability/consequence) of mechanistic LOCA's are comparable to those of the PWR.

Having postulated that a LOCA has occurred because of some arbitrary initiating event, such as the rupture of an inlet header, the next question to address is whether the accident can be contained within the limits of regulatory requirements. As noted above, Canadian LOCA calculations indicate that the worst LOCA is an intermediate size break in an inlet header. The peak clad temperatures occur early in the transient (before ECCS flow begins) when the pump flow nearly equals the break flow, resulting in a nearly stagnant condition in the core for a short period of time. As has been mentioned, the maximum clad temperature exceeds, for a short period of time, the 2200°F limit imposed by present U.S. regulatory criteria, as given in Appendix K of 10CFR50. Because this limit is exceeded for only a short period of time, it should not lead to significant embrittlement or clad oxidation. The 2200°F limit on clad temperature imposed by Appendix K is viewed as somewhat arbitrary, and the time at temperature criteria imposed by Canadian licensing is viewed as technically correct and sufficient to ensure that extensive fuel damage does not occur during the loss-of-coolant event; however, it must be noted that the Canadian approach is not consistent with the temperature limit currently imposed in the U.S. Since this peak clad temperature occurs early in the transient, prior to the time when ECC can be initiated, the peak clad temperature can be reduced only by reducing the peak linear heat rate. This could be accomplished either by increasing the number of fuel rods in each bundle (i.e., increasing the number of feet of fuel) or by reducing channel power (i.e., by increasing the number of fuel channels).

For later periods in the loss of coolant transient, peak clad temperatures are determined by the performance of the emergency core cooling system. In the PWR, peak clad temperatures often occur during this later phase of the accident (the reflood phase). However, as previously mentioned, Canadian LOCA calculations indicate that the peak clad temperatures which occur during reflood are much lower than those which are calculated early in the transient. As a result, the performance of the ECC system on the HWR is less critical than the performance of the corresponding PWR system. In developing the conceptual HWR, the ECC system has been modified to incorporate the high pressure injection normally employed for PWRs. This should provide improved ECC performance, although Canadian analyses demonstrate the adequacy of the present gravity-fed system employed on operating plants.

It should be pointed out that this qualitative discussion of LOCA for the heavy water reactor has been based totally upon Canadian analyses. Present U.S. regulatory criteria (Appendix K of 10CFR50) mandate a set of requirements and limits which must be followed in the U.S. for evaluation of LOCA transients in PWRs and BWRs, and it is not clear whether the Canadian evaluations incorporate all of the requirements imposed by these regulations. The perceived consequences of a LOCA may be somewhat different if one attempted to analyze the event consistent with the requirements imposed by Appendix K. However, it should be recognized that these requirements came about during the development of U.S. LWRs and cannot be arbitrarily applied to heavy water reactors. For example, items in Appendix K concerning annulus bypass, reflood rates, and heat transfer rates during reflood are not applicable to the pressure tube HWR because they are specific to the design of PWRs. Thus, some modification of the U.S. Appendix K requirements will be necessary, and a reanalysis of the HWR LOCA will be required before one can definitively conclude that the HWR meets the (modified) U.S. regulatory criteria.

5.5.5.3 Qualitative Licensing Assessment of LOCA Event for a HWR

Several features of the HWR which make it significantly different from a PWR in regard to the LOCA transient that will affect the licensability of HWRs in the U.S. are listed below:

- ° The fuel rods are oriented horizontally rather than vertically. There are limited heat transfer data for rods having horizontal flow patterns available in the open literature which can be used for the type of flow regimes expected to occur during a postulated LOCA.
- ° The fuel rods are contained in pressure tubes which make up a core of several parallel, horizontal flow channels. This is in contrast to the open core of a PWR in which all of the fuel rods are oriented vertically in the same flow channel. There are no published experimental data for conditions typical of a LOCA which would verify methods for calculating the flow rates through the various channels given the conditions at either end of the channels.
- ° The void reactivity coefficient is positive. This is due primarily to less resonance capture in U-238 due to decreased neutron scattering in the fuel cluster. A shutdown system (two independent systems are provided) is required to reduce reactor power following a large scale LOCA. This is in contrast to PWRs which have a negative void reactivity coefficient which results in shutdown from voiding should a large break LOCA occur.
- ° The HWR has two coolant circuits through the core. During normal operation the only connections between the circuits are through the pressurizer system and the D₂O purification system. The broken circuit is isolated from the intact

circuit following a LOCA, preventing complete voiding of the intact circuit thus limiting the amount of positive reactivity insertion (this is not a requirement, e.g. a single circuit is employed in Bruce). Loop isolation requires the operation of isolation valves. Some modification of this system may be required for U.S. licensing to provide redundancy in these isolation valves and to the auxiliary feedwater flow to the steam generator required to provide heat removal from the intact loop.

Since the above features are unique to HWRs it is expected that the NRC would probe these areas thoroughly.

5.5.5.4 Data Required to Develop and Verify a LOCA Evaluation Model

The development of the mathematical models necessary for a U.S. evaluation of the performance of the ECCS during the various LOCA transients would require a considerable quantity of experimentally obtained data. These data include:

- ° Heat transfer data applicable during and following blowdown for the whole spectrum of break sizes,
- ° pressure drop versus flow rate characteristics of the flow channels for conditions applicable to LOCA,
- ° parallel channel test results which investigate the possibility of channel to channel variations in flow and the possibility of parallel channel instabilities during LOCA transients, and
- ° system effects test results which would aid in modeling the feedback effects of the rest of the system (steam generators, pumps, etc.) on the flow through the core.

5.5.5.5 Proposal for Future LOCA Evaluation of HWR

It is expected that current PWR LOCA models for the blowdown period of a LOCA could be modified sufficiently to allow scoping calculations to be performed for the blowdown portion of the HWR LOCA transient. A study such as this would be very beneficial since it would give more insight into the nature of the HWR transient and because it is during this period that the Canadians predict the peak clad temperature to occur. Such a study would require about 3.0 man-years and 10 computer hours to complete. These figures might be reduced if many of the necessary input parameters and computer nodding schemes were supplied by the Canadians. It should be noted that this study does not investigate the post-blowdown phase of the LOCA transient. Additional time would be required to develop analytical techniques applicable to this phase of the transient.

FIGURE 5.5.5-1

FLOW STAGNATION IN PART OF CORE DURING EARLY PORTION OF
INTERMEDIATE BREAK TRANSIENT (PUMPS ARE POWERED)

5.5.6 Qualitative Description and Licensability Assessment of Other Postulated Events

Under U.S. licensing practice, postulated events are subdivided into various categories depending upon the anticipated frequency of occurrence. Different acceptance criteria are placed upon each category. In particular, moderate frequency events must meet an acceptance criteria of no fuel damage, while fuel damage limits are placed on infrequent events; different site boundary dose limitations are also imposed for each category. As a result, it is necessary to analyze non-limiting faults in the U.S., in addition to limiting (infrequent) faults, since it must be demonstrated that each category of events meets the appropriate acceptance criteria for that event.

As discussed in Section 5.5.2, Canadian licensing practice requires the analysis of single (the failure of a process system) and dual (the simultaneous failure of a process system and a mitigating safety system) failure. No distinction is made between the severity of various events within the single and dual failure categories. Separate site boundary dose limits are placed on single and dual failures, and each event within the single or dual failure category must meet these dose limitations independent of the probability of the accident. Acceptance criteria are based solely upon site boundary dose, and potential fuel failure during moderately frequent events is not viewed as a safety issue, but is considered solely of economic concern.

As a result of the Canadian licensing philosophy, the single failures which are typically evaluated are various types of limiting faults. Accidents which have less severe consequences than these limiting faults need not be analyzed since they are evaluated against the same acceptance criteria as the more severe events. Typical single failures which are analyzed as part of

the Canadian licensing development are given in Table 5.5.6-1. For each of the events listed in this table, a number of initiating causes may be explored to ensure that the most limiting case in each category is analyzed.

The results of a number of Canadian analyses (Reference 5.5-7) are discussed in subsequent sections of this chapter in order to provide an indication of the types of analyses performed as part of the Canadian licensing evaluations, and of the consequences of these limiting faults.

5.5.6.1 Loss of Regulation

The loss of regulation accident is viewed as the event which limits the core power rating. The loss of regulation event is defined as any occurrence of a loss of control of reactor power causing a power excursion with the potential of releasing significant radioactivity from failed fuel, if not terminated by the protective system. The loss of regulation event can be initiated from a number of control system malfunctions, as tabulated in Table 5.5.6-2. These initiating events essentially involve the malfunction of reactivity control devices, so as to introduce positive reactivity into the core.

Typical control and protective system actions which are taken as a result of a loss of regulation event are illustrated schematically in Figure 5.5.6-1. As local power increases, the control system would first attempt to reduce power through a controlled ramp-down (setback). The next action point encountered would initiate a stepback in which control rods are dropped into the core. The stepback and setback control actions are utilized to avoid the necessity of actions by the reactor protective system. Should these control actions fail, a trip signal would be initiated first in Shutdown System 1, and that failing, subsequently in

Shutdown System 2. Each shutdown system is provided with redundant trip signals from diverse detectors (2 out of 3 logic is employed). The primary trip signal refers to the first trip signal which would normally be activated; should this trip signal be for some reason unavailable, additional trip signals would be generated from other sensors (for example, in the high-power range, rapid transients would first generate a signal from the high rate log neutron power sensor and subsequently from the high power trip). The primary and backup trip signals are shown at the bottom of Figure 5.5.6-1 for fast, intermediate, and slow loss of regulation transients originating from either high- or low-power operation.

It is Canadian practice to analyze transient rates which bracket that possible for a control system malfunction. Figure 5.5.6-2 shows the relative increase in coolant power (heat flux) as a function of time for reactivity increases ranging between 0.5 to 2 mk/sec. This particular transient is from full power, where the high-power trip is the primary shutdown system for intermediate and slow transients and the high rate log neutron power trip is the primary signal for fast transients. In analyzing the loss of regulation event, flux shapes which can occur during normal operation are considered at the initiation of the event.

In order to ensure that the site boundary dose limitations imposed by Canadian acceptance criteria are met, it is sufficient to demonstrate that the pressure boundary is not breached as a result of the event. Prevention of fuel centerline melt is established as a sufficient and conservative criterion which precludes breach of the pressure boundary. Regional flux trip set points are set at approximately 112% of full power, which provides sufficient margin to prevent centerline melt when all uncertainties are added together in a conservative fashion.

As Figure 5.5.6-2 illustrates, for the various loss of regulation ramps considered, the event is terminated before the coolant power (heat flux) reaches the 115% level. The limiting transient is the slowest ramp with the most severe localized power peaking because the effect of the heat flux delay time is eliminated. For this ramp, the coolant power reaches 115% at the point where the 115% flux power trip occurs. No fuel dryout will occur unless there is an adverse combination of uncertainties and/or severe localized peaking not fully measured by the nearest detectors, both of which are highly unlikely; in any case, no fuel centerline melt will occur, thus precluding breach of the pressure boundary.

5.5.6.2 Loss of Load

The loss of load event includes both the conventional loss of turbine load and also the loss of condenser vacuum which is more limiting because steam cannot be bypassed. 100% power capacity secondary side relief valves are provided so that the event can proceed to a temporary stable condition at an elevated inlet temperature corresponding to the secondary relief valve set points even if there were no immediate control or protective system actions. An adequate initial core thermal margin exists such that dry-out in the fuel channel would not occur in this situation. However, the reactor would eventually trip on low boiler level due to a reduction of secondary side water inventory. Enough water is available to provide boiler heat removal for 30 minutes at which time the operator will have provided an alternate heat sink.

5.5.6.3 Loss of Class 4 Power

In the Class 4 loss of power event, power to the primary coolant circulating pumps is lost, resulting in transient undercooling. As a result of the positive coolant coefficient of reactivity, core reactivity and hence power level will increase as a result

of this transient. The result of Canadian calculations of this transient is shown in Figure 5.5.6-3. The top of the figure illustrates the effect of the transient on the power to the coolant (heat flux). As this figure illustrates, heat flux increases only slightly before the reactor power is reduced by control actions (step-back) or by reactor trip.

Trips are provided on high primary system pressure and also on low flow for shutdown system 1. Normally, the transient would be terminated by a trip from shutdown system 1 which occurs at about 3 seconds into the transient and which terminates the transient well before fuel channel dry-out is reached ($MCHF < 1$). Should shutdown system 1 be unavailable (an assumption which must be made under Canadian licensing practice, but not under U.S. licensing practice), a trip would occur between 4 and 5 seconds via shutdown system 2; in this case transient dryout could occur.

5.5.6.4 Pump Seizure

The Canadian analysis of the pump seizure event is illustrated in Figure 5.5.6-4. For the case illustrated, it has been assumed that shutdown system 1 was unavailable (an assumption which need not be made under U.S. licensing criteria). Since the pump is presumed to seize in 2 seconds, neither shutdown system will alter the short term flow response, and, as a result, transient dry-out occurs. Either shutdown system, however, will prevent fuel centerline melting.

TABLE 5.5.6-1

TYPICAL SINGLE FAILURES

- Fuel Handling Accident
- Loss of Regulation
- PHT Pipe Breaks
- Secondary Side Pipe Breaks
- Loss of Class IV Power
- Pump Seizure
- Loss of Instrument Air
- Loss of Pressure Control

TABLE 5.5.6-2
LOSS OF REGULATION EVENT

"An occurrence of a loss of control of reactor power causing a power excursion with the potential to lead to a significant release of radioactivity from failed fuel, if not terminated by the protective system."

- (a) Failures in the computer system hardware and software causing uncontrolled power increase.
- (b) Liquid zone failures causing draining of the zones, when fast moderator level lowering is unavailable.
- (c) Spurious out-drive of one or more adjuster rods, when fast moderator level lowering is unavailable.
- (d) Uncontrolled moderator level increase.
- (e) Insufficient moderator poison when the moderator level lowering is unavailable.
- (f) Incorrect withdrawal of one or more banks of shut-off rods.
- (g) Man-machine interface failures including errors by operators, maintainers, computer programmers and other personnel.
- (h) Reactor power measurement failures.
- (i) Power supply failures.
- (j) Control center equipment failures.
- (k) Start-up instrumentation failures.

Reference:

S. Basu, F. J. Kee, "Reliability Study of Reactor Control System for a Candu-PHW Nuclear Unit," presented at the IEEE/ASME/ASCE Joint Power Generation Conference, Los Angeles, CA, September 18-21, 1977.

FIGURE 5.5.6-1
LOSS OF REGULATION

LOR-HIGH POWER

FAST
INTERMEDIATE
SLOW

HIGH RATE LOG N, HIGH POWER
HIGH POWER, HIGH RATE LOG N
HIGH POWER (LOCAL), MANUAL

LOR-LOW POWER

FAST
INTERMEDIATE
SLOW

HIGH RATE LOG N, HIGH P
HIGH P, HIGH POWER (LOCAL)
HIGH POWER (LOCAL), MANUAL

FIGURE 5.5.6-2
LOSS OF REGULATION TERMINATED BY SHUTOFF RODS

FIGURE 5.5.6-3
COMPLETE LOSS OF CLASS IV POWER

FIGURE 5.5.6-4
PUMP SEIZURE

Seismic design criteria applicable to US nuclear power plants fall into two categories: Regulatory Standards which are obligatory and Regulatory Guides which are recommended but not mandatory. All criteria are based upon regulation 10CFR50 which states that no person shall manufacture, acquire, or use any nuclear power facility without a license issued by the Nuclear Regulatory Commission (NRC). Appendix A of that regulation requires that all safety related equipment associated with such a plant, be designed to remain functional following a "Safe Shutdown Earthquake" and that the plant itself be designed to operate during and after an "Operating Basis Earthquake" (OBE). Appendix A of regulation 10CFR100 defines an SSE as the maximum credible earthquake that could occur at the site of a proposed nuclear plant and an OBE as the most severe seismic event that is apt to occur during the plant's forty year life. Also described is the site evaluation work that is required to quantify both.

The methods used to assure that safety related equipment (seismic category 1) will survive an SSE and that the plant will remain operational through an OBE can consist of a suitable dynamic analysis or a seismic qualification test. The previously mentioned regulatory guides provide guidance as to what equipment is safety related, what instrumentation is required in a test, the required shape of the SSE and OBE response spectra, the allowable damping values associated with different equipment and what load combinations must be considered when interpreting results. The design and seismic analysis of nuclear facilities is also guided by a series of codes and standards developed by ANSI, ASME, ASCE AND AISC and other industry codes, although, at present the ASME Boiler and Pressure Vessel Code is one of the few independent codes devoting special consideration to the problems of nuclear power plants. The above regulations, guides, and codes are carefully worded to

insure their applicability to most types of nuclear facilities including an HWR.

Seismic requirements for systems and structures will be identical for the HWR as for a LWR. With the exception of the reactor assembly, the two concepts are similar enough that current design practice used in the USA should satisfy all licensing requirements.

With respect to the reactor assembly the analytical techniques required to satisfy seismic requirements have probably already been developed in this country for use with LWR's. Those techniques will be applicable, with minor modification, to an HWR provided that the mathematical modeling reflects its different physical characteristics. One of the most important of these differences lies in the fuel's lateral restraint. In an LWR, a fuel assembly is vertical, its weight acts along its axis and any motion less than the surrounding gap is resisted solely by its small lateral stiffness. When the vibratory amplitude becomes sufficient to close this gap, contact is made with an adjacent bundle, and so on, until a peripheral bundle makes contact with the comparatively stiff core shroud. Thus, restraint forces originating in the core shroud must act through intervening fuel assemblies. This interaction couples the fuel assemblies across the core, and increases the complexity of the mathematical model required for their analysis. In an HWR, the fuel assemblies are horizontal and reside in individual pressure tubes which are in turn contained within unpressurized calandria tubes spaced approximately 6 inches apart.

The seismic response of an HWR fuel channel will differ from that of an LWR fuel bundle because of the former's lack of dynamic coupling and because of the horizontal position of its fuel inside the pressure tubes. The effect of the former will be to prevent the interstitial gaps from stacking up and causing high impact loads on the peripheral fuel assemblies as commonly occurs in an LWR. Also, because of the fuel's horizontal position, its

weight will tend to prevent free motion across the available gap, further limiting impact loads. Since the preload due to gravity may be sufficient to eliminate impacting entirely, a response spectrum analysis could be sufficient to satisfy the requirements of 10CFR50. This type of analysis involves very little computer time and a comparatively simple model but usually cannot be used for the final seismic analysis of an LWR because of the complex interactions that occur within its core. Instead, it is used to supply preliminary design loadings which are later refined with a nonlinear time history analysis. Whether or not a more detailed nonlinear analysis is required in the case of an HWR can be determined only by examining the results of a response spectrum analysis to determine whether channel-to-channel interactions are predicted for the more severe seismic demands imposed by U.S. geology.

In light of the low seismic impact loads that are expected on its fuel combined with the small bending moments that will be developed in its comparatively short bundles, it should not be difficult for an HWR to satisfy NRC requirements for fuel integrity. Among these are requirements that the fuel assemblies remain coolable after an SSE and that they not leak fission products into the primary coolant following an OBE. Even if the latter were to occur, the on-line refueling capability of an HWR as well as its capability for allowing detection of the flow channels containing damaged fuel assemblies would allow their replacement without a plant shutdown.

There is one major difference between the HWR and LWR that could complicate licensing the former in the US. The combined assembly of pressure tubes and calandria tubes is quite long and will have a natural frequency, when full of fuel, of from 2 to 3 cps. That frequency range is quite close to the peak of the design response spectrum of NRC Regulatory Guide 1.60, and so substantial seismic loads can be expected. Because the pressure tubes are made of Zircaloy and are exposed to the full neutron flux, some embrittlement

can be expected. Analysis will be needed to demonstrate that the stresses caused by an earthquake (based upon U.S. geology) will be acceptable.

5.5.8 Determination of Core Physics Parameters

The response of a reactor to various transients and postulated accidents depends upon the inherent response of the reactor core (i.e., the response to changes in such parameters as power level, coolant temperature, or coolant void content) and of the overall design of the reactor plant. The plant design must be such as to accommodate the inherent transient performance of the reactor core, i.e., to limit the consequences of various transients and postulated accidents to within acceptance criteria. The assessment of whether overall plant response leads to acceptable consequences of various transients and postulated accidents requires the detailed modeling of numerous postulated events. Such an analysis, of course, has been performed as part of the Canadian licensing development for the natural uranium-fueled CANDU-PHWR, and some attention has also been devoted to the implications of plutonium fueling (Ref. 5.5-8). Plant responses for these fueling modes have, of course, been found acceptable with respect to Canadian licensing requirements, and as discussed in Section 5.5.2 should also prove acceptable in the context of U.S. licensing. However, it is possible that the fueling modes contemplated for the conceptual HWR, the use of slightly-enriched uranium fuel and thorium-bearing fuels, could change the inherent response of the reactor core sufficiently that the consequences of various accidents and postulated transients would become unacceptable without changes to the reactor plant design. The increased primary system pressure (and temperatures) incorporated into the conceptual HWR design might also alter the transient response of the reactor core.

A complete evaluation of the effect of slightly enriched and thorium fueling, and of the effect of increased primary pressures and temperatures would require an exacting analysis of all possible transients and postulated accidents, such as has been performed for uranium fueling; such an extensive, qualitative analysis is

beyond the scope of the present study. However, an understanding of the likely acceptability of these design differences can be obtained through a review of the safety-related core physics parameters. Should these core physics parameters prove to be no more adverse than those of natural uranium fueling in the standard CANDU design, one can conclude the consequences of various transients and postulated accidents will also be no more adverse. Should these core physics parameters prove to be more adverse, then detailed and quantitative analyses will be required to establish definitively whether or not the consequences of various accidents remain within the bounds of acceptance criteria and/or to identify changes in plant design which will make the consequences of these postulated events acceptable. Of prime concern are the fuel temperature and void coefficients of reactivity, since these coefficients influence the transient performance of the reactor core.

The fuel temperature coefficient can have an influence on reactor transients which result in an increase in core power, although this influence is typically small in CANDU because the magnitude of the fuel temperature coefficient is small. As core power increases, fuel temperature increases. If the fuel temperature coefficient is negative, increases in core power (and hence of fuel temperature) will decrease core reactivity and hence serve to help terminate the overpower event; overpower transients will be terminated more rapidly, with lower peak transient power densities, the more negative the fuel temperature coefficient. When the fuel temperature coefficient is positive, the increase in fuel temperature which results from an increase in power will add additional reactivity to the core and hence contribute to further increases in core power; with a positive fuel temperature coefficient, other feedback mechanisms or the reactor protection system must be relied upon to terminate the transient. If design on fueling changes result in fuel temperature coefficients which

are strongly positive, more severe demands will be placed on the control and protective systems, while more negative fuel temperature coefficients should result in accident consequences which are less severe than those predicted for the natural uranium fueled system.

Fuel temperature coefficients for the slightly enriched conceptual HWR are compared with those of natural uranium fueling in Figure 5.5.8-1. This figure illustrates that the fuel temperature coefficients throughout life are slightly more negative with slightly-enriched fueling in the conceptual design than for natural uranium fueling in the reference CANDU PHWR. Both the use of slightly enriched fuel and the higher primary system pressure (and coolant temperatures) are seen to contribute toward making the fuel temperature coefficient more negative. For both the natural uranium CANDU-PHWR and for the slightly-enriched conceptual HWR, the fuel temperature coefficient is initially negative, but becomes progressively more positive with irradiation; for both systems, the coefficient becomes positive after approximately two-thirds of the achievable discharge burnup has been obtained.

While the positive coefficient at high burnups may be important in evaluating local power distribution effects, the core performance is determined by the average coefficient which is approximated by the coefficient at the core average burnup (one-half of the discharge exposure). The values of the fuel temperature coefficient at the core average exposure are summarized in Table 5.5.8-1. As this table illustrates, the fuel temperature coefficient for the natural-uranium-fueled system is negative, thus ensuring that the overall response of the core to increases in fuel temperature is self-limiting, i.e., that core reactivity and hence power level will decrease as fuel temperature is increased. In the case of the slightly enriched uranium-fueled conceptual HWR, the fuel temperature coefficient at the core average burnup is more negative

than that for uranium fueling in the CANDU-PHWR. Thus, it can be concluded that the fuel temperature coefficient of the slightly enriched conceptual HWR is such as to result in less severe transients than that which has been established and found acceptable for the natural uranium fueled CANDU-PHWR.

Figure 5.5.8-2 provides similar information for thorium fueling in the conceptual HWR. Included are the three thorium fuel cycles of interest: denatured U-235, denatured U-233, and the plutonium enrichment of thorium. As Figure 5.5.8-2 indicates, fuel temperature coefficients of reactivity for thorium fuel remain negative throughout the irradiation lifetime. Consequently, it can be concluded that, from the standpoint of fuel temperature coefficients of reactivity, the use of thorium fuel should result in less severe overpower transients than are predicted and found acceptable for natural uranium fueling.

The second core physics parameter which has an important impact on transient analysis is the void, or coolant density, coefficient of reactivity. Voiding of coolant channels can occur as a result of overpower transients, underflow transients, or as the result of a loss of coolant accident. When the void coefficient of reactivity is positive, transients which introduce voiding in the coolant channels will increase core reactivity, thereby increasing the core power level which may result in still further voiding and reactivity additions. The effect of a positive coolant coefficient of reactivity is partially compensated for by the negative fuel temperature coefficient. The reactor protection system must be designed so as to terminate overpower transients sufficiently rapidly so as to result in acceptable consequences of postulated accidents. If the slightly enriched uranium or thorium fueling, or the higher primary system pressures and temperatures, contemplated for the conceptual HWR design result in a more positive coolant coefficient of reactivity, greater

demands will be placed upon the prompt action of the protective system.

The effect of coolant voiding on core reactivity is shown in Figure 5.5.8-3, where the reactivity gain due to complete coolant voiding is compared for natural uranium fueling in the standard CANDU-PHWR (1600 psia) with slightly enriched uranium fueling in the conceptual HWR (2250 psia) as a function of exposure. The reactivity gain shown in this figure is for complete voiding of the core. The amount of reactivity addition is approximately linear with void content, as is illustrated in Figure 5.5.8-4. As these figures illustrate, the coolant coefficient of reactivity is positive, both for slightly enriched and natural uranium fueling (i.e., voiding causes an increase in core reactivity).

As Figure 5.5.8-3 illustrates, the amount of reactivity added by voiding decreases with fuel irradiation. At low burnups, the void coefficient of reactivity for the natural uranium fueled system is more positive than that for slightly enriched fueling in the conceptual HWR, but for burnups greater than about 2,500 MWd/t the opposite proves true. Since the largest reactivity insertions occur for fresh fuel, and the reactivity insertion at low burnups is greater for natural uranium than for the slightly enriched system, local power distribution perturbations which might result from voiding should be no more severe in the slightly enriched conceptual HWR than in the natural uranium fueled CANDU PHWR. More important is the core averaged coolant coefficient of reactivity, which can be approximated as the coefficient at the core average burnup. The amount of reactivity inserted due to complete coolant voiding at the core averaged burnup in the natural uranium and slightly enriched systems is compared in Table 5.5.8-2. As this table illustrates, reactivity insertions are comparable (slightly less for 1.2 w/o fueling of the conceptual HWR than for natural uranium fueling of the CANDU-PHWR).

Reactivity inserted upon voiding in thorium-fueled systems is shown in Figure 5.5.8-5. The trends with exposure are somewhat different than for uranium fueling, with the amount of reactivity inserted upon voiding increasing as exposure increases; however, even at high exposures the amount of reactivity inserted due to coolant voiding is less than that of the uranium-fueled systems. Consequently, it can be concluded that the effects of coolant voiding on the consequences of various transients and postulated accidents will be less severe than would occur if natural uranium fueling were employed.

The thorium fuel cycle also differs from the uranium cycle in that fissile U-233 builds up after shutdown from the decay of protoactinium. The resulting increase in reactivity could necessitate modifications to the regulating systems, but is not expected to alter significantly the consequences of various postulated accidents.

In summary, the use of slightly enriched or thorium fueling and the higher primary system pressure and temperatures contemplated for the conceptual HWR design do not adversely alter the coefficients of reactivity which are important parameters in establishing the consequences of various transients and postulated accidents. The standard CANDU-PHWR, using natural uranium fueling, has been the subject of extensive Canadian safety evaluations to demonstrate the adequacy of CANDU design to limit transients and postulated accidents to acceptable consequences. The impact of design changes--fuel type and primary system pressure and temperature--which can impact transient reactor response were evaluated, and in each case found to be less adverse than the corresponding parameters for natural uranium fueling. Consequently it is concluded that the use of slightly-enriched fueling, the use of thorium-bearing fuels, and the higher primary pressures and operating temperatures incorporated into the reference HWR design will not result in a more adverse situation with respect to the

performance of the HWR during transients and postulated accidents than is presently predicted for natural uranium fueling in the CANDU-PHWR. More detailed evaluation of the transient performance of the conceptual HWR employing slightly enriched or thorium fueling, however, should be performed to substantiate the conclusions of this preliminary evaluation.

TABLE 5.5.8-1

CORE AVERAGE FUEL TEMPERATURE COEFFICIENTS

Natural Uranium Fueled Candu	- $0.11 \times 10^{-5} \Delta\rho/^{\circ}\text{F}$
1.2 w/o Fueled Conceptual HWR	- $0.22 \times 10^{-5} \Delta\rho/^{\circ}\text{F}$

TABLE 5.5.8-2

REACTIVITY INSERTED UPON COMPLETE COOLANT VOIDING

Natural Uranium Fueled Candu	+ 1.37% $\Delta\rho$
1.2 w/o Fueled Conceptual HWR	+ 1.31% $\Delta\rho$

FIGURE 5.5.8-1
FUEL TEMPERATURE COEFFICIENTS FOR H.W. LATTICES
(UO_2 FUELING)

FIGURE 5.5.8-2
FUEL TEMPERATURE COEFFICIENTS FOR ThO_2 HWRS.

FIGURE 5.5.8-3
REACTIVITY GAIN FOR COMPLETE COOLANT VOIDING

FIGURE 5.5.8-4
REACTIVITY AS A FUNCTION OF VOID FRACTION
1.2 W/O H.W. 2200 PSIA AT 50 MWD/MT

5.5.9 Safety & Licensing Implications of Design Changes

A number of modifications to existing Candu-PHWR design have been incorporated into the conceptual HWR design to reduce U_3O_8 requirements and/or plant capital costs. The major changes incorporated into the conceptual HWR design with respect to the standard Candu 600 are:

- ° A net electrical output of 1260 MW(e), as compared to 600 MW(e) for the standard Candu design.
- ° The use of slightly-enriched (1.2 w/o) uranium fuel rather than natural uranium.
- ° Primary system coolant pressure of 2250 psi, rather than 1600 psi, in order to increase net electrical efficiency.
- ° A two-loop primary system configuration incorporating 4 steam generators and 4 primary coolant pumps, in contrast to the three-loop configuration which has typified large Candu plant conceptual designs.
- ° A D_2O reflector thickness of 33 cm, about half the thickness employed in the standard Candu design.
- ° An increase in D_2O moderator temperature from 160 to 200°F to facilitate moderator cooling with mechanical draft cooling towers.

The purpose of this section is to briefly review safety and licensing implications of these design changes with respect to the safety and licensability of the standard Candu-PHWR.

° 1260 MW(e) Plant Size

The conceptual HWR design is predicated upon a plant having a thermal output of 4029 MW(t) and a net electrical output of 1260 MW(e). The largest Candu plants currently in operation are the 791 MW(e) units of the Bruce Station; 850 MW(e) plants are also under construction at the Darlington Station.

It is anticipated that the increase in electrical output to 1260 MW(e) can be readily attained. In the pressure tube HWR, the increase in capacity is obtained by adding additional pressure tubes (fuel channels). The calandria tank size must also be increased to accommodate the additional fuel channels, but since the calandria tank operates at near atmospheric pressure, expansion of tank size should result in no unusual design problems (this contrasts to increasing the size of LWRs, where the technology for manufacturing massive pressure vessels of increased size must be developed to increase station capacity). Other components, such as the steam generators, have been manufactured in the required size for LWRs, however, pump size exceeds that yet manufactured for U.S. LWRs.

The two main safety and licensing implications of the larger capacity of the conceptual HWR are expected to be the impact of core size on reactor power distribution stability and the effect of larger components on the loss of coolant accident.

Because of the larger reactor size, the reactor core will be more unstable with respect to xenon oscillations than the current size of Candu reactors. In order to ensure that design limits are not exceeded, the HWR is provided with an automatic system (via the in-core instrumentation, plant computer and zonal control system) to monitor and adjust

local power distributions. Separate in-core instrumentation is also utilized to provide a local high-power density trip on each of the two shutdown systems, to shut down the reactor in the event that the control system malfunctions. Consequently, the HWR is provided with a system both to stabilize the core against xenon oscillations, and to trip the plant should such control actions fail. Although increasing plant capacity to 1260 MW(e) may place additional demands on the design of the control system, it should introduce no fundamentally new considerations to control system design. It should also be noted that the use of slightly enriched fueling in the conceptual design will have a stabilizing effect.

The second area of potential safety and licensing impact is the effect of the larger plant components upon the loss-of-coolant accident. As a result of the increase in plant capacity to 1260 MW(e), the size of inlet and outlet headers and of primary circuit piping will increase. Consequently, the size of the postulated break area will increase, increasing the rate of primary coolant discharge during the loss-of-coolant accident. However, as discussed in Section 5.5.5. (Qualitative Description and Licensability Assessment of LOCA Event for the HWR) the limiting size break from the standpoint of peak clad temperature is not the large break, but rather a break about 30% of the largest postulated break area. The larger rate of primary coolant discharge during the full size break has been included in the containment design. Consequently, the use of larger primary components should not adversely affect the LOCA analysis.

o Slightly-Enriched Uranium Fuel

The conceptual HWR design employs slightly-enriched (1.2 w/o) uranium fuel rather than the natural uranium currently utilized in Candu reactors. The use of slightly enriched

uranium fueling could potentially impact safety and licensability for a number of reasons, such as the effect of the higher discharge burnup on fuel integrity, changes in the coefficients of reactivity and power peaking, the effect of the increased burnup on fission product inventory, and the impact of the use of slightly enriched fuel on criticality during fuel handling.

The use of slightly enriched fuel will result in a discharge burnup of about 20,000 MWd/t, compared to the 7,500 MWd/t burnup currently being obtained in natural uranium fueled Candu reactors, raising the question of whether fuel performance will remain satisfactory at these higher discharge exposures. The evaluation of fuel rod design, reported in Section 4.8, indicates that fuel of the present Candu design should be capable of achieving the 20,000 MWd/t discharge burnups anticipated in the conceptual HWR design. Test irradiations to burnups of about 20,000 MWd/t have also been performed in Canadian test reactors, lending further support to the supposition that fuel performance to these burnups will be acceptable. The question of failure rate of Candu-type fuel irradiated to these high burnups is viewed as primarily one of economic significance, rather than a safety and licensing issue, since the pressure tube configuration and on-line refueling capability of the HWR allow failed fuel to be readily detected and removed from the reactor. Because of the economic implications of premature fuel failure, a more extensive program to demonstrate the performance of Candu-type fuel to burnups of 20,000 MWd/t would be required, possibly performed in Canadian test reactors, as part of the development program to commercialize the conceptual HWR design. It is anticipated that such a demonstration would confirm the analytical and existing experimental evidence that the current fuel design is suitable to burnups of

20,000 MWd/t; at any rate, it is evident that fuel can be designed to attain these burnups, since much higher discharge exposures have been achieved in the LWR.

The use of slightly-enriched fueling (or of thorium fueling) could change the coefficients of reactivity and hence the consequences of various postulated accidents. These coefficients were reviewed in Section 5.5.8 and found to be comparable to those of natural uranium fueling, and hence are believed to introduce no new safety and licensing issues. The effect of slightly-enriched fueling on power peaking was discussed in Section 4.1. As noted in that section, the effect of using slightly-enriched fuel is to increase core peaking. This increase in core peaking can be partially accommodated by employing local enrichment zoning (cf Section 4.6). The remainder of the core peaking is accommodated by increasing the number of fuel channels (i.e., of reducing the average power density so as to maintain equivalent peak power densities). In this fashion, the use of slightly enriched fueling should not adversely impact the thermal margins inherent in the standard Candu design. However, more detailed calculations, which model the complete reactor core, would be required to definitively assess the impact of slightly-enriched fueling on core power distributions and upon the coefficients of reactivity.

As a result of the higher discharge exposure employed in the conceptual HWR design, the fission product burden present in the discharge fuel will be somewhat larger, and contain a somewhat different isotopic distribution, than that of natural uranium. For long decay periods, decay heat generation will be somewhat greater than for natural fuel, and the radiation dose will also be somewhat greater. Radiation levels and decay heat generation will, however, be somewhat

smaller than for the more highly irradiated PWR fuel, and so no unique fuel handling considerations are introduced. Immediately after discharge, the decay heat generation rate is comparable to that of natural uranium, since for periods shortly after discharge decay heat generation is dominated by the short-lived isotopes which reach equilibrium concentrations early in the irradiation cycle. Consequently, no modifications to the refueling machines to accommodate the decay heat (or radiation fields) of the higher discharge exposure fuel is thought necessary; however, should more detailed engineering analyses dictate some change in refueling machine design, such changes would appear straightforward.

Unlike natural uranium, slightly enriched (1.2 w/o) fuel can become critical in light water when arranged in optimal fuel arrays. Fresh fuel for LWRs is normally stored in dry storage facilities. Storage facilities for the 1.2 w/o HWR fuel would be similar; the design, however, could be much simpler because of the lower fuel enrichment. Criticality safety in the refueling machines should pose no problem because of the limited quantity of fuel contained in the machines and the massive amounts of steel present; relatively simple analyses should be sufficient to substantiate this conclusion. Although spent fuel would be clearly subcritical under any conditions of moderation, it will probably be necessary to design the spent fuel storage facility under the assumption that fresh fuel was inadvertently discharged to this facility. Storage racks of relatively simple design should be sufficient to guarantee that criticality cannot occur even if the spent fuel racks were inadvertently loaded with fresh fuel.

° Increased Primary System Pressure

The conceptual HWR design employs a higher primary coolant pressure (2250 psi) than does the conventional Candu design (1600 psi). Primary temperatures, and secondary pressures and temperatures are also increased. The higher secondary temperature and pressure would not engender new safety and licensability issues; since these pressures and temperatures are comparable with those currently employed in PWRs, and since secondary side energy releases were considered in the containment evaluation (cf Section 5.5.10). The effect of the higher primary temperature and pressure on the coefficients of reactivity were also evaluated (cf Section 5.5.8) and found to have no adverse impact. Of more concern is the implication of the higher temperatures and pressures on pressure tube integrity and upon the potential for pressure tube to pressure tube failure propagation during LOCA.

As discussed in Section 4.2, pressure tube wall thickness has been increased from 4.34 mm to 5.79 mm in order to accommodate the increase in primary system pressure and temperature. The modified pressure tube wall thickness of the conceptual HWR was selected so as to maintain the same hoop stress (i.e., strength or resistance to creep) as is present in the standard Candu design. As noted in Section 4.2, the materials data base for irradiated Zr-Nb tubing at the temperatures and pressures of the conceptual HWR is inadequate. Consequently, additional experimental information will be required to verify whether the selected pressure tube thickness does indeed provide comparable strength. It should be noted that the wall thickness of the standard Candu design has been increased by about 25% to accommodate the increase in primary system pressure, and that this increase in pressure tube wall thickness results in a relatively

small (about 5%) reduction in achievable discharge burnup when slightly enriched uranium fueling is employed. Thus, modest increases in pressure tube wall thickness, should they be necessitated by the results of proposed pressure tube materials evaluation programs, could probably be accommodated without seriously compromising uranium resource utilization.

The possibility of pressure tube to pressure tube failure propagation during a rupture of a pressure tube (LOCA) is considered to be insignificant in the standard Candu design, since the catastrophic pressure tube failure which might be postulated to lead to failure propagation is precluded by the fracture mechanics of the thinwalled pressure tube, and experiments have been performed which demonstrate that failure propagation will not occur even in the unlikely event of a catastrophic pressure tube rupture (Reference 5.5-9). Pressure tubes are 100% inspected before installation to ensure that detectable faults are not present, and pressure tubes are subject to periodic in-service inspection which is capable of establishing that any developing faults or faults caused by refueling are much smaller than the critical crack size. Because of the thin wall thickness of the pressure tubes, incipient faults will propagate through the pressure tube, and consequently leak, before the length of the crack approaches the critical crack size. Leaking of the pressure tube is detected by the surveillance system which monitors the gas in the annulus between the pressure tube and calandria tube, providing ample warning of impending pressure tube failure.

Although the thickness of the pressure tube has been increased in the conceptual HWR design to accommodate the higher primary system pressure, pressure tube wall thickness remains much smaller than the critical crack length. Consequently,

incipient pressure tube failures will also result initially in detectable coolant leakage, rather than catastrophic tube failure, thereby also eliminating the mechanism which might be postulated to lead to failure propagation. In order to provide increased confidence that pressure tube failure propagation will not occur, it may be desirable to extend the failure propagation experiments mentioned above to the primary system pressures and temperatures proposed for the conceptual HWR design.

The higher primary system pressure and the increased pressure tube wall thickness of the conceptual HWR design will also require some extension of the technology for forming the rolled joint which mechanically fastens the pressure tubes to their end fitting. No fundamental problems are anticipated with the rolled joint in the conceptual HWR design, but it is recognized that the tooling to form the rolled joint from the thicker pressure tube material must be developed and the integrity of the rolled joint at the higher primary system pressures demonstrated. The higher primary system pressure will also be a design basis for all other parts of the pressure boundary, such as primary system piping and refueling machines. Redesign of some components, such as the refueling machine, will be required to ensure that they can accommodate the increased primary system pressure. Such redesign should be readily accomplished in the context of normal engineering, and should not require technological advances or raise new safety and licensability issues.

° Primary System Configuration

The conceptual HWR design employs a two-loop primary system configuration incorporating four steam generators and four primary coolant pumps, in contrast to the three-loop configuration which has been considered in conceptual large Candu

plant designs. The latter approach has the advantage of allowing net generating capacity to be increased to about 1200 MW(e) through the addition of a third loop containing components of proven design (i.e. the same size components as are currently used in Candu reactors.) Discussions with the Canadians (Reference 5.5-7) indicate that the three-loop configuration is preferred primarily for the reason of standardization; two-loop configurations have also been evaluated and no fundamental safety and licensing problems have been identified. Because of the increase in primary system pressure of the conceptual HWR design, equipment redesign would be required even if a three-loop configuration were employed. In view of the necessity of modifying equipment to accommodate the higher primary system pressure, a two-loop configuration was selected since this configuration allows containment size and equipment cost to be reduced, thereby reducing the plant capital costs.

Although the size of the steam generators are larger than those currently employed in Candu reactors, steam generator size is smaller than those of LWRs. Detailed designs will have to be developed for the conceptual HWR steam generators, but no fundamental problems are anticipated. The primary coolant pumps are larger than any yet constructed. Here too, a development program will be required, but no fundamental problems are anticipated. The primary safety and licensing impact of the use of a two-loop, rather than a three-loop, configuration is the increased energy deposition to containment during the loss-of-coolant accident. This aspect is discussed in Section 5.5.10.

° Moderator Temperature

The moderator temperature of the conceptual HWR design has been increased to 93°C, as compared to the 71°C temperature of the standard Candu plant, to facilitate moderator cooling using the mechanical draft cooling towers of the conceptual HWR. In Canadian licensing practice, it is necessary to evaluate the consequences of accidents in which a simultaneous failure of a safety system is assumed. In particular, it is necessary to evaluate the loss-of-coolant accident (LOCA) when the ECCS is unavailable. In this event, credit is taken for heat transfer from the fuel to the moderator in order to limit fuel damage to acceptable levels. Increasing the moderator temperature might consequently be unacceptable under the Canadian licensing approach because of the adverse effect on heat transfer to the moderator when a simultaneous LOCA and ECCS failure is assumed.

In the conceptual HWR design, the Candu ECC system has been replaced with one similar to that used in PWRs. In particular, redundant trains of high pressure safety injection pumps have been included in the conceptual design. Since this ECC system meets U.S. design criteria, there is no necessity of assuming simultaneous ECCS failure (the assumption of simultaneous ECCS failure is not made for LWRs). Thus, under U.S. licensing criteria the increase in moderator temperature should pose no safety and licensability issue.

° D₂O Reflector Thickness

The conceptual HWR employs a D₂O reflector thickness of about 33 cm, about half the thickness employed in the standard

Candu design, in order to reduce D_2O inventory. There is no obvious safety and licensing implication to this design change. It should be noted that the D_2O reflector also serves as part of the biological shield. The thickness of the light water reflector which surrounds the D_2O reflector has been increased by a corresponding amount, so as to obtain an equivalent degree of shielding.

5.5.10 Containment Evaluation

The containment design and arrangement for the conceptual HWR as described in this report was established by the optimization study presented in Section 4.5. The optimization study concluded that a four steam generator - two loop arrangement would result in reducing capital cost as compared to a three-loop arrangement.

The resulting containment is a reinforced concrete cylindrical structure with a hemispherical dome and a flat reinforced concrete base. The reinforced concrete base is 165 feet diameter, 10-feet thick, and founded 34 feet below finished grade. The upright cylindrical portion of the containment has an inside diameter of 146 feet, measures 183 feet - 6 inches from the top of the foundation mat to the springline of the dome, and has a 4 foot 6 inch thick wall. The dome portion is a hemispherical-shaped head having an inside height of 73 feet, and a 3 foot 6 inch thickness of reinforced concrete. The inside height from the top of the mat to the inside of the dome is 256 feet, 6 inches. The gross volume of the containment is 3,174,000 cubic feet with a free volume of 2,700,000 cubic feet.

To evaluate the design conditions for the containment an evaluation was made of the mass/energy releases to the containment during a loss-of-coolant accident and for a main steamline break. The results of these evaluations indicated that the loss of coolant accident was more limiting with respect to containment design releasing more than twice the energy to the containment, 489×10^6 Btu compared to 203×10^6 Btu for the steamline break accident.

The loss of coolant accident (LOCA) mass/energy release data for containment sizing was generated using the following methodology. The conservatisms include: (1) no credit for flow resistance from the affected primary loop to the break; (2) no credit for

steam binding; (3) no credit for condensation of steam by ECCS flow; and (4) no credit for conductivity limitation of primary metal heat transfer. It should be noted that these conservatisms were employed to simplify the analysis and would not necessarily be employed in more detailed design calculation.

The break is assumed to be a double ended rupture in a hot leg at the surge line connection. The unaffected primary loop, pressurizer and surge lines are assumed to release mass and energy until an isolation valve closes. The affected primary loop blows down completely. Primary metal heat and secondary side energy are released as ECCS coolant refloods the affected loop. Table 5.5.10-1 lists pertinent data for the analysis. Primary and secondary volumes have been increased to account for pressure and temperature expansion from room temperature to the normal operating conditions. The thermal properties of deuterium oxide have been considered.

The unaffected primary loop mass/energy release rates are given in Table 5.5.10-2. The unaffected loop and pressurizer are isolated by a valve which closes on a containment high pressure signal (6 psig, 5 psig setpoint plus 1 psi tolerance). The valve closure time is assumed to be 6 seconds including a 1 second actuation delay. For the unaffected loop blowdown, the surge line flow resistance has been considered.

The affected primary loop mass/energy release rates are given in Table 5.5.10-2. No credit for flow resistance is taken. For the blowdown, no metal heat transfer is considered.

During the reflood period, affected primary loop metal and affected side steam generator secondary coolant are assumed to be cooled to 300°F by the time of containment peak pressure. Typical of United States pressurized water reactors, 10 percent of affected

side steam generator secondary side metal energy is assumed to be released by the time of containment peak pressure. The reflood mass release rate is assumed to be equal to the maximum ECCS flow. The release is conservatively assumed to be pure steam. The reflood period mass/energy release is assumed to end when the integral energy release rate equals the decay heat integral (120 percent of the infinite decay heat rate) plus metal heat plus the ECCS energy integral.

Taking credit for the affected primary loop flow resistance and steam binding would reduce the reflood release rate giving the containment walls more time to soak up energy and reduce containment peak pressure. ECCS water would condense some of the primary loop steam thus reducing the release rates. If metal heat conductivity had been considered in the analysis, the thick affected primary loop walls would be at a temperature above 300°F at the time of containment peak pressure thus reducing the release rates and containment pressure.

A similar evaluation was performed for the main steamline break accident (ASLB) with the corresponding mass/energy release information given on Table 5.5.10-3.

To evaluate the Mass/Energy Release Data the UE&C developed Computer code (CONTRAST-S) was used to calculate the transient response of the containment to a loss of coolant accident. The mass and energy release rates are the input to this code. The code has provisions for direct energy addition to the containment atmosphere as well as to the sump water. The containment system model used in the code includes active heat removal systems such as containment sprays, fan coolers, containment spray and residual heat removal heat exchangers and component cooling water heat exchangers. The containment building is considered a multi-region heat conducting structure. The heat transfer coefficient

can either be calculated internally in the code using various correlations or be supplied as input to the code. The one dimensional heat conduction equation is used to predict the thermal transients in the structures. The containment volume is assumed to have separate liquid and gas regions with no heat transfer between them. The mass and energy balance equations are solved assuming steady state conditions and thermodynamic equilibrium in each region during a small time step to calculate the pressure-temperature transients in the containment. A more complete description of this analysis technique is given in Reference 5.5-17.

The results of the containment analysis are shown as pressure and temperature histories during the LOCA on Figures 5.5.10-1 and 5.5.10-2. As shown on Figure 5.5.10-1 the containment pressure reaches a maximum of 23 psig during the blowdown phase of the accident and then develops a secondary peak of 65 psig during the reflood portion of the transient. This secondary peak is significantly higher than would be expected from a two loop system where only slightly more than half the stored energy is available for release during the transient, and is not thought to be physically realistic but rather due to the extreme conservatism of the analysis. In particular, this high containment pressure is due to the following:

1. A very conservative model was used for developing the mass/energy release data as described in the preceding discussion. Given sufficient time a model could be developed which would more realistically account for such things as steam binding and condensation of steam. This would reduce the peak pressure in the containment.
2. The shutdown cooling system capacity was sized for the same performance as exists in a Candu, a cool down capability of 300°F per hour. This rate of cool down is considerably

higher than is provided in a PWR due to the Nil Ductility Transient Temperature limitations on the pressure vessel and was provided on this conceptual design to maintain the ability for rapid shutdowns. The shutdown cooling system also serves as the low pressure safety injection system for LOCA. The high capacity provided results in a high rate of energy release to the containment during the transient, producing high containment pressures. It is expected that this high capacity is more than adequate to preserve the integrity of the fuel cladding since CANDU is not limiting during the reflood phase and thus cooldown rate could be sacrificed for a reduction in the containment design pressure.

3. An assumed heat sink capacity was used for the containment evaluation. Considerable effort would be required to determine the exact value that would properly represent the conceptual HWR described in this report. A thorough evaluation of the available heat sink could significantly affect the containment pressure.

Based on the above it is anticipated that a containment design of 30 to 40 psig would result from a detailed evaluation based on specific modeling for a HWR design. Pressures of this magnitude are within the design strength of the containment incorporated in the conceptual HWR design.

TABLE 5.5.10-1

LOCA MASS/ENERGY RELEASE DATA

	Coolant Volume ft ³	Coolant Mass lbm.	Coolant Energy ⁽¹⁾ 10 ⁶ Btu	Metal Energy ⁽¹⁾ 10 ⁶ Btu
Affected Loop Primary	7,587	323,679	199	144
Affected Loop Core	----	-----	---	15
Unaffected Loop Primary	7,587	323,679	199	144
Unaffected Loop Core	----	-----	---	15
Pressurizer & Surge Line Liquid	1,366	50,628	35	25
Steam	973	6,193	7	
Steam Generator (1 of 4) Liquid	2,370	107,589	60	38
Steam	4,026	102,59	11	
Safety Injection Tanks (SIT)				
Number	Two tanks per loop			
Water Volume	1927 ft ³ per tank			
Safety Injection Pumps (ECCS)				
High Pressure (HPSI)				
Number	Two			
Flow	9000 gpm per pump			
Low Pressure (LPSI)				
Number	Two			
Flow	815 gpm per pump			

(1) Relative to 32°F.

TABLE 5.5.10-2
LOCA MASS/ENERGY RELEASE DATA⁽⁴⁾

Time, Seconds	Unaffected Loop ⁽¹⁾		Affected Loop	
	Mass Rate Lb/Sec.	Energy Rate $\times 10^6$ Btu/Sec.	Mass Rate Lb/Sec.	Energy Rate $\times 10^6$ Btu/Sec.
0.0	3573	2.506	97800	60.438
0.5	3566	2.501	93224	57.574
1.0	3559	2.496	87783	54.164
1.5	3553	2.491	81364	50.136
2.0	3546	2.487	73578	45.247
2.5	3539	2.482	64328	39.435
3.0	3532	2.477	54043	32.974
3.5	3526	2.472	42983	26.036
4.0	3519	2.467	31937	19.137
4.5	3512	2.462	22230	13.117
5.0	3504	2.457	14297	8.257
5.5	3498	2.452	7684	4.331
6.0	3491	2.447	4179	2.280
6.5	3484	2.442	2281	1.206
7.0	3477	2.437	1113	0.569
7.5	0	0.0	361	0.178
8.0	0	0.0	224	0.107
8.5	0	0.0	199	0.092
9.0	0	0.0	175	0.076
9.5	0	0.0	154	0.062
10.0(2)	0	0.0	137	0.048
10.1	0	0.0	2660	3.458
95.0	0	0.0	2660	3.458
95.0(3)	0	0.0	0	0.0
Integrals	22,900 Lbm	16×10^6 Btu	546,000 Lbm	489×10^6 Btu

- (1) Assumes unaffected loop and pressurizer are isolated from containment. Isolation is based on 1 second delay time for containment pressure to reach 6 psig setpoint, 1 second instrumentation delay, and 5 second valve closure.

If no isolation valves are used, approximately 850,000 pounds and 550 million Btus will be released to the containment over 400 seconds from the "unaffected" loop in addition to the 546,000 pounds and 489 million Btu from the affected loop.

- (2) End of blowdown, start of reflood
- (3) End of reflood
- (4) Total release to containment is sum of releases from unaffected loop and affected loop

TABLE 5.5.10-3

MSLB MASS/ENERGY RELEASE DATA⁽¹⁾⁽²⁾⁽³⁾

Time, Seconds	Unaffected Loop		Affected Loop	
	Mass Rate Lb/Sec.	Energy Rate $\times 10^6$ Btu/Sec.	Mass Rate Lb/Sec.	Energy Rate $\times 10^6$ Btu/Sec.
0	3194	3.796	3182	3.782
2	3211	3.817	2890	3.444
4	3206	3.810	2695	3.219
6	3186	3.787	2546	3.046
8	3157	3.753	2432	2.912
10	3119	3.710	2334	2.797
12	3122	3.714	2246	2.694
14	3117	3.707	2166	2.599
16(3)	3102	3.690	2091	2.511
18	2572	3.075	2036	2.446
20	2132	2.559	1976	2.375
25	1341	1.614	1823	2.193
30	827	0.994	1679	2.021
35	524	0.626	1546	1.862
40	304	0.361	1425	1.717
45	151	0.178	1315	1.585
50	65	0.077	1214	1.463
60	0	0.0	1047	1.260
70	0	0.0	907	1.091
80	0	0.0	798	0.845
100	0	0.0	625	0.749
120	0	0.0	447	0.532
140	0	0.0	325	0.385
160	0	0.0	231	0.273
180	0	0.0	166	0.195
200	0	0.0	149	0.178
200.1	0	0.0	0	0.0
Integrals	82,000 Lbm	98×10^6 Btu	169,000 Lbm	203×10^6 Btu

- (1) Total release to containment is sum of releases from unaffected and affected loop.
- (2) Case assumes NSIV failure and no loss of non-emergency power. No loss of containment cooling train or diesel delay need be considered with this data but see Note 3.
- (3) To determine worst single active failure, loss of one containment cooling train must also be considered. For loss of containment cooling train case, no NSIV failure need be postulated. For loss of containment cooling train case, mass/energy release data from unaffected loop beyond 16 seconds can be taken as zero. No diesel delay need be considered.

FIGURE 5.5.10-1
HEAVY WATER REACTOR PLANT
CONTAINMENT PRESSURE HISTORY

FIGURE 5.5.10-2

HEAVY WATER REACTOR PLANT

CONTAINMENT TEMPERATURE HISTORY

5.5.11 Environmental Impact of HWR

From an environmental standpoint, HWRs of the Candu-PHWR type are very similar to light-pressurized-water reactors. Both systems employ ceramic fuel encapsulated in zirconium fuel rods contained within a primary circuit to provide barriers against the release of radioactive fission products. In both systems, steam generators are used to develop the steam in the secondary circuit to drive the turbine generator; in both cases, the secondary circuit contains light water.

There are, however, a number of attributes of the HWR system which are sufficiently different from those of the LWR so as to quantitatively change the environmental impact. These aspects are summarized below;

- ° The conceptual reference HWR design operates at a somewhat lower efficiency than that of the PWR, 31.3% vs. 33-34% for the PWR. This lower efficiency results because of heat loss to the moderator; the heat deposited in the moderator must be rejected to the environment since the moderator temperature is too low for electrical power generation. As a consequence, 5-8% more heat will be rejected to the atmosphere.
- ° Because of the pressure tube design and on-line refueling feature of the HWR, failed fuel can be easily detected and removed from the reactor. Thus the potential for the release of radionuclides to the environment is diminished.
- ° Because the HWR is cooled and moderated with heavy water, production of radioactive tritium is much higher (factor of 100 or more) than in the LWR. Careful attention is paid in the design of HWRs to minimizing D_2O /tritium losses, both because of the radiological impact and because of the cost of D_2O . Nevertheless, some tritium will reach the environment during normal operation and considerable inventories of

tritium are potentially available for release during accidents. Tritium production rates and concentration are given in Table 5.5.11-1.

- ° For the purpose of an environmental impact analysis, a tritium effluent release of 18,100 curies per year* is usually assumed for a 1000 MW(e) LWR (Ref. 5.5-10). Tritium releases for the Pickering Station (total of 2,160 MW(e)) have generally ranged between 15,000 and 60,000 curies per year, averaging about 17,300 curies per year per 1000 MW(e) of installed capacity (Ref. 5.5.11). The tritium releases from the HWR may thus be considered, on the average, to fall within the environmental impact postulated for the LWR. It should be noted, however, that the actual tritium releases from LWRs are only a small fraction of that postulated for the environmental impact analysis, and that the tritium releases from the HWR will be an order of magnitude or more larger than that observed in LWRs. However, operating experience indicates that releases from the HWR (Ref. 5.5.11) will result in only a small fraction of the maximum permissible concentrations of tritium.

Other aspects important to the environmental assessment, such as gaseous liquid and solid effluents, land utilization, critical materials, and socio-economic factors will be similar to those of the LWR.

The release mechanisms from a HWR nuclear power plant are; (1) through the containment and up the stack, (2) through the penetrations building, (3) through the turbine-generator and auxiliaries, and (4) onsite storage of tritiated components and wastes. Before an individual may enter the containment,

*Environmental impact analyses for the LWR fuel cycle have been predicated upon reprocessing of spent fuel. The great majority of the tritium effluent release is from reprocessing. Tritium releases from LWRs are typically 1-2% of this value.

or other enclosed areas of the plant, the tritium concentration would be limited to $5\mu\text{Ci}/\text{m}^3$, which is the maximum permissible concentration (MPC) in a restricted area base on a 40-hour week.

To keep the tritium concentration in the containment to a minimum during normal operations and to reduce the tritium to acceptable levels for reentry after a small leak, a recovery system is furnished, namely, the D_2O vapor recovery system. The D_2O vapor recovery system is zoned according to the release rates in the areas. All zones are maintained under a negative pressure in the structures and a dew point of 0°F to minus 30°F .

In addition, proper design of reactor coolant system and interfacing systems which operate at high pressure and contain D_2O , will ensure that leakage from these systems is minimal. All pipe joints will be welded. Components such as filters and tanks will be designed with "leak tight" covers, hand holes, etc. This can be accomplished with seal welding if required. Pumps, such as charging pumps, will have shaft seals and leak off's which will be piped to a D_2O storage tank.

The RCS does not contain valves, however, connecting systems containing high pressure D_2O do. Leakage past the stem on these valves will be controlled by selecting valve designs that either limit leakage or have double packing and leak off connections to control any leakage past the first packing.

In the moderator and connecting systems, which also contain D_2O , leakage control is less of a problem because of the

lower operating pressure. However, leakage control methods as described above would be employed.

The main unresolved licensing issue, in addition to issues such as waste disposal which are common also to the LWR, is the establishment of appropriate operational limits for tritium release. Assuming that HWR operation will not be constrained by criteria for tritium release no environmental R&D should be required nor should environmental licensing considerations constrain commercial introduction dates. However, if it proves necessary to reduce tritium emissions due to environmental considerations, R&D may be required to demonstrate practical methods of separating tritium from heavy water. Systems capable of separating tritium from deuterium have been in operation for sometime, based on the cryogenic distillation of deuterium gas. AECL has a conceptual design under development which would provide a process simplification, and a demonstration plant has been proposed to service the NRX, NRU, NPD, and Douglas Point reactors.

TABLE 5.5.11-1
Tritium Concentration⁽¹⁾

Coolant:

$$\frac{\text{Atoms Tritium}}{\text{barn-cm}}^{(2)} = 2.21 \times 10^{-10} + 5.80 \times 10^{-8} (1 - e^{-.05622t})$$

Moderator:

$$\frac{\text{Atoms Tritium}}{\text{barn-cm}} = 2.06 \times 10^{-6} (1 - e^{-.05622t})$$

-
- (1) As a function of time (t) of full power years of operation. D₂O leakage has been neglected.
- (2) Assumes 0.5% fuel failure.

5.5.12. Proliferation Risks and Safeguards

Evaluations of proliferation resistance have confirmed the general expectation that no fuel cycle or reactor type is completely free of proliferation risk, and the HWR is no exception to this rule. Indeed, depending upon the weight placed upon various factors in the proliferation analysis, the HWR might be viewed as less resistant to proliferation than are the LWRs currently in widespread use: the use of on-line refueling in the HWR complicates material accountability, larger quantities of plutonium are generated per unit energy produced (although the enrichment of plutonium in spent fuel is lower than that of the LWR), and the ability of the HWR to operate on natural uranium potentially reduces the effectiveness of fuel supply sanctions. However, the differences in proliferation risk between reactor types is generally viewed as small compared to the differences inherent in fuel cycle options.

In particular, the use of the once-through uranium fuel cycle is generally viewed as less subject to proliferation than are fuel cycles involving reprocessing and the purification and use of plutonium. Although methods for reducing the potential proliferation risk of plutonium utilization are being explored within DOE's Nonproliferation Alternative Systems Assessment Program (NASAP) and the International Fuel Cycle Evaluation (INFCE) Program, a continued reliance on the once-through fueling mode may therefore be desirable, at least for the foreseeable future while institutional and technical safeguards for plutonium use are developed. At the same time, it is recognized that in the absence of breeder reactor deployment, the benefits of nuclear power are limited by the size of the uranium resource base and the efficiency with which that base can be utilized. Consequently, reactor types which more efficiently utilize uranium and which allow more expensive uranium ores to be economically employed might contribute to the goal of nonproliferation by extending the viability of the once-through fueling option and forestalling the eventual need of plutonium-fueled breeder reactors.

In order to improve the nonproliferation risks the conceptual HWR has been arranged to provide controlled access to and within sensitive plant structures. Figure 5.5.12-1 depicts the arrangement of the structures and supporting facilities of the plant. The entire area is surrounded with fencing. This controlled area is divided into four zones; 1) administration and parking; 2) switchyard; 3) cooling tower; 4) nuclear plant, turbine building, and safety-related supporting facilities. Any person entering or leaving must pass through a check-point at the entrance to the property.

Employees and visitors to the nuclear facility must enter the area through the security building (see Figure 5.5.12-2). The details of the structure are described in the Structure Design Description for the Plant Security Building. The building is arranged to provide the maximum control of people passing through the detection area and a maximum of security to the guard personnel and monitoring equipment.

Alarms and monitors are located in a protected area in the building. A separate emergency diesel-generator power supply is located at the security building to provide electricity to the detection and monitoring equipment and flood lighting through the area.

An enclosed monitored walkway is provided between the security building and entrance to the turbine building or the control, auxiliary, and diesel (CAD) building. Entrance to the CAD building is through either a key-locked door or to a key-locked door of an elevator. Access to other floors of the CAD building or to the other buildings in the safety-related structure complex is through a check-point on the control room level. All areas of the safety related building are monitored in the security building.

Emergency exit-only doors would be provided on the ground floor of all buildings and structures. These doors would be provided with crook bar locks for exiting from the building, and with key locks that could only be opened by the security force. All doors

would be monitored and provided with contacts that would actuate an alarm in the security building.

Intrusion detection devices would be located around areas and other locations outside and inside safety-related buildings.

The Canadians in cooperation with the International Atomic Energy Agency have been evaluating various techniques of improved safeguards as reported in Reference 5.5-12. A system which is being developed complements the normal nuclear materials accounting activities by providing a redundancy with some novel highly instrumented equipment which performs surveillance, monitoring and counting. Although no provisions have been made to incorporate a system of this type into the conceptual HWR, a system once developed could be included to add some measure of safeguards improvement.

PLANT NOMENCLATURE

- 1-REACTOR CONTAINMENT BUILDING
- 2-REACTOR SERVICE & FUEL HANDLING BUILDING
- 3-TURBINE BUILDING
- 4-HEATER BAY
- 5-CONTROL & DIESEL BUILDING
- 6-ADMINISTRATION BUILDING
- 7-CONDENSATE STORAGE TANK
- 8-RAILROAD
- 9-DEMINERALIZED WATER STORAGE TANK
- 10-PENETRATION BUILDING
- 11-ANNULUS BUILDING
- 12-DIESEL STORAGE TANKS
- 13-SECURITY BUILDING
- 14-TRANSFORMER AREA
- 15-CIRCULATING WATER & SERVICE PUMP BUILDING
- 16-AIR INTAKE STRUCTURE
- 17-MAKE-UP WATER INTAKE STRUCTURE
- 18-MAKE-UP WATER PRETREATMENT BUILDING
- 19-HOLDING POND PLANT EFFLUENT
- 20-ULTIMATE HEAT SINK COOLING TOWER STRUCTURE
- 21-COOLING TOWERS
- 22-COOLING TOWERS SWITCHGEAR BUILDING
- 23-SWITCHYARD
- 24-FIRE WATER PUMP HOUSE
- 25-FIRE WATER STORAGE TANKS
- 26-FUEL OIL STORAGE TANK
- 27-PARKING AREA
- 28-D₂O UPGRADING TOWER
- 29-GUARDHOUSE
- 30-SECURITY FENCE
- 31-ANHYDROUS AMMONIA STORAGE TANK
- 32-AMMONIA-WATER STORAGE TANK

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FIGURE 5.5.12-1
PLOT PLAN

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FIGURE 5.5.12-2
SECURITY BUILDING
GENERAL ARRANGEMENT

5.5.13 Summary and Conclusions

° Design Approach

The safety and licensability of a reactor concept is due partially to its engineered features and partially to its inherent characteristics (those features which cannot be readily altered by engineering change). Although the engineered features of the standard Candu reactor are viewed to be satisfactory and to adequately protect the health and safety of the public, the basic approach employed in developing the conceptual HWR design was to replace these engineered features with PWR-type design so as to minimize questions of licensability under U.S. licensing criteria. Thus, such aspects of the engineered features as containment design and liner, emergency core cooling system, and seismic restraints have been replaced with U.S. type systems. Safety and licensing questions thus center upon the inherent characteristics of the HWR, and upon the design changes which have been incorporated into a conceptual HWR to enhance uranium utilization and to improve power cost economics.

° Inherent Characteristics

When considering the inherent features of a reactor concept, it must be recognized that certain features will be less satisfactory and certain features more satisfactory than those of other alternatives. Of prime concern is the overall implication to safety when all the unique attributes of a system, both positive and negative are considered. The prime inherent attributes of the pressurized heavy water reactor which contributes to increased plant safety are considered to be the use of the pressure tube configuration with the resulting mitigating effect on the consequence of

the loss of coolant accident, the ability to detect and replace failed fuel because of the on-line refueling feature, and the redundant shutdown systems provided in the HWR. The use of the pressure tube configuration in the HWR, coupled with the injection of emergency core cooling water into both the inlet and outlet headers, facilitates the supply of coolant water to the fuel channels in the event of a loss-of-coolant accident. The presence of large quantities of relatively cool moderator surrounding the fuel channels provides a backup heat sink in the event of failure of the emergency core cooling system. The location of fuel within separate fuel channels coupled with the ability to do on-line refueling also allows failed fuel to be more readily detected and replaced, and as a result the radionuclide inventory of the primary circuit can be minimized thus reducing the quantities of radioactivity which can be released to the environment during postulated accidents.

The inherent attributes of the HWR which might be viewed as being less satisfactory than those of the LWR are the use of thin-walled pressure tubes as the primary pressure boundary and the positive void coefficient of reactivity. Pre-service and in-service inspection should, however, be sufficient to guarantee that pressure tubes will leak prior to failure. Leaking pressure tubes can be readily detected through the surveillance system which monitors the gas contained in the annulus between the pressure and calandria tubes, providing ample time to take corrective action. Experiments have also demonstrated that failure propagation will not occur even in the highly unlikely event of a pressure tube failure. The positive void coefficient of reactivity will result in increases in reactor power during under-pressurization and under-flow events. This increase in reactor power is terminated by reactor trip and two independent shutdown systems are

provided to ensure that these transients are terminated in the event of failure of one of them.

° Design Modifications

In arriving at the conceptual HWR design, a number of changes to the standard Candu plant have been made. These changes include an increase in net electrical output to 1260 MW(e), the use of slightly enriched (1.2 w/o) uranium fuel, an increase in primary coolant pressure to 2250 psi, a reduction in D₂O reflector thickness and an increase in moderator temperature. A two-loop primary system configuration is also employed. Although detailed design evaluations will be necessary before each of these design modifications can be incorporated into a commercial HWR, significant safety and licensing questions are engendered primarily by the increase in system pressure.

Of concern is the effect of the change of primary coolant pressure and temperatures on pressure tube integrity, and upon the integrity of the rolled joint which is used to mechanically fasten the pressure tube to its end fittings. In order to accommodate the increase in primary system pressure and temperature, pressure tube wall thickness has been increased so as to maintain an equivalent degree of resistance against creep as is present in the standard Candu design. Allowances for increased primary system pressures and temperatures have consequently been made, to the extent possible with available information, in evolving the conceptual HWR design. However, it is recognized that the data base on pressure tube material performance at the postulated pressures and temperatures of the conceptual design is inadequate, and that R&D to determine pressure tube material strength and corrosion resistance will be necessary.

The conceptual HWR also employs slightly enriched fuel and achieves discharge exposures of about 20,000 MWd/t, significantly higher than that achieved with natural uranium fueling.

Although the performance of fuel at these burnups is viewed to be primarily an economic issue (since failed fuel can be readily detected and replaced because of the on-line refueling feature of the HWR) irradiation demonstration programs to demonstrate the performance of Candu-type fuel to higher burnups are also desirable. Such irradiation programs could probably be performed in existing Canadian test reactors.

◦ Licensing Evaluation

The performance of a detailed and quantitative safety and licensing evaluation, in which the consequences of various transients and postulated accidents are calculated and compared to U.S. acceptance criteria, was beyond the scope of this study. However, some insight into the likely outcome of such an analysis can be obtained by reviewing Canadian licensing and acceptance criteria.

Canadian licensing practice requires that all accidents be analyzed under both the assumption that safety systems function properly* (single failure) and under the assumption that any one of the safety systems is unavailable (dual failure). Under U.S. licensing requirements, only the single failure must be addressed. Acceptance criteria are based solely upon site boundary doses. Different site boundary dose limits are imposed for the single and dual failures, with the latter roughly consistent with U.S. acceptance criteria for limiting faults. Single failures (which include limiting faults such as LOCA) are limited to much lower site boundary doses than would be allowed for

*Credit is taken for the proper action of only one of the two shutdown systems, and in the case of dual failures only the least effective.

such limiting faults under U.S. acceptance criteria. In this respect, Canadian licensing criteria are more restrictive than are U.S. criteria.

U.S. licensing criteria, however, impose additional limitations on fuel damage for moderate frequency and infrequent faults (no fuel damage for moderate frequency faults, limited fuel damage for infrequent faults). Canadian licensing does not impose such limitations on fuel failure, with acceptance criteria being based solely on site boundary dose and fuel failure considered to be of economic significance only. The imposition of fuel failure criteria on the HWR design may necessitate a reduction in core power density. Peak clad temperatures during the limiting loss-of-coolant event also exceeds that specified by U.S. regulation (Appendix K); some decrease in peak linear heat rate may be necessary to conform to U.S. LOCA criteria. Since peak clad temperatures occur early in the transient, before emergency core cooling water can be injected, the detailed performance of the ECC system is not as critical, i.e., there is considerable margin (to peak clad temperature limits during the reflood phase of the LOCA) inherent in the design of the ECC system to accommodate, if necessary, U.S. requirements.

In summary, Candu reactors are designed to licensing acceptance criteria which are, in places, more stringent than U.S. requirements, and in places somewhat less stringent. In order to definitively assess whether the Candu system conforms to U.S. licensing criteria, a detailed quantitative safety analysis addressing the consequences of anticipated transients and other posutlated accidents would be required.

- Operational Performance Characteristics

The operational characteristics of the HWR are comparable to the PWR in most instances. It appears that the HWR is capable of performing the typical load following maneuvers required of the PWR, although more detailed evaluations will be necessary to demonstrate their supposition. The HWR is also capable of more rapid rise to power than is the PWR. The primary operational constraint placed on the HWR is the necessity of returning to power within about 30 minutes from reactor trip. If the reactor is not returned to power within this time period, it becomes necessary to wait 36 hours (for xenon decay) before a return to criticality is possible.

- Seismic Design

Because of differing geology, seismic assumptions used in the U.S. differ somewhat from Canadian assumptions. A preliminary qualitative review of seismic considerations suggests that the conceptual HWR will meet U.S. seismic requirements. However, quantitative analyses should be performed to demonstrate the adequacy of the HWR design for U.S. seismic conditions.

- Alternate Fuel Cycles

The evaluation of safety-related core physics parameters (coefficients of reactivity) indicate that important core parameters for the slightly enriched (1.2 w/o) uranium and denatured thorium systems are comparable to those obtained for natural uranium fueling. Increases in primary system temperature and pressure were also found to have no detrimental effect on core parameters. Consequently, it is concluded

that the use of slightly enriched fueling or of the denatured thorium fuel cycle, and of the changed primary system operating temperatures and pressure, will not significantly alter the predicted consequences of safety analyses performed for natural uranium fueling.

° Environmental Impact

Because the HWR is cooled and moderated with heavy water, production of radioactive tritium is much higher than in the LWR. Careful attention is paid in the design of HWRs to minimize D₂O/tritium losses, both because of the radiological impact and because of the cost of D₂O. The measured releases of tritium from operating Candu plants has been low, only a fraction of the allowable release rate, and the average releases appear to be within postulated values assumed for U.S. environmental impact evaluations. However, tritium releases will be a factor of ten or more greater than that observed in operating LWRs.

As a result of the lower efficiency of the conceptual HWR (31.3% vs. 33-34% for the PWR) 5 to 8% more heat will be rejected to the atmosphere.

Because of the pressure tube design and on-line refueling feature of the HWR, failed fuel can be easily detected and removed from the reactor. Thus the potential for the release of radionuclides to the environment via this pathway is diminished.

° Potential Safety, Licensing, and Environmental Issues

The Canadian experience with the construction and operation of Candu plants and the brief licensing review performed as

part of this evaluation provides a favorable outlook on plant safety. However, there are a number of aspects which will require more detailed evaluation and/or the development of a suitable data base prior to the design and licensing of plants based upon the conceptual HWR design. These areas are summarized briefly below.

1. Development programs are necessary to develop a data base for zirconium-niobium alloy pressure tubes operating under the pressures and temperatures postulated for the conceptual HWR design. Of concern is the resistance to creep at these elevated pressures and temperatures, as well as corrosion characteristics. Pressure tubes, which are subject to the full neutron flux and primary system pressure, are not presently covered by the ASME boiler and pressure vessel code; the development of an appropriate ASME code should be part of the development program.
2. A R&D program should be initiated to demonstrate that rolled joints can be successfully fabricated using the thicker pressure tube material of the conceptual HWR, and to demonstrate the integrity of these rolled joints at higher operating pressures. An ASME code for rolled joints should be developed as part of this program.
3. The probability and consequences of pressure tube failure and of the potential for damage propagation to adjacent tubes requires further evaluation, particularly as it relates to the higher system pressure of the conceptual design. Tests of calandria tube integrity during simulated pressure tube bursts might be considered also.

4. Radiation demonstration programs should be initiated to verify the anticipated acceptable performance of Candu-type fuel to the higher discharge exposures contemplated in the conceptual HWR design.
5. Detailed three-dimensional fuel management analyses should be performed for slightly-enriched uranium fueling to establish whether the use of slightly enriched fueling results in excessive core peaking. This analysis should also address the effect of possible small positive power coefficients in the operating range and assess the ability of the control system to accommodate these coefficients.
6. A detailed quantitative evaluation of the consequences of anticipated transients and other postulated accidents should be performed. This evaluation should identify design basis events, establish acceptance criteria for these events, and determine whether the consequences of these accidents fall within the acceptance criteria. The analysis should include an evaluation of the loss-of-coolant accident, as well as of other transients which can limit the power capability, and of anticipated operational occurrences which have potential for fuel damage.
7. A seismic evaluation should be performed to demonstrate the suitability of the HWR design under U.S. geology and seismic requirements.
8. A detailed assessment should be made of the effect of tritium levels on personnel exposure during inspection and maintenance operation, and of the effect of potential tritium releases to the environment.

9. A more extensive evaluation should be made of the implications of the positive void coefficient of reactivity on transient undercooling and underpressurization accidents. Although from a technical standpoint this positive coefficient of reactivity can be accommodated within design, (i.e., will result in accident consequences which fall within acceptance criteria) this supposition should be tested at an early date through discussions with licensing authorities.
10. In order to obtain a more complete licensing review of the conceptual HWR design, a preliminary safety information document, closely paralleling the content of the PSAR, should be developed and pursued through discussions with licensing authorities.

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6. Commercial Introduction of the HWR in the U.S.

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6.0 Commercial Introduction of the HWR in the U.S.

6.1 Introduction

The purpose of this chapter is to delineate those factors and considerations which might impact the potential for commercial introduction of the HWR into the U.S. In general, the discussion is limited to identifying the factors which can influence commercialization, and no attempt has been made to predict whether or not the HWR would successfully compete with light water reactors or other alternate concepts, or the degree to which the HWR might penetrate the U.S. market. The latter type of analyses are beyond the scope of work, since these analyses involve evaluation of the competition of the HWR with alternate concepts and hence require the development of detailed commercialization information for these other alternatives.

Although it is perhaps too obvious to need be said, commercialization requires two essential ingredients: a manufacturer capable and willing to supply the reactor concept and a utility willing to purchase and deploy the concept. These two aspects are, of course, interrelated since a utility is unlikely to consider purchasing the alternate reactor concept, unless the concept is reasonably well developed and a manufacturing base exists while a manufacturer is unlikely to invest in concept development or manufacturing base unless there is a reasonable likelihood of a commercial market for the product. The development of a commercial HWR in the U.S. requires the design and performance characteristics of an HWR plant suitable for the U.S. market be determined, that the concept be licensable, and that a manufacturing capability be developed. A utility's willingness to adopt the HWR concept will depend upon the relative power cost economics and upon the perceived risk (technical, licensing, and plant performance) that the expected economic performance will not be attainable.

Thus, it should be clear that in order to commercially deploy a concept, there must be marketplace incentives both for a manufacturer to develop a concept and for a utility to purchase and deploy the alternative. Possible scenarios for the introduction of heavy water reactors into the U.S. are discussed in Section 6.2. In developing this section, it has been assumed that the HWR concept is viable both from a technical and licensing standpoint, and that there is an incentive in both the manufacturing and utility sectors to develop and deploy the HWR concept. The remaining sections of this chapter explore the question of whether this assumption used in developing possible scenarios is true, with emphasis on areas where further development is required in order to resolve issues which might currently preclude commercialization. In particular, Section 6.3 and 6.4 address potential technical problem areas and licensability, respectively, while Section 6.5 identifies the Research Development and Demonstration (RD&D) requirements which are a prerequisite for commercialization. Lastly, Section 6.6 discusses commercial viability, i.e., the question of whether there really is an incentive to develop and deploy a U.S. HWR design from a domestic commercial standpoint.

Traditionally, the stages of reactor deployment can be divided into three steps: proof of principle, demonstration plant, and lead commercial unit. The three stages are typically distinguished by the size of the units constructed, and each stage typically involves some degree of component design and licensing development. The main purpose of the proof of principle experiment is to demonstrate the basic principles of the concept, i.e., in the case of the Candu-PHWR, to demonstrate the basic concept of a heavy water moderated and cooled pressure tube reactor operating with natural uranium fueling. An example of such a proof-of-principle facility is the Canadian NPD plant which has a net power output of 22 MW(e) and began operation in 1962. Such a proof-of-principle experiment, of course, is normally preceded by extensive R&D programs, which often include the deployment of test reactors in order to provide experimental information in support of the proof-of-principle and subsequent larger-scale facilities (the NRX reactor is an example of such a test reactor).

The second stage, that of a demonstration plant, is intended to demonstrate system and component performance. Normally, such a demonstration plant has a net electrical power output less than half that of a fully commercial unit, but is sufficiently large to provide reasonable confidence that the concept can be scaled to full-size commercial units. Although it is usually necessary to use some subscale components in such an intermediate size demonstration plant, full-scale components are used to the greatest extent practical in order to facilitate the transition to full-size commercial units. The 208 MW(e) Douglas Point Station, which began operation in 1967, might be considered an example of such an intermediate size demonstration plant.

The last stage in the demonstration plant sequence is typically a lead plant. This plant typically has a size 50 to 100% of that of the commercial unit, and may actually be the first plant in a sequence leading up to the reference offering. This lead plant provides a test of full-scale components, and proof of operating performance, licensability, and plant economics. Pickering Unit 1, which began operation in 1971 at a net electrical power of 514 MW(e) is considered to be the Canadian lead plant for the Candu-PHWR concept.

The Candu-PHWR, operating with natural uranium fueling, has, of course, been fully commercialized by Canada. Reactors of this type having electrical generating capacities of between 500 and 750 MW(e) are in operation in Canada and in operation or under construction in a number of other nations such as India, Pakistan, Argentina and Korea. Extensive operating experience has been obtained for Candu reactors, particularly for the Pickering station which contains four 500 MW(e) plants. The first of the Pickering units went into operation in mid-1971, and a fourth unit in mid-1973. Performance of the Pickering station has generally been excellent; thus, it can be concluded that the technical feasibility of heavy water reactors operating with natural uranium fuel has been conclusively demonstrated.

Because of the extensive Canadian operating experience with pressurized heavy water reactors, it is clear that a small proof-of-principle experimental facility would not be required to initiate the introduction of heavy water reactors into the U.S. Subsequent steps in the plant demonstration sequence, however, will depend strongly upon the degree of U.S.-Canadian cooperation. In developing the basic introduction scenarios discussed below, strong Canadian/U.S. cooperation has been assumed, i.e., it has been assumed that the HWRs introduced into the U.S. are introduced either under Canadian license or as a joint U.S./Canadian construction venture. Under this assumption, the complete experimental,

analytic, and commercial performance and licensing data base developed for the Candu-PHWR would be available, as would be critical manufacturing capability for those plant components which may not be available in the U.S. Under the assumption of such joint U.S./Canadian venture, the primary commercialization concerns associated with deploying Candu-type reactors in the U.S. are questions of U.S. licensability (or more explicitly the effect of the licensing process on plant design, performance, and economics) and of the development and demonstration of the design changes with respect to the standard Candu plant which have been incorporated into the conceptual reference design (the most notable of these design changes are slightly-enriched fueling, the increased primary pressures, and the larger--1260 MW(e)--electrical generating capacity). With such a joint U.S./Canadian venture, the first step in deploying Candu reactors in the U.S. could be the construction of a lead plant, although, as discussed below, this lead plant may not incorporate all of the features of the conceptual HWR reference design described in this report. At the end of this section, an introduction scenario is discussed briefly in which it has been assumed that introduction of the HWR in the U.S. is performed independent of the Canadian program. Only a limited amount of attention has been devoted to this scenario, since it is clear that without close U.S./Canadian cooperation the commercial introduction of the HWR in the U.S. would be significantly delayed, perhaps by 10 years or longer.

6.2.1 Introduction of HWRs as a joint U.S./Canadian Venture

In this section, a number of possible scenarios for the introduction of the HWR into the U.S. are presented, all of which are based upon the assumption of close Canadian/U.S. cooperation. These scenarios all rely on the use of Canadian technology, either transferred to the U.S. under some licensing arrangement or transferred more directly by constructing the U.S. plants as a joint U.S./Canadian venture. It is assumed that the Candu-type

reactor will be constructed by U.S. architectural engineers, and that the U.S. LWR supply infrastructure can be used to supply most HWR components and systems; however, unique HWR components or systems which could not readily be supplied by U.S. vendors would be obtained from Canadian sources. In particular, it has been assumed that D_2O supply does not constrain deployment, under the assumption that the requisite D_2O supplies could be obtained from Canada should D_2O requirements exceed domestic supply capability. Likewise, unique Candu components, such as the refueling machines, might initially be obtained from Canadian sources.

As discussed in Chapter 5, the conceptual HWR design developed in this report differs from operating Candu units in two important aspects: design changes to accommodate U.S. licensing and construction practices, and design changes to improve fuel utilization and economic performance. Although design changes were incorporated into the conceptual design to facilitate licensing in the U.S. and to conform to U.S. construction practices (c.f., Section 5.2); nevertheless, a number of potential licensing issues remain (c.f., Section 5.5), and it is clear that many of these issues will not be resolved in the required detail until the concept undergoes detailed licensing review. Consequently, one of the main objectives of a U.S. lead plant demonstration program would be to undertake such a detailed licensing review. It should be noted that of prime concern is not the question of licensability itself (since there is little doubt that heavy-water reactors can be constructed to satisfy U.S. licensing requirements), but rather the detailed design changes which may be necessitated as a result of a U.S. licensing review, and the concomitant impact of the design changes on HWR operational and economic performance. The second aspect of the demonstration program would be to demonstrate the design modifications which have been incorporated into the conceptual HWR design to enhance uranium utilization or to improve economic performance. Significant among these modifications is

the use of slightly-enriched uranium fuel, higher primary (2250 psia vs. 1600 psia) and secondary system pressures, and the greater (1260 MW(e)) net electrical output of the conceptual design.

Two basic introduction scenarios might be envisioned. The first of these would attempt to separate the question of HWR licensability from the question of demonstrating the performance improvement options. In this scenario, the initial lead plant unit would be similar to that of operating Candu plants, and would incorporate only those design changes dictated by U.S. licensing or construction practices. With this introduction scenario, modifications to improve plant performance would be introduced at a later date, once the basic questions of licensability have been resolved. This introduction scenario is discussed below under the heading "Case A". In the second introduction scenario, one would attempt to demonstrate both U.S. licensing and the various performance improvement options in the initial lead plant. In general, this option would result in the most rapid commercialization of the reference plant, but of course involves greater risks and hence is less likely to be employed. Introduction scenarios which seek to demonstrate both U.S. licensing and the various improvement options in a single lead plant demonstration are discussed under the heading "Case B".

Case A - Gradual Introduction of Design Changes

The Case A introduction scenario is based upon the gradual introduction of performance improvement features in a sequence of plants designed eventually to prove the licensability, operating performance, and economics of the conceptual HWR reference design. The initial, or demonstration plant, of this sequence would be similar to Candu reactors currently in operation, so as to minimize the Research and Development (R&D) requirements for this initial unit

and to minimize risk by using, to the greatest extent possible, demonstrated Canadian component technology. Subsequent plants in the sequence would incorporate various improvement options as the prerequisite R&D is completed, until eventually all of the features of the conceptual HWR design would become demonstrated. This option has the advantage of minimizing risk, since questions of licensability in the U.S. are resolved early without the complication of having to simultaneously license undemonstrated performance improvement modifications. The initial licensing demonstration should also be able to proceed more rapidly, since it would not be necessary to successfully complete the R&D required to implement various improvement options at this time.

The first introductory scenario for this case is shown in Figure 6.2-1 (Case A.1). The initial, or demonstration plant, would be similar in design to the standardized Candu-600, as represented by the Gentilly 2 or Point Lepreau Stations currently under construction. The design of this unit would, however, be modified to conform to U.S. licensing criteria and construction practices, i.e., the design would incorporate those design features discussed in Section 5.0 which are related to licensability, but exclude those design changes related to improved performance.

Although not strictly necessary, it would seem reasonable to employ slightly-enriched uranium fueling in this unit based upon the supposition that the use of slightly-enriched fueling would not complicate licensability and that the performance of slightly-enriched fuel could be demonstrated by prior test irradiations in operating Canadian plants. Predicating this initial unit on slightly-enriched fueling should not unduly increase risk, since the plant can always be operated initially with natural uranium fueling should the test irradiations or licensing concerns indicate that this course be prudent. Although the increase in station capacity from that of current Canadian designs is thought to be

straightforward engineering development problem, the demonstration plant might most prudently be of the 600 MW(e) size, both to avoid introducing additional questions of licensability at this stage, and to minimize capital investment. Increased net electrical capacity was consequently assumed to be deferred to subsequent units. As Figure 6.2-1 illustrates, the demonstration of other improvement options was assumed to begin after the performance of the demonstration plant had been proven (i.e., after this plant had been in operation for a period of about two years). Although these various performance improvement options could be gradually introduced, only the increase in primary system temperature and pressure is viewed as evoking serious technological questions, and hence it should be possible to demonstrate all of the performance improvement options predicated for the conceptual reactor design in a single unit.

The time spans shown in Figure 6.2-1 are based upon the assumption that it requires 10 years to license and construct plants of established and accepted design in the U.S. (i.e., that it would take 10 years to license and construct an LWR, or 10 years to license and construct an HWR once that design had previously been licensed). This 10-year period is divided into a four-year period for licensing (the time between PSAR submittal and the issuance of a construction permit), and a six-year period for construction (the time between construction permit and initial operation). The actual time period for licensing and construction of course, has varied considerably from plant to plant. Most recently, licensing and construction times in the U.S. have been somewhat longer than this assumed 10-year period, while in the past and in other countries this time period has been considerably shorter. In developing the introduction scenario of Case A.1, a licensing and construction time span (PSAR to operating license) of 12 years has been assumed. This schedule would provide the NRC with one additional year (five rather than four) to review the PSAR submittal, and one additional year for construction (7 rather than 6 for an established unit).

This 12-year licensing and construction period is preceded by a two-year R&D period (R&D activities, however, continue past PSAR submittal). Since it has been assumed that the demonstration plant is essentially of conventional Candu design (modified only as required to conform to U.S. licensing and construction practices), this two-year R&D period would be devoted primarily to demonstrating slightly enriched fuel performance in Canadian plants and to PSAR development, i.e., to the development and documentation of a design and safety analysis for U.S. siting. A 14-year period from project initiation to initial operation is consequently postulated. The introduction scenario described in Figure 6.2-1 also assumes that licensing and construction of subsequent units is not initiated until the initial plant has successfully operated two years.

The deployment schedule discussed above can be characterized as having a relatively low risk. In particular, the introduction of an advanced HWR design is assumed to proceed in a step-wise fashion, with each step demonstrated by successful reactor operation before commitments to additional or more advanced units are made. The demonstration unit employed is of conventional (proven) Canadian technology, and more speculative technological advances are not postulated until this initial unit has been licensed and has demonstrated successful operational performance on a U.S. utility network. This type of scenario is probably appropriate to a situation where the commercial incentive for the HWR is uncertain, i.e., where there appears to be an incentive for the deployment of the HWR, but where there also appears to be substantial risks (e.g., a risk that the capital costs will be sufficiently high to negate this incentive or that capital costs will increase as a result of the licensing process). As discussed in Section 6.6 (Commercial Viability), the HWR is viewed as having some commercial incentives in the U.S., but that these incentives are somewhat longterm and obscured by the uncertainties in capital cost and licensing impact. Consequently, the introduction scenario specified as Case A.1 is judged to be the most realistic in today's economy.

Case A.2, illustrated in Figure 6.2-2, presents an alternative and somewhat accelerated introduction scenario. In this introduction scenario, the basic philosophy of separating the demonstration of U.S. licensing from the demonstration of improvement options has been retained. However, in this case commitment to the second unit (large lead plant) is assumed to occur earlier, so that licensing review proceeds concurrently with the initial years of operation of the previous unit. Although commitment is made in this scenario prior to the operation of the demonstration plant, construction of the second unit is not initiated until after the demonstration plant has operated successfully for two years, thus deferring the great majority of capital expenditures until after the performance of the previous unit has been demonstrated.

Inherent in this scenario is a somewhat greater risk, and hence one would not expect this scenario to materialize without increased government support unless the incentives for the HWR were perceived to be relatively large. Although the risks are somewhat higher for this accelerated deployment schedule, they are not considered to be unreasonably high. Recognizing that the demonstration plant design is essentially one of proven Canadian technology, the prime questions to be resolved by this demonstration plant are those associated with licensability and of the impact of design changes necessitated by licensing on plant capital cost; unless the licensing process necessitates fundamental design changes, operating performance comparable to the highly successful Canadian units should be anticipated. At the time commitment is made to the second unit, design changes necessary for U.S. licensing will be clearly identified and their impact on plant capital cost and operational performance assessable. Although the earlier commitment to the second unit does necessitate some financial commitment, the majority of capital outlays occur during the construction period which is not initiated until the demonstration unit has operated for a period of two years to demonstrate successful performance.

Case A.3 (Figure 6.2-3) presents a still more rapid deployment scenario. Here, a commitment is made to the second unit (large plant demonstration/ lead plant) immediately after licensing approval is obtained for the initial demonstration plant, and construction of the second unit is assumed to begin prior to the successful operation of the initial demonstration plant. Commitment is thus made to the second unit immediately after U.S. licensing has been demonstrated (at which time the impact of licensing on plant capital cost and operational performance is presumably known), with substantial capital commitments for construction occurring prior to the operation of the initial demonstration plant. Inherent in this scenario is thus a higher risk. This scenario would be appropriate where U.S. licensing and the impact of licensing on plant design were the primary concerns, but where the successful operation of the plant was viewed to be relatively certain.

Case B

The Case B scenarios are similar to those discussed above, except that it has been assumed that both U.S. licensing and the technical viability of those options incorporated into the reference conceptual design to improve plant performance are demonstrated in the initial lead plant. Clearly, the risks and uncertainties in adopting this approach are greater than for the Case A scenarios discussed above, since the concept now differs significantly (primarily because of the higher primary system temperatures and pressures) from that of demonstrated Canadian units. This deployment scenario, however, will demonstrate more rapidly the licensability and performance of the reference conceptual design, upon which the assessments of commercial viability are based.

The first of the Case B scenarios is illustrated in Figure 6.2-4. This deployment scenario is similar to that of Case A.1 in that an attempt is made to minimize risk by requiring that the lead unit to operate for a period of two years, to demonstrate performance,

prior to the commitment of additional units. A somewhat longer time period for licensing review (7 years compared to 5 years in the Case A scenarios) has been assumed, since the design changes incorporated to improve reactor performance will certainly necessitate a longer NRC review. The R&D period prior to submittal of the PSAR has also been increased in order to allow sufficient time for R&D programs to develop information on pressure-tube performance at the elevated temperatures and pressures of the conceptual design, and for the more extensive licensing development program which would be required to prepare the PSAR.

Case B.3 (shown in Figure 6.2-6) parallels the logic of Case A.3. Here, the deployment schedule is accelerated by the assumption that a commitment is made to deploy additional units at the time that the lead plant construction permit is obtained. Although many of the uncertainties associated with deploying the conceptual design will be resolved by this time (i.e., those uncertainties associated with licensability and the impact of licensability on capital cost), there is, nevertheless, considerably greater risk associated with this approach than was the case in Scenario A.3, because the performance of the conceptual HWR would be undemonstrated. Such an accelerated deployment schedule would undoubtedly require significant federal incentives in the form of subsidies or risk abatements for the initial few "commercial" units.

An intermediate case which has somewhat lower risk is illustrated in Figure 6.2-5. Here, commitments are made to commercial units prior to the successful operation of the lead plant, but this decision is delayed sufficiently to ensure that operating experience from the lead plant is available prior to the initiation of construction on the commercial units; i.e., licensing review would be initiated prior to the operation of the lead plant, but construction of the plant would be deferred until lead plant operating experience was on hand in order to minimize sunk expenditures in the event of unsatisfactory lead plant performance.

6.2.2 Introduction of HWRs as a U.S. Venture

In the preceding section, it was assumed that HWRs were introduced into the U.S. as a joint U.S./Canadian venture. Such a joint venture would facilitate the introduction of the HWR into the U.S. by making available basic design and safety data, critical component manufacturing capacity, and the design experience which has resulted from the construction and operation of Candu plants.

If the U.S. were to elect to develop independently HWRs for domestic deployment, additional RD&D would be necessary to develop the basic design and safety information required for plant design and licensing, delaying the introduction of the initial demonstration unit by perhaps three to five years. It would also probably be necessary (or at least prudent) to construct a small prototype plant to gain design and operating experience. The design and construction of such a prototype would probably require 10 to 15 years. Consequently, it is evident that the independent development of an HWR concept by the U.S. would defer commercial deployment well beyond the turn of the century. Since the introduction of HWRs at such a late date is unlikely to be of interest, no further consideration was given to this scenario.

Figure 6.2-1
CASE A.1 - GRADUAL INTRODUCTION OF DESIGN CHANGES
LOW RISK

(1) 600 MW(e), SIMILAR TO CANDU PLANTS IN OPERATION

(2) 1260 ME(e) CONCEPTUAL HWR

Figure 6.2-2
CASE A.2 - GRADUAL INTRODUCTION OF DESIGN CHANGES
MODERATE RISK

Figure 6.2-3

CASE A.3 - GRADUAL INTRODUCTION OF DESIGN CHANGES HIGH RISK

6.2-15

- 1) 600 MW(e), SIMILAR TO CANDU PLANTS IN OPERATION
- 2) 1260 MW(e) CONCEPTUAL HWR

Figure 6.2-4
CASE B.1 - RAPID INTRODUCTION OF DESIGN CHANGES
LOW RISK

(1) 1260 MW(e) CONCEPTUAL HWR

Figure 6.2-5

CORE B.2 - RAPID INTRODUCTION OF DESIGN CHANGES MODERATE RISK

1) 1260 MW(e) CONCEPTUAL HWR

Figure 6.2-6

CASE B.3 - RAPID INTRODUCTION OF DESIGN CHANGES HIGH RISK

6.2-18

1) 1260 MW(e) CONCEPTUAL HWR

6.3 Technical Problem Areas

As has been frequently mentioned, the conceptual HWR design described in this report is based upon the Canadian Candu-PHWR. Reactors of this type have achieved a commercial status in Canada and have demonstrated quite satisfactory performance. Like the LWR or other reactor types, Candu reactors have been subject to various technical problems which have been resolved in the course of a normal engineering design. Consequently, there appears to be no major technical problem areas associated with the basic Candu concept itself, although like any reactor concept additional operating experience with commercial-size units is necessary to fully demonstrate the long-term performance of certain design aspects. For completeness, several of the more major uncertainties in the long-term performance of Candu reactors is reviewed in Section 6.3.1. Of greater concern are the modifications to the basic Candu design which have been incorporated into the conceptual HWR to improve uranium utilization or economic performance. The potential technical problem areas introduced by these design changes are reviewed in Section 6.3.2.

6.3.1 Uncertainties in the Basic Candu Concept

As mentioned above, there appears to be no major technical problems associated with the basic Candu concept which cannot be resolved by normal engineering design, or which are disordinately large as compared to similar problem areas in light water reactors. However, like the LWR, there are several areas where the long-term performance remains to be fully demonstrated. The most significant of these areas are summarized briefly below, for completeness.

- Pressure Tube Lifetime

Pressure tube problems have occurred in the past, first at Pickering Units 3 and 4 due to an improperly rolled joint, and more recently on all operating Candu units as a result of greater-than-expected growth of the pressure tube. There are a number of other potential problems which could limit the useful lifetime of the pressure tubes, such as pressure tube corrosion, wear, or hydriding. Each of these potential problems has been evaluated and allowance for these phenomena has been included in the pressure tube design. However, the Candu concept is still quite a distance from demonstrating that pressure tubes will actually last their full 30-year design lifetime (a similar comment, of course, can also be made with respect to various LWR components).

Although the lifetime of the Candu pressure tubes is considered an area of technical uncertainty, it should be noted that pressure tubes can, and have been, replaced. The main implication of such potential pressure tube problems is consequently not one of safety but of economics in areas such as potentially higher maintenance requirements and lower availability of the units as the systems age.

- Steam Generator Integrity

In contrast to light water reactors, Candu steam generator reliability has been excellent; there has been only one incident of reported steam generator tube leakage (Reference 6-1). The question remains, however, of whether this quite satisfactory steam generator performance can be maintained. Although there is no indication that steam generator performance will not remain satisfactory in the future, the consequences of poor steam generator performance are potentially more

severe in the Candu-PHWR than in light water reactors, because of the economic consequences of heavy water losses. It should also be noted that both the standardized Candu-600 unit and the conceptual HWR design developed in this report employ steam generators considerably larger than those which have been demonstrated on operating Candu plants. Consequently, the performance of such large-scale steam generators remains to be demonstrated.

° Tritium Concentration

The concentration of tritium in the reactor coolant and moderator will increase to higher levels with continued reactor operation, and as a result could potentially increase the radiation dose received by plant personnel during maintenance. Although the operational exposure at operating Candu plants is only a small fraction of the allowable limit (Reference 6-2), about 1/3 of this dose is attributable to tritium, and hence occupational exposures might be expected to increase as the tritium content builds up.

It may be necessary sometime in the life of the PHWR to either replace the D_2O inventory or to remove tritium from the D_2O coolant and moderator in order to limit plant exposures. The former option, though straightforward, is expensive. Technology for separating tritium from D_2O , based upon cryogenic distillation of D_2O gas, has been demonstrated. AECL has been developing a conceptual design to provide process simplification, with a demonstration plant proposed to serve the NRX, NRU, NRD and Douglas Point reactors. Thus, the buildup of tritium concentration with reactor operation is thought to pose no fundamental problem, but may necessitate expenditures sometime in the future to maintain tritium concentrations at acceptable levels.

6.3.2 Conceptual Design Technical Problem Areas

Although as noted above there are possibly several areas of technical uncertainty associated with the long-term economic performance of the basic Candu reactor (as there would be for any reactor concept), these areas are not thought to be of major concern or to significantly impact R&D requirements. Of greater concern are the modifications to the basic Candu design which have been incorporated into the conceptual HWR described in this report so as to improve its uranium utilization and economic performance. In several instances, these design modifications reach beyond proven HWR technology. In particular, the design modifications incorporated to achieve the conceptual HWR design create several major technical problem areas which will require RD&D: large plant (1260 MW(e)) design, moderator cooling system, slightly-enriched fueling, and increased primary system pressures and temperatures. Each of these potential technical problem areas are reviewed briefly below.

° Large Plant Design

The conceptual HWR design attains a net power output of 1260 MW(e), a power output comparable to that of large light-water reactors. Because of economies of scale, the deployment of large (1000 MW(e) or greater) units is thought to be essential to the commercial viability of the concept in the U.S., since smaller units are unlikely to compete economically with larger light-water reactors. The largest Candu reactor currently in operation is the 750 MW(e) Bruce units, while slightly larger units of 800 MW(e) capacity are currently under construction at Darlington. Both the Bruce and Darlington units, as well as the standardized Candu-600, employ two coolant loops. The net electrical generating capacity of Candu-type reactors could be increased most readily by the

addition of a third coolant loop. If a third coolant loop were added, net generating capacity could be increased to the 1000-1200 MW(e) range, using components of a size which has been, or which shortly will be, demonstrated. While increasing the number of coolant loops is the simplest approach to increasing station capacity (and consequently may be the preferred approach in the initial HWR units introduced to the U.S.), the conceptual HWR design described in this report is based upon the use of a two-loop configuration. As discussed in greater detail in Section 4.5, plant capital costs are reduced when the two-loop configuration is employed, both because a fewer number of larger size components (e.g., steam generators and pumps) are utilized and because containment size can be reduced.

Although the component sizes employed in the two-loop configuration of the conceptual HWR design are, in general, comparable to those used in large pressurized water reactors, these components are, nevertheless, considerably larger than those which have been employed to date in Candu units. The primary coolant pumps employed in the conceptual HWR design are larger than similar pumps employed to date in Candu or light-water reactors. The conceptual HWR design employs 4 pumps, each having an operating load of 9,275 kW; the largest pumps employed to date in Candu units are the 8,200 kW Bruce units and the largest for PWR's is 6600 kW. Consequently, some development will be necessary to design the slightly larger pumps employed in the conceptual design. However, the development of such larger pumps is thought to be straightforward, requiring some RD&D, but offering no fundamental technical problems. It should also be noted that a larger number of smaller pumps could also be employed in the conceptual design; larger pumps were selected for the conceptual design because of the lower capital cost.

The conceptual HWR design employs 4 steam generators, each of which has an evaporation rate of about 2,000 Mg/hr. Although these steam generators are not large compared to some PWR units (some PWR units will employ steam generators with evaporation rates twice as large), these steam generators are quite a bit larger than those currently in operation in Candu units. The largest Candu steam generators currently in operation are those of the Bruce Station, which have an evaporation rate of 525 Mg/hr; the standardized Candu plant (e.g., Gentilly 2 or Point Lepreau) will have somewhat larger steam generators, with each having an evaporation rate of about 930 Mg/hr. The development of the larger steam generators of the conceptual HWR design is not thought to pose fundamental technical problems, since larger steam generators have been designed for light-water reactors. However, some caution is warranted here because of the importance of steam generator integrity in limiting irrecoverable heavy-water losses in the HWR concept. It is essential that the quite satisfactory performance of operating Candu steam generators be retained in the larger components of the conceptual HWR design. It should also be noted that it is possible to employ a larger number of smaller steam generators (for example, 8 steam generators of the Candu-600 size could be used instead of the four larger steam generators of the conceptual design). Although this would largely eliminate the uncertainty associated with the performance of the larger steam generators, it would result in a significant increase in plant capital cost.

° Moderator Cooling System

All operating Candu plants employ once-through cooling, while the conceptual HWR design is based upon closed cycle cooling (mechanical cooling towers) which was a basis under

which the conceptual HWR design was developed. The difference between once-through cooling and closed-cycle cooling is particularly significant to the design of the moderator cooling system. The moderator circuit operates at near atmospheric pressures, and hence is limited to operating temperatures less than about 200°F. The size of the heat exchanger used to cool the moderator depends upon the temperature differential between the moderator and the cooling water. Once-through cooling typically provides a relatively low temperature heat sink, allowing the moderator heat exchanger to be of reasonable size. However, with closed-cycle cooling the temperature of the cooling water increases, thus necessitating a larger and more costly moderator heat exchange system. In order to minimize the size and the cost of the moderator cooling system, an ammonia absorption system (refrigeration system) was selected for cooling the moderator fluid, rather than employing conventional water-cooled heat exchangers. Although such systems are in widespread use in non-nuclear applications, R&D will be required to develop the system for moderator cooling application. Alternately, a more conventional system could be employed, but here too some design development would appear warranted in order to optimize this system for closed-cycle cooling.

◦ Use of Slightly-Enriched Fuel

The conceptual HWR employs slightly-enriched (1.2 w/o) fuel. The use of such slightly-enriched fuel is considered a fundamental feature of the conceptual design; improved U_3O_8 utilization and fuel-cycle costs result directly from the use of slightly-enriched fuel. In addition, a number of other design modifications, such as the increase in primary system pressure, which contributes to the reduced capital cost of the conceptual HWR are facilitated by the use of

slightly-enriched fueling; these modifications are unattractive when natural uranium fueling is employed because of the reduced burnup.

The use of slightly-enriched fueling, however, does introduce a number of possible technical problem areas. As a result of the use of slightly-enriched fueling, the discharge exposure increases to approximately 20,000 MWd/t, as compared to the 7,500 MWd/t achieved with natural uranium fueling. The evaluation of fuel performance (c.f. Section 4.8 - Fuel Design Assessment) indicates that Candu fuel of existing design should be capable of accommodating this increase in fuel exposure. However, irradiation proof testing will be required to demonstrate that satisfactory fuel performance can be obtained at these higher burnups. Of course, it will be possible to develop a fuel design capable of obtaining these burnups should the irradiation testing indicate that some modification of current Candu design is required, since light water reactor fuel is routinely designed for higher discharge exposures. However, such design modifications could reduce the achievable discharge exposure or in some other fashion compromise the performance of the conceptual HWR design, and hence it is important to the commercialization of the concept to demonstrate that acceptable fuel performance can be obtained at higher burnups with little or no modification to present Candu fuel design.

The use of slightly-enriched fuel will also result in an increase in core peaking factor (c.f. Section 4.1). An allowance for this increase in peaking factor has been incorporated into the conceptual design by reducing slightly channel power density, by employing enrichment zoning to reduce local peaking, and by reducing the number of fuel assemblies replaced at each refueling visit. More detailed

evaluations of the power peaking anticipated with slightly-enriched fuel, however, will be required to clearly establish whether or not further design modifications will be required. It is believed that the use of slightly enriched fueling, with the concomitant increase in core peaking, does not constitute a fundamental technical problem. However, further engineering development and demonstration will be required before slightly-enriched fueling can be commercialized.

° Increased Primary System Pressure/Temperature

In the conceptual HWR design, the primary system pressure has been increased to 2250 psia (a value comparable to that of LWRs) in order to allow primary temperatures and secondary temperatures and pressures to be increased without adversely affecting thermal margin. The increase in secondary pressure increases the net electrical efficiency of a plant and, because of the increased electrical output, results in a significant reduction in the plant capital cost (\$/kWe). In order to accommodate the increased primary system temperature and pressure, the wall thickness of the pressure tube has been increased, so as to preserve the same hoop stress present in existing Candu designs.

Review of pressure tube design considerations (c.f. Section 4.2) indicates a likelihood that the Zr-Nb pressure tubes can operate successfully under the higher coolant temperatures and pressures postulated for the reference conceptual design. However, as noted in that section, the data base on the performance of pressure tubes at these temperatures and pressures is inadequate. The performance of pressure tubes at the elevated temperatures and pressures postulated for the conceptual design are thus considered a technical problem area which will require considerable RD&D prior to commercialization.

Licensability of the conceptual HWR design has been discussed in some detail in Section 5.5. Questions of licensability, however, are summarized briefly in this section because of the potentially significant impact on the prospects for commercial introduction.

Licensability of the conceptual HWR design in the U.S. can be conveniently divided into two aspects: the licensability of the basic Candu concept and the licensability of the various design modifications to the basic Candu concept which have been incorporated into the conceptual design to improve uranium utilization or economic performance. In addition, a commercialization assessment should consider the potential implications of the licensing process itself; this licensing process introduces a significant element of uncertainty both as to whether additional design modifications may be required (which could potentially compromise economic performance), and in the time and effort required to license the HWR concept in the U.S.

The safety and licensability of a reactor concept are due partially to its engineered features and partially due to its inherent characteristics. Although the engineered features of the Candu reactor are viewed to be satisfactory from a licensing and safety standpoint, the basic approach employed in developing the conceptual HWR design was to replace these engineered features with PWR-type designs so as to minimize questions of licensability under U.S. criteria and construction practices. Thus, such aspects of the engineered features as containment design and liner, emergency core cooling system, seismic restraints, etc., have been replaced with U.S.-PWR type systems. Consequently, licensability questions should center upon the inherent characteristics of the HWR and upon the design changes which have been incorporated into the conceptual HWR to enhance uranium utilization and improve power cost economics.

The inherent characteristics of the plant are defined to be those features which cannot be readily altered by engineering changes. When considering the inherent features of a reactor concept, it must be recognized that certain features will be less satisfactory and certain features more satisfactory than those of other alternatives. Of prime concern is the overall implication to safety when all the unique attributes of the system, both positive and negative, are considered. The HWR concept has a number of inherent attributes which contribute to increased plant safety: the use of the pressure tube configuration with the resulting mitigating effect on the consequence of the loss-of-coolant accident, the ability to detect and replace failed fuel because of the on-line refueling feature, and the redundant shutdown systems provided in the HWR are considered to be the most significant of these positive attributes. Licensability assessments, however, are apt to dwell on those inherent attributes which appear less satisfactory than those of alternately available systems, such as the light water reactor.

The inherent attributes of the HWR which might be viewed as being less satisfactory than those of the LWR are the use of thin-walled pressure tubes as the primary pressure boundary, the positive void coefficient of reactivity, and increased tritium releases to the environment.

These inherent features are reviewed in greater depth in Section 5.5.2.2 where it is concluded that on balance the inherent features of the HWR provide at least an equivalent degree of safety as is present in light-water reactors, and hence the inherent features of the HWR should not prove a fundamental barrier to U.S. licensability; a similar conclusion was reached in a preliminary NRC review of the HWR concept (Reference 6-3). However, it should be recognized that the successful licensing of the HWR in the U.S. will probably necessitate some modification of the licensing

criteria and regulations which have been developed in the U.S. for light water reactors. The licensability of the HWR concept and the degree to which further design changes may be necessitated by licensing will depend, in part, on the degree to which the licensing process recognizes the inherent differences between light water and heavy water reactors, and on the degree to which the licensing process attempts to force HWR design to conform to criteria which, having been developed for LWRs, may not be wholly applicable to the HWR concept. If an inflexible licensing approach were adopted, the resulting increase in plant capital cost and/or degradation in HWR performance could result in a commercially unviable concept.

The licensability of the design modifications which have been incorporated into the conceptual HWR design to improve uranium utilization or power cost economics (i.e., primarily slightly enriched uranium fueling and the increase in primary system pressure) are discussed in detail in Section 5.5.9. Of primary concern is the effect of the change of primary coolant pressure and temperatures on pressure tube integrity, and upon the integrity of the rolled joint which is used to mechanically fasten the pressure tube to its end fittings. Allowances for increased primary system pressure and temperature have been made to the extent possible with available information in evolving the conceptual HWR design. However, it is recognized that the data base on pressure tube material performance at the postulated pressures and temperatures of the conceptual design is inadequate, and that RD&D to determine pressure tube material strength and corrosion resistance will be necessary. The increased primary system and temperatures (as well as the use of slightly-enriched fuel) can also alter the core transient characteristics during postulated accidents. The effect of the design changes on important core transient parameters is discussed in Section 5.5.8, where it is concluded that the design modifications result in either insignificant

or advantageous changes in these parameters; consequently, no adverse effect on the consequences of various transients and other postulated accidents are anticipated. However, it is clear that a thorough reevaluation of the consequences of various transients and postulated accidents will be required to clearly establish whether the conceptual HWR fully meets appropriate U.S. acceptance criteria. The development of such analyses should be part of the RD&D effort leading to the eventual commercialization of the concept.

The discussion above has attempted to summarize a number of technical licensability concerns which could potentially impact the prospects for the commercial introduction of the HWR in the U.S. These technical concerns are discussed in greater detail in Section 5.5. In addition to these technical questions of licensability, the licensing process can also introduce nontechnical and more subtle barriers to commercialization. In particular, the licensing process introduces significant uncertainties which cannot be fully resolved by a discussion of the technical merits of a system. The licensing process has significantly impacted the capital cost of light water reactors, and it has proven difficult to anticipate or predict future licensing requirements or their cost ramifications. As discussed in Section 6.6 - Commercial Viability, there appears to be an economic incentive for the HWR, particularly as the price of U_3O_8 increases. However, this economic incentive is viewed as relatively small compared to the capital cost increases which could potentially occur as a result of licensing reviews.

In developing and pricing the conceptual HWR, an attempt has been made to incorporate as fully as is possible what is believed to be U.S. NRC requirements and guide lines. To the extent that this study has successfully anticipated these licensing requirements, the LWR/HWR capital cost differential should be relatively insensitive

to evolving licensing requirements. However, capital cost development and the impact of licensing requirements on the economic competition between the HWR and LWR are so uncertain that it would be, at best, speculative to conclude that an HWR licensed in the U.S. would be an economically viable alternative to current light water reactors.

A second concern introduced by the licensing process is the length of time required to obtain licensing approval. The period of time required to obtain licensing approval has important impacts both on the capital cost of the plant (due to escalation) and upon the ability of the utility to meet projected load growths. The period of time currently required to license a light water reactor is rather long, in spite of the fact that most elements of the design have been subject to extensive evaluations and public hearings in the past. The time period required to license an HWR in the U.S. could prove considerably longer both because of the uniqueness of many aspects of the HWR design, and because the licensing of a new concept could involve extensive public hearings and interventions.

In summary, HWR licensability and the licensing process are viewed to introduce significant uncertainties in the economic (and to some degree also the uranium utilization) performance of a heavy water reactor sited in the U.S. It is concluded that this uncertainty is sufficiently large so as to effectively preclude a definitive conclusion on the potential for the commercial introduction of the HWR into the U.S. at the present time, and it is believed unlikely that a U.S. utility would elect to commercially deploy a heavy water reactor because of these uncertainties. The construction of a U.S. sited demonstration facility, complete with full commercial-type licensing is consequently viewed as a prerequisite to commercial introduction of the HWR into the U.S.

Estimates of RD&D costs are summarized in Table 6.5-1. The RD&D is divided into generic technology programs and the reactor plant demonstration program, both of which are considered to be elements of the RD&D cost. Two sets of costing numbers are provided for the reactor plant demonstration, corresponding to the two basic introduction scenarios discussed in Section 6.2. The generic technology programs are intended to provide the basic information required to embark upon the HWR reactor plant demonstration in the U.S. and the cost of these generic programs are thought applicable to either of the two demonstration paths.

Before discussing these RD&D costs, it is appropriate to briefly review the basis for their development. The major assumptions in developing these costs are summarized below.

- ° A mission oriented RD&D program has been assumed, directed explicitly toward the reactor plant demonstration. Thus, only a relatively modest effort is devoted to generic technology programs; the majority of the RD&D effort is focused explicitly on the design and licensing of the demonstration plants.
- ° The RD&D programs are aimed at developing a single conceptual HWR design for commercial application. The development of multiple designs (e.g., to develop HWR offerings from multiple vendors) would add additional RD&D cost.
- ° RD&D costs are predicated on a joint U.S./Canadian venture. Generic technology programs are thus limited almost exclusively to the development of the basic technology necessary to support the design changes incorporated into the conceptual HWR plant. With the exception of safety and licensing development, such generic technology programs would not be

necessary to deploy a standard Candu unit since this information would be available from the Canadian program. Likewise, safety and licensing and first-of-a-kind engineering costs are minimized because of the assumption of a joint U.S./Canadian effort, although substantial RD&D is still thought necessary for licensing development and design changes required for U.S. siting.

- The RD&D costs shown in Table 6.5-1 are the total industry and government costs associated with the actual design and construction of the demonstration plants. Excluded from these RD&D estimates are such indirect costs as DOE program administration and the internal costs of reviews by the NRC and other Federal agencies.
- Also excluded are the costs of developing a domestic manufacturing infrastructure (for example, for D₂O production or for the manufacture of refueling machines), under the assumption that critical components and materials could be imported from Canada where domestic manufacturing capacity was lacking.
- Royalties and/or fees for the use of Canadian technology were also excluded from the RD&D estimates.
- The complete identification of potential technical problems and the development of detailed work programs to resolve these problems are necessary to accurately estimate RD&D costs. Although this study did succeed in identifying a number of areas which will require RD&D, the conceptual nature of the study and the lack of detailed quantitative safety analyses preclude such a thorough development of RD&D costs. RD&D cost estimates are thus somewhat subjective. Since they must include some allowance for unforeseen problem areas, a range of estimated cost is consequently included in

Table 6.5-1... The low cost estimate generally consists of the expected cost of RD&D programs necessary to resolve identified problem areas. The "expected value" includes additional RD&D funding to resolve unidentified problem areas; since it is impossible to cost such unidentified problems, these additional RD&D costs are highly subjective and based primarily on judgment. The "high" cost estimate is based upon the assumption that additional or more fundamental problems will be encountered which will require additional RD&D.

Figure 6.5-1 presents a milestone schedule for the generic technology programs. As mentioned above, these generic technology programs are aimed primarily at developing the analytical and experimental data base required in support of the design modifications incorporated into the conceptual HWR design. Also included in these generic technology programs are the preliminary evaluations of design and licensing to U.S. criteria in order to provide the basis for the subsequent detailed design and licensing development required as part of the reactor plant demonstration.

Three areas are included under basic research: the development of safety-related physics parameters for slightly-enriched fueling, and initial evaluations of the loss-of-coolant transient and the seismic event. These evaluations are primarily analytic in nature and are considered necessary in the early part of the RD&D program to determine if design modifications, extensive modeling revisions, or verification testing are required. Although Canadian cooperation is assumed, the cost of this R&D effort is significantly impacted by the degree to which Canadian involvement in U.S. licensing evaluations is assumed. The expected values are based upon a joint U.S./Canadian program and the identification of no major technical problem areas. The high range reflects minimum Canadian participation and/or unexpected problems which would

require extensive evaluation to demonstrate conformance with U.S. criteria or to develop design changes which would force such conformance. The low range reflects a highly successful program with little or no effect on plant design.

The second item, core and fuel development, is intended primarily to extend the data base on pressure tube and rolled joint performance to the conditions of the conceptual design, and to demonstrate the higher burnup associated with slightly enriched fueling through irradiations in existing Canadian plants. The range of cost represents primarily the uncertainty in the cost of these RD&D programs; in each case, it has been assumed that major modifications to the basic Candu concept or to pressure tube materials are not required.

No RD&D costs are included under the category of plant systems development, since these systems are considered to be state of the art in the U.S., requiring only the normal engineering development required as part of the design of the new plant. Plant system development costs are consequently covered in the category of first-of-a-kind engineering under the reactor plant demonstration program.

The component development program includes the RD&D necessary to develop the larger steam generators, primary coolant pumps, valves, seals, etc., required for the conceptual HWR. This RD&D program does not include the cost of first-time engineering or tool development (included in the first-of-a-kind engineering within the reactor plant demonstration program). The range represents the uncertainty in determining the extent of these programs without detailed programmatic development.

The items included under Safety and Generic Licensing are intended to include only truly generic R&D, since plant specific safety

and licensing has been included as an element in the reactor plant demonstration program. Included here are such items as criteria development, code changes, and the development of basic data and programs which are a necessary precursor to the more detailed safety and licensing evaluations performed as part of the reactor plant demonstration.

The reactor plant demonstration program RD&D costs, tabulated in Table 6.5-1, include cost estimates for the two basic introduction scenarios discussed in Section 6.2. The first scenario (Case A) assumes that the initial HWR demonstration plant is as similar as practical to Candu plants currently under construction in Canada. This demonstration plant would have an electrical capacity of about 600 MW(e) and would operate at the primary pressures now employed in Candu units. Design changes would be limited to those dictated by U.S. licensing and construction practices, so as to most fully utilize proven Canadian technology and safety and licensing development. For this unit, safety and licensing RD&D costs are thus incurred primarily from the modification of existing Candu designs for U.S. siting and from the subsequent safety analysis of that design, along with concomitant SAR preparation and defense. First-of-a-kind engineering costs reflect the cost of component development for U.S. siting, as necessitated by the safety and licensing development and by consideration of U.S. construction practices.

The second plant in the Case A introduction scenario would seek to demonstrate the features of the conceptual HWR plant, including the larger (1260 MWe) plant size and the increase in primary system temperature and pressure. Safety and licensing and first-of-a-kind engineering costs for this second unit will also be substantial because of the departure from proven Canadian technology. In particular, the increase in primary system pressure will necessitate significant safety and licensing development, while

first-of-a-kind engineering will be required both because of the higher pressure and larger plant size.

In the second introduction scenario, Case B, it has been assumed that the lead plant seeks to demonstrate all features of the conceptual HWR. Safety and licensing and first-of-a-kind engineering RD&D will thus be significantly larger than that required for either of the two plants in the Case A introduction scenario, since this lead plant now attempts to demonstrate both the design modifications incorporated into the conceptual HWR and U.S. licensing. RD&D costs for the second unit in this introduction scenario, however, should be low since all elements of the plant design will have been demonstrated by lead plant design and licensing; some additional RD&D should nevertheless be expected since design, licensing and construction experience with the initial unit will undoubtedly motivate further design changes.

The range in safety and licensing and first-of-a-kind engineering RD&D cost reflects the uncertainty in the required safety and licensing effort and of the potential ramifications of the safety and licensing review on plant design. The plant costs shown in Table 6.5-1 include the cost of the initial heavy water inventory, but are exclusive of interest and escalation during construction and of owner's cost. No attempt has been made to develop a range of plant costs, but for estimating purposes a range of minus 5% to plus 10% appear reasonable for the low and high range values, respectively.

TABLE 6.5-1
ESTIMATED RD&D COSTS
(Millions of January 1978 Dollars)

	<u>Expected Value</u>	<u>High</u>	<u>Low</u>				
A. Generic Technology Programs							
1. Base Research	3.0	10.0	1.5				
2. Core and Fuel Development	6.5	8.5	5.0				
3. Plant Systems Development	--	--	--				
4. Component Development	1.5	2.1	1.0				
5. Safety and Generic Licensing	<u>3.0</u>	<u>5.0</u>	<u>2.0</u>				
	14.0	25.6	9.5				
B. Reactor Plant Demonstration							
	<u>Case A</u>	<u>Case B</u>	<u>Case A</u>	<u>Case B</u>	<u>Case A</u>	<u>Case B</u>	
1. First Unit							
Safety & Licensing	25	50	35	65	10	30	
First-of-a-Kind Engineering	15	40	30	60	10	30	
Plant Costs**	<u>637</u>	<u>910</u>	*	*	*	*	
	677	1,000					
2. Second Unit							
Safety and Licensing	30	5	40	10	25	5	
First-of-a-Kind Engineering	35	5	50	10	30	5	
Plant Costs**	<u>910</u>	<u>910</u>	*	*	*	*	
	975	920					

Case A assumes that the first unit (600 MWe) is similar in design to operating Candu plants; the second unit is a 1260 MWe conceptual HWR.

Case B assumes that both the first and second units are 1260 MWe conceptual plants.

* Estimated to be 95% and 110% of the expected values for the low and high range, respectively.

**Excludes interest and escalation during construction and owner's cost; cost of initial D₂O inventory is included.

Figure 6.5-1
MILESTONE SCHEDULE FOR GENERIC TECHNOLOGY PROGRAM

1. COMPLETE DNB TESTING
2. PHYSICS EXPERIMENTS TO VERIFY NUCLEAR DATA AND METHODS
3. SEISMIC EVALUATION COMPLETED
4. LOCA EVALUATION COMPLETED
5. PHASE ONE FUEL TESTING COMPLETED
6. PHASE TWO FUEL TESTING COMPLETED
7. PRESSURE TUBE/ROLLED JOINT DEVELOPMENT COMPLETED
8. STEAM GENERATORS AND PRIMARY COOLANT PUMP DEVELOPMENT COMPLETED
9. SAR PREPARATION

In order for a concept to be commercially viable there must be a manufacturer willing to develop and market the concept, and a utility customer willing to purchase the marketed reactor design. There are a number of factors relating both the technical/economic viability of the HWR concept and to the more general economic/political status of the nuclear industry which weigh heavily on the likelihood that manufacturers will develop and utilities will purchase alternate reactor designs such as the HWR.

The development of a reactor design as a commercial offering typically necessitates significant capital commitments. Investments are required to construct manufacturing capacity, to perform the detailed engineering of the plant design, and for licensing development. Although these costs could be minimized by employing to the greatest extent possible existing Candu manufacturing capacity and Candu design development, the investment necessary to offer commercially the conceptual HWR design in the U.S. would still be significant. In order for a manufacturer to justify investing in the development of an alternate reactor concept, there must be a reasonable expectation that the investment would be recovered, i.e., that the market for the new reactor concept will be sufficiently large and long in duration to allow a reasonable return on invested capital.

Review of the technical problem areas and manufacturing development requirements of the heavy water reactor indicates no unsurmountable problem to the development of a heavy water reactor for domestic application, although there are certainly many uncertainties, particularly with respect to design changes which may be necessary for licensing. Indeed, the prospects for the successful development of a commercial U.S. HWR are greatly enhanced by the commercial status of the Candu reactor and by the existing manufacturing

base which has been developed for these reactors; the prospects of successfully developing a commercial HWR for domestic application are greatly enhanced by the assumption that Canadian technology and manufacturing capability would be available to the U.S. program. Marketing surveys (Reference 6-4) also indicate that U.S. industry is capable and willing to undertake such development ventures, provided, of course, a reasonable return on investment is anticipated. Consequently, one can conclude that the first ingredient to commercial viability, that of developing and marketing an HWR for U.S. siting, is in principle possible; whether or not an HWR design will be developed and marketed depends upon the perception of whether the market for such HWRs will be sufficient to warrant the development in the HWR concept. Simply stated, there appears to be no reason why the U.S. industrial sector could not or would not develop and market an HWR for domestic application, provided that doing so was perceived by them as a prudent investment.

However, for a number of reasons discussed below, the market for HWRs in the U.S. is currently perceived to be small, or at best highly uncertain, and so the deployment of an HWR design and supporting manufacturing infrastructure for U.S. siting is viewed as a high-risk venture. Consequently, the development of a domestic HWR design and manufacturing capability is not expected to evolve in the absence of further incentives. Thus, it appears clear that a period of precommercial deployment would be necessary before the heavy water reactor can be commercially introduced into the U.S. During this period of precommercial deployment, it will be necessary for the government to, in one fashion or another, reduce this risk to acceptable levels (i.e., to levels comparable to the risk of other investments). This will undoubtedly require significant government funding of the initial HWR units constructed in the U.S. Fully commercial introduction of HWRs could proceed once the market demand for this type of reactor became clearly established.

6.6.1 Potential Demand for the HWR Reactor

It is concluded that the current demand (market) for the HWR in the U.S. is small, and that the market potential for the HWR is likely to remain small in the near term (perhaps the next 10 or 20 years), unless some form of incentive to deploy HWRs is established. This perceived lack of domestic market for heavy water reactors makes it doubtful that private industry will develop commercial HWR designs or a manufacturing infrastructure without significant governmental involvement. The supposition that a market for heavy water reactors does not presently exist in the U.S., nor is likely to evolve in the near future, is based partially upon the assessment of the economic viability of the conceptual HWR design and partially upon the present political/economic status of the nuclear and utility industries.

Utility decisions to purchase a particular reactor design are based upon a number of considerations, the great majority of which pertain to the economic viability of the concept either directly or indirectly through reactor performance considerations. A number of important factors are listed below:

- Plant reliability
- Operating flexibility
- Licensability
- Evaluated power cost
- Uncertainties in reliability, power cost, and licensing

Reliability is perhaps the single most important factor in utility decision making; reliability, of course, is a major factor affecting the cost of power. As discussed in Section 5.4, the reliability (capacity factor) of operating Candu plants has been excellent, and this highly satisfactory operating experience should provide greater assurances than is possible for many alternate concepts

which have still to achieve significant operating experience. However, the conceptual HWR is predicated upon the use of slightly enriched uranium fueling and of increased primary system temperatures and pressures. Both these design changes introduce elements which can affect the reliability and performance of the HWR, so that it becomes questionable whether the excellent performance of operating Candu reactors can validly be extrapolated to the conceptual HWR design. It is also recognized that other changes which may be necessitated by the licensing process could seriously compromise the performance, and possibly the reliability, of the conceptual HWR, either by necessitating some reduction in power output from the plant or through design changes which might reduce achievable discharge burnup. These uncertainties in the performance and reliability of the conceptual HWR design are expected to hamper the acceptance of the HWR concept in the U.S. Until these uncertainties are resolved by the operation of a U.S.-sited demonstration plant, utilities will tend to favor light water reactors whose performance and reliability have been established through years of domestic operating experience.

Operational flexibility, or the ability of a plant to accommodate expected maneuvering or load changes, is a second consideration which can affect a utility's choice of reactor concept. The operational performance characteristics of the HWR are discussed in Section 5.5.4. Overall, these operational characteristics are viewed to be comparable to those of the light water reactor, and so considerations of operating flexibility typically are not expected to be a strong influence in the choice between the LWR or HWR concepts.

Licenseability is considered to be an extremely important aspect of commercial viability, and one which places the HWR, or any other alternate system unlicensed in the U.S. at a disadvantage with respect to the light water reactor. As discussed in Section

6.4, the conceptual HWR design is viewed to be eventually licensable in the U.S., but there remains the possibility that significant design changes may be required as a result of the licensing process. As a result, HWR power costs may be higher than predicted in this evaluation, either because of increases in plant capital cost or through adverse impacts on plant reliability and performance. There is also considerable uncertainty as to the length of time it would take to license a heavy water reactor in the U.S.; the length of time required for licensing and construction affects both the plant capital cost, through interest during construction and escalation, and also impacts on the ability of a utility to bring capacity on line on a schedule consistent with anticipated load growth. It is noted that over the past years the licensing process has contributed to the escalation of LWR capital cost and to a lengthening of the time period required to license and construct plants. The uncertainty as to whether the HWR will be subject to similar escalations as a result of licensing is a barrier to HWR acceptance in the U.S. It is concluded that licensability is likely to be a very serious barrier to commercialization; the commercial-type licensing of a U.S.-sited large plant demonstration is considered to be a prerequisite to eliminating the barriers to commercial viability imposed by uncertainties in plant licensability.

Evaluated power cost is an important consideration in choosing between two well-established alternatives, e.g., in choosing between two different LWR designs. The evaluated power cost becomes a less significant parameter in the selection between two alternatives, when the difference in evaluated power cost between the alternatives is small in comparison to the uncertainty in economic performance. This is presently the situation for the HWR, since the economic incentives (or disincentives) for HWR deployment in the U.S. are small compared to the potential economic ramifications of uncertainties in plant performance, reliability,

capital cost, and the impact of the U.S. licensing process arising from a lack of direct U.S. utility experience with licensing, constructing, and operating a plant of this type. Consequently, it is not anticipated that a domestic utility would elect or reject the HWR concept based upon the evaluated power cost presented in Section 5.4; these evaluated power costs should be considered as no more than an indication of the potential long-term economic performance of the concept, assuming that the dominating uncertainties are favorably resolved.

As Section 5.4 indicates, the conceptual HWR design is expected to have a slight economic disadvantage at current uranium prices, to have power cost comparable to that of the LWR when the price of uranium rises to about \$55/lb (a value which might be anticipated by the time HWRs were deployed in the U.S.), and to exhibit lower power cost than the LWR for higher uranium prices. The economic evaluations thus indicate no current or near-term incentive for the HWR in the U.S. In the longer term, as uranium prices increase, the evaluated power costs do indicate an economic incentive. However, this economic benefit is small compared to the uncertainties in capital cost (as impacted by performance and licensability considerations), and hence provides little incentive to underwrite the substantial financial risk associated with the aforementioned uncertainties.

In summary, the question of the commercial viability of the conceptual HWR in the U.S. is clouded by uncertainties - uncertainties in the HWR plant capital cost development, uncertainties in the expected reliability and performance of the conceptual design with the concomitant uncertainty in power cost, and uncertainties of the effect of the licensing process on plant capital cost and performance and reliability. The HWR concept will not be commercially viable in the U.S. until these uncertainties are reduced; the construction and operation of a large U.S.-sited demonstration plant is required. Furthermore, there appears to be no strong near-term economic benefit for the HWR (assuming the resolution

of the aforementioned uncertainties proves favorable), with the consequence that strong governmental involvement will be required (for development, HWR large plant demonstration, and also possibly for a precommercial deployment period) until uranium prices rise sufficiently to provide a commercial incentive for HWR deployment. In the longer term, the HWR could be a commercially viable concept as the price of uranium increases. The evaluation performed in Section 5.4 indicates that this break-even price of U_3O_8 (above which there is an incentive for HWR deployment) is about \$55/lb., but it is also recognized that this break-even uranium price could increase significantly should technical or licensing considerations result in increased plant capital cost or adversely affect plant reliability or performance. Thus, the long-term commercial viability of the HWR depends crucially upon the success of the initial demonstration program in establishing satisfactory HWR plant performance and capital costs.

A second consideration is the general political and economic climate prevalent in the nuclear industry today. The ability to introduce a new reactor concept is greatly enhanced by a healthy nuclear industry. In the past, there had been little public opposition to nuclear power and expectations of large and continuing growth in nuclear capacity. Under these circumstances, the potential for new sales was high and it was reasonable to assume that capital invested in new reactor designs and manufacturing capability was well invested. Currently, the political and economic climate of the nuclear industry is severely depressed, and there is little reason to believe that the situation will be altered in the near term. In particular, current DOE projections of nuclear capacity indicate a much less rapid expansion of nuclear capacity. On the high side of the predicted range, barely enough capacity expansion is predicted to support the present U.S. light water industry; with these "high" nuclear capacity projections, the full industrial capability in the U.S. is significantly under-utilized. In the case of the "low" estimates of required nuclear capacity, it is unlikely that there is sufficient

market to support a viable nuclear industry at all. In the face of presently under-utilized industrial capacity and limited expectations of a future market, it is difficult to envision private industry investing in the development or manufacturing infrastructure of an alternate reactor concept.

1. The conceptual HWR is not likely to be commercially introduced in the U.S. today. From a U.S. utility viewpoint, there is little economic incentive for HWR deployment, while uncertainties in conceptual HWR plant capital cost, plant performance and reliability, and the impact of the licensing process result in significant risks that the power costs of the conceptual HWR design may prove to be higher than established in this evaluation. The risks inherent in introducing a new reactor type (including one that is commercially successful elsewhere) suggest that commercial introduction would have to be preceded by a demonstration project.
2. In order for the HWR concept to potentially become commercially viable in the future, a large-scale U.S.-sited demonstration will be required to resolve the many uncertainties associated with HWR cost, performance, and licensability. Since, as mentioned above, there appears to be little near-term incentive for the HWR, this large plant demonstration (including prior R&D) will require strong government support.
3. In the longer term, the HWR concept has potential for becoming commercially viable in the U.S. This conclusion is derived from the observation that HWR power costs are less sensitive to the price of uranium than are those of the LWR; consequently, there is some price of uranium above which the HWR becomes more economic than light water reactors. This uranium price, however, is highly dependent upon the relative capital cost and performance of the HWR with respect to its prime competitor, the light water reactor. Should the HWR demonstration program demonstrate the capital costs and performance expectations cited in this report, the HWR could become commercially viable at a uranium price of about \$55/lb.

However, should the development program indicate that the capital cost of the HWR is higher than that developed in this study, or that design modifications lessen HWR reliability (capacity factor) or performance, the break-even uranium price may rise to a value above which converter reactors no longer represent a viable economic alternative.

4. The prime incentive for the HWR from a utility standpoint is a reduced sensitivity of power costs to the price of U_3O_8 . This reduced sensitivity results from the lower uranium requirements of the HWR, about 65% of that of the LWR on the equivalent once-through slightly-enriched uranium fuel cycle. These lower uranium requirements would help to partially mitigate the effect of a shortfall in uranium supply or production and reduce the effect of uranium price increases which might be expected to accompany such a shortfall. However, uranium requirements remain substantial. If such a shortfall in uranium supply were really anticipated, the development of breeder reactors would probably be the preferred utility alternative.
5. In view of the highly satisfactory operation of Candu plants on the natural uranium fuel cycle, the major technical problem areas are considered to be related to the changes incorporated into the conceptual design to improve uranium utilization and power cost economics. In particular, development and demonstration of slightly enriched fueling will be required to demonstrate acceptable fuel performance at the higher discharge burnups obtained and to fully ensure that the use of slightly enriched fueling does not reduce thermal margins or in some other fashion compromise plant safety and licensability. The development of pressure tubes capable of withstanding the elevated temperatures and pressures of the conceptual design is a second potential technical problem area which will necessitate significant R&D.

6. No technical problem has been identified which should preclude licensability of the conceptual HWR design in the U.S. However, detailed U.S. NRC review will be required to confirm that a particular plant design is, in fact, licensable. As has been the case for LWRs, licensing review may necessitate design modifications which could potentially have a serious impact on plant capital cost and performance. These uncertainties associated with plant licensability must be resolved through a licensing review of a demonstration plant before the HWR can be considered a viable commercial alternative in the U.S.
7. There are a number of possible introduction scenarios for the introduction of the HWR, depending upon the availability of Canadian technology and upon whether larger or smaller financial risks are thought to be warranted, based upon the incentives for HWR deployment. Assuming a joint U.S.-Canadian program and incentives sufficiently high to warrant significant financial risks, the first commercial-type HWR unit could begin operation as early as 1997. However, scenarios which attempt to minimize risk (i.e., which require the successful operation of a demonstration plant before commitment is made to construct commercial units) would defer the operation of the first commercial unit to beyond the year 2000. In the absence of a joint U.S.-Canadian program, HWR deployment in the U.S. would be delayed by another five to ten years or longer.

References

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