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

By
N. L. Shapiro
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September 1979

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Combustion Engineering, Inc.
Windsor, Connecticut

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Volume 1

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WATER REACTOR FOR U.S. SITING

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1. Executive Summary

In his energy statement of April 7, 1977, President Carter enunciated a nuclear policy emphasizing nonproliferation of nuclear weapons. Subsequently, the Non-Proliferation Alternative Systems Assessment Program (NASAP) was initiated in the U.S. to identify reactor and nuclear fuel cycle options which pose inherently low risk of nuclear weapons proliferation while retaining the major benefits of nuclear energy. Although the light water reactor operating on the slightly enriched uranium once-through cycle is viewed to impose an inherently low risk of nuclear weapons proliferation, reactors of this type require relatively large quantities of uranium fuel for continued operation. One goal of the NASAP program is consequently to identify economic nuclear alternatives which produce more energy per pound of U_3O_8 and per unit of separative work expended.

In this context, the pressurized heavy water reactor can make a significant contribution since when operated on the slightly enriched uranium once-through fuel cycle the HWR can surpass all other reactors in energy production per unit of U_3O_8 and separative work. Compared with the LWR in the same fueling mode, the HWR can produce about 70% more energy per unit of U_3O_8 consumed. In the longer term, it may be necessary to improve the efficiency with which nuclear fuel is utilized through reprocessing and fissile recycle. In this situation, the HWR could play an important role as a high efficiency converter when operated on the denatured uranium-thorium cycle.

The study of the heavy water reactor (HWR) presented in this report is primarily intended to develop a reliable data base for use by DOE in comparative assessments and evaluations to establish the extent to which this particular reactor concept, coupled with various fuel cycles, could contribute to the multiple goals of

fuel efficiency, nonproliferation, and economic power generation. The main elements of this study consisted of:

- ° The conceptual design of a large HWR for U.S. siting optimized to meet the objectives of the NASAP program.
- ° Characterization of the uranium utilization and power cost economic performance of the conceptual HWR design operating on various fuel cycles, to serve as a data base for NASAP assessments.
- ° An identification of safety-related and environmental problems which might necessitate significant amounts of RD&D.
- ° The identification of RD&D requirements and of factors which might influence the commercial acceptance of the HWR in the U.S.

After considering the various types of HWR plants in the world (i.e. Eugen, Cirene, Atucha, CANDU, SGHWR) the Candu-PHWR was selected as the point of departure in developing the conceptual HWR design, since reactors of this type are already widely deployed in commercial operation in Canada as well as in other parts of the world. Although the operation of Candu PHW reactors on the natural uranium fuel cycle has been excellent, preliminary screening analyses performed under NASAP have indicated that heavy water reactors of existing design would fall somewhat short of the goals and objectives of NASAP, and were unlikely to compete successfully with light water reactors currently in operation in the U.S. In particular, it was recognized that in order to meet the objectives of NASAP alternate reactor concepts would have to achieve significantly better uranium utilization than that of the LWR. The resource utilization achieved with natural uranium fueling is only moderately better than that of the corresponding

LWR once-through cycle, about 20% lower, and this reduction was viewed as insufficient to warrant development and deployment of a new reactor concept in the U.S. or to significantly alter the time at which reprocessing for the plutonium-fueled breeder reactor might be required. NASAP objectives also recognized that an alternate reactor concept must compete economically with light water reactors if it is to be widely accepted and deployed in the U.S. Previous studies have indicated that the natural uranium fueled PHWR, with modifications required to meet U.S. licensing criteria and construction practices, would have a significantly higher capital cost than that of the LWR and consequently was unlikely to significantly penetrate into the U.S. market. The conceptual HWR developed in this study, though based on the Candu design, features a number of design changes which serve either to improve uranium utilization or to reduce plant capital cost. In particular, the conceptual HWR differs from the Candu design primarily because of an increase in station size to 1260 MW(e), the use of slightly enriched uranium fuel, modifications to increase station efficiency, and because of design changes made to facilitate conformance to U.S. design or licensing practices. Where possible, the design and features of the Candu-PHWR have been retained in order to take advantage of the extensive development and proven performance of the Canadian heavy water reactor design.

In developing the conceptual HWR design, net station output was increased to 1260 MW(e) to take advantage of economies of scale. Further reductions in plant capital cost were obtained by retaining the two-loop configuration currently employed in operating Candu plants. As compared to the alternative of employing three smaller loops, the two-loop configuration allows further economies of scale to be obtained by employing fewer larger components, and also allows a more compact component arrangement which minimizes containment size, thereby reducing plant capital cost.

Slightly-enriched uranium fueling was selected both because of improved uranium utilization and fuel cycle cost economics. An initial enrichment of 1.2 w/o was found to be near optimal from both a uranium utilization and economic standpoint. With an initial enrichment of 1.2 w/o, uranium requirements are reduced by about 29%, as compared to natural uranium fueling. There is little incentive for increasing the initial enrichment beyond 1.2 wt% since only small further reductions in uranium requirements result.

The use of slightly enriched fuel also affords an opportunity for incorporating a number of design changes to reduce plant capital cost. These design changes result in some reduction in achievable discharge burnup which are insignificant at the burnups achieved with slightly enriched fueling, but are unacceptable for natural uranium where the burnup is already quite low. One such example is the higher primary operating pressure which has been incorporated into the conceptual HWR design in order to increase the net electrical efficiency, thereby reducing plant capital cost. In order to accommodate this higher operating pressure, it is necessary to increase the thickness of the pressure tubes, and as a result the achievable burnup is decreased. For the natural uranium fueled system, the effect of increasing the pressure tube thickness is to reduce the achievable burnup of 7500 MWd/t by about 1400 MWd/t, or by nearly 20%, and, as a result of this reduction in discharge burnup, uranium requirements increase by about 10%. At an initial enrichment of 1.2 wt% and a burnup of 20,000 MWd/t, the burnup penalty is about 1000 MWd/t, or about 5% of the total cycle length. At this enrichment, the small burnup penalty is more than compensated by the increase in net efficiency, and consequently a slight improvement in uranium utilization results.

In developing the conceptual HWR, U.S.-type balance-of-plant designs were employed, rather than the designs currently in use on Candu

reactors, in the belief that U.S. architect engineers would prefer the more familiar PWR design concepts. Questions of the acceptability of the Candu design under U.S. licensing criteria frequently center on these differences in balance-of-plant design; the use of a U.S.-type design thus simplifies the licensing evaluation, allowing attention to be focused on the more fundamental characteristics of the HWR plant.

The lifetime (30-year) U_3O_8 and separative work requirements (SWU), as well as the annual fissile plutonium production, are summarized in Table 1-1 for the various fuel cycles evaluated in the conceptual HWR plant. Also shown for comparison are values for the standard PWR; the standard PWR is typical of light water reactors currently in operation, and does not include improvements which are currently under development by DOE. 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 5000 ST for the natural uranium HWR and 6000 ST for the PWR; thus, the U_3O_8 requirements for the natural uranium fueled HWR are about 40% greater than that of slightly enriched fueling, while U_3O_8 requirements for the LWR are approximately 70% greater. Separative work requirements for the slightly enriched HWR are about one-fourth as large as those of the LWR. 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 less than half the discharge of the natural uranium fueled system.

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 once-through cycle), or by the use of the denatured U-235/thorium cycle (with uranium recycle and plutonium storage) where U_3O_8 requirements are about 70% of the slightly enriched once-through fuel cycle. The uranium requirements

for the U-235/thorium cycle can be reduced slightly (4% additional reduction) through the use of plutonium recovered from the spent fuel. The use of denatured thorium fueling increases separative work requirements by about a factor of 2.5 as compared to the slightly enriched once-through cycle; separative work requirements, however, are still well below those of the light water reactor. The annual discharge fissile plutonium in spent fuel is also decreased significantly, to about 10% of that of the slightly enriched once-through cycle.

Table 1-2 provides a similar summary of the busbar power costs for each of the fuel cycles. The power costs are provided for both near and long-term economic assumptions, which are characterized primarily by a U_3O_8 price of 40\$/lb or 100\$/lb, respectively. The cost of reprocessing and of fuel refabrication are also somewhat lower under the long-term economic assumptions. These economic assumptions, as well as the sensitivity of power cost to various economic parameters, are discussed in detail in the body of the text (See Section 5.4). Busbar power costs for the HWR are based upon a total capital cost (including interest during construction and owner's cost) of 768\$/kWe, exclusive of heavy water inventory (January 1978 dollars). This evaluated HWR capital cost (excluding D_2O), is about 8% higher than that of the LWR or about 34% higher when D_2O is included. The HWR power costs are also evaluated at a slightly higher capacity factor than that of the LWR, 77.5 vs 75%, because of the increased availability attributable to on-line refueling in the HWR.*

*Although the differences in LWR and HWR design would seem to justify only the 2.5 percentage point difference in capacity factor cited above, the lifetime average capacity factor of Ontario Hydro Candu-PHWR's is nearly 20 percentage points greater than the world average of PWR plants (Ref. 1-1). This difference in capacity factor is not fully understood, and is believed to be due, at least in part, to factors other than inherent differences in HWR and LWR reactor plant design. However, if this larger differential in plant capacity factor could be realized for a U.S. sited HWR, the difference between HWR and LWR power costs (Table 1-2) would be reduced by about 3 mills/kw-hr, and the once-through slightly enriched fuel mode in the conceptual HWR would be economically competitive with the LWR for uranium prices at or below current market prices.

As Table 1-2 indicates, 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; HWR power costs are lower than those of the LWR if the price of uranium is greater than about 55\$/lb. Once-through fueling is seen to be more economic than any of the plutonium recycle or denatured U-235/thorium cycles considered in this study, for both the near and long-term economic assumptions; the price of uranium must rise to about 200\$/lb before these fuel cycles become economically competitive with once-through fueling. A similar conclusion can be reached on the U-233/thorium cycle, although the economic performance of this cycle is less easy to ascertain because it is dependent upon the price for which U-233 can be purchased.

A qualitative assessment of safety and licensability was also performed as part of the study to identify safety-related and environmental issues which would require significant licensing review, or which represent potential problem areas impacting research and development needs. This assessment addressed both the safety and licensability of the basic HWR concept and the implications of the design changes which were incorporated into the conceptual HWR design to improve either uranium utilization or economic performance. The safety and licensing evaluation concluded that there was no fundamental reason why the HWR should not be licensable in the U.S. In general, Candu plants are designed to acceptance criteria which are as, or more, demanding than comparable U.S. criteria. U.S. licensing criteria, however, place additional limitations on fuel damage and also impose a more limiting acceptance criterion on clad temperature during LOCA; detailed quantitative assessments will be required to determine whether the HWR meets these U.S. requirements or whether slight design changes are required (e.g., a small increase in the number of fuel channels) to reduce power density.

The prime inherent characteristics of the HWR reactor which contribute to increased plant safety are considered to be the use of the pressure-tube configuration, the on-line refueling feature, and the separate low pressure moderator system. The use of the pressure tube configuration in the HWR, coupled with the injection of emergency core cooling water into both inlet and outlet headers, facilitates the supply of coolant 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, and also facilitates incorporating a second and independent shutdown system. The location of fuel within separate fuel channels, coupled with the ability to refuel on line, 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 potentially available for release 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 pressure tubes as the primary system boundary, the positive void coefficient of reactivity, and the increased tritium production which is inherent in heavy water systems. Although the review of these inherent attributes indicates that they should not pose a fundamental problem to licensing of the HWR in the U.S., it is nevertheless anticipated that these items would become the subject of extensive licensing review and discussion.

Although detailed design evaluations will be necessary before each of the modifications incorporated into the conceptual HWR can be judged licensable, the potential safety and licensing implications of these design changes were reviewed with the conclusions that the core transient characteristics are not significantly altered, and that the consequences of various postulated accidents should remain acceptable. The design change which engenders the most significant safety and licensing questions is the increase in primary system

pressure and temperature. R&D will be required in order to develop additional information on pressure tube strength and corrosion resistance at the elevated temperatures of the conceptual design, and a more detailed assessment of the implications of this design change on the loss-of-coolant accident will be required.

The conceptual HWR is not likely to be introduced in the U.S. today. While the conceptual HWR meets the goals and objectives of NASAP, as viewed from a U.S. utility viewpoint there is little difference between the economics of present light water reactor plants and the proposed reactor. The risks inherent in introducing a new reactor type (including one that is commercially successful elsewhere) to their utilities systems suggests that commercial introduction would have to be preceded by a demonstration project.

TABLE 1-1
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	1.18
HWR - Natural U Fueling (Once-Through)	5000	1.41	0	0	350	2.35
SEU* HWR - 1.2 w/o (Once-Through)	3550	1.00	927	1.0	149	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	0.15
37,000 MWD/MT	2500	0.71	2360	2.55	12.0	0.08
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	-	-

*SEU - Slightly Enriched Uranium

TABLE 1-2
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

References

- 1-1 "Comparison of Ontario Hydro with World Power Reactors, 1978",
Ontario Hydro.

2. Introduction

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2. Introduction

2.1 Role of HWR in NASAP

In his energy statement of April 7, 1977, President Carter enunciated a nuclear policy emphasizing non-proliferation of nuclear weapons. Largely through his initiatives, the International Fuel Cycle Evaluation (INFCE) was established to identify nuclear fuel cycle options for worldwide application which pose inherently low risks of nuclear weapons proliferation while retaining the major benefits of nuclear energy. At the same time, the Non-proliferation Alternative Systems Assessment Program (NASAP) was initiated in the U.S. to provide input to INFCE and to evaluate various reactor and fuel cycle options which might potentially be implemented in the U.S.

The principal proliferation concern in civilian nuclear power cycles is the possible diversion of fissile material to the fabrication of nuclear weapons. The prospect of the successful, or rapid, diversion of fissile material is enhanced if the fissile material is available in a form which can be directly used in weapons, or in a form where weapons-usable material could be obtained by simple chemical processing. The slightly-enriched uranium fuel utilized in light or heavy water reactors is unsuitable for use in nuclear weapons because the fissile content is too low; obtaining weapons-usable material would require isotopic concentration, a very difficult technology currently performed in only a few facilities in the world. In contrast, the plutonium contained in fresh plutonium-uranium fuel can be extracted by relatively simple chemical processes.

Although the light water reactor operating on the slightly-enriched uranium once-through cycle is viewed to impose an inherently low risk of nuclear weapons proliferation, reactors of this type require relatively large quantities of uranium fuel for continued operation. Although the size of the economically recoverable uranium resource base is uncertain, there is, nevertheless, an incentive for considering alternatives which can utilize that resource base more efficiently, so as to defer into the future the possible need for utilizing plutonium. One goal of the NASAP program is, consequently, to identify economic nuclear alternatives to LWRs which produce more energy per pound of U_3O_8 and per unit of separative work expended.

In this context, the pressurized heavy water reactor can have a significant role in NASAP, since when operated on the slightly-enriched uranium once-through fuel cycle, the HWR can surpass all other reactors in energy production per unit of U_3O_8 and separative work. Compared with the LWR in the same fueling mode, the HWR can produce about 70% more energy per unit of U_3O_8 consumed. In the longer term, it may be necessary to improve the efficiency with which nuclear fuel is utilized through reprocessing and fissile recycle. In this situation, the HWR could play an important role as a high efficiency converter when operated on the denatured uranium-thorium cycle. A nuclear generating capacity more than twice as large as would be possible with standard LWRs could be supported by the uranium resource base with denatured U-235/Th fueling in the HWR, and the HWR could also serve as a highly efficient utilizer of U-233. Thus, the HWR is not only a highly efficient system when operated on the slightly enriched uranium once-through cycle, but can also accommodate the denatured thorium fuel cycle when it has been developed and accepted.

Study Objectives

The study of the heavy water reactor (HWR) presented in this report is primarily intended to develop a reliable data base for use by DOE in comparative assessments and evaluations to establish the extent to which this particular reactor concept, coupled with various fuel cycles, could contribute to the multiple goals of fuel efficiency, non-proliferation, and economic power generation. Essential to achieving this goal has been the development of a conceptual HWR design for U.S. siting. The Candu-PHWR was selected as the point of departure in developing this conceptual design, since this concept has been extensively developed by Canada and is already commercially in use in Canada as well as other parts of the world. Operating experience with these plants has proved excellent, so that a substantial technological base exists for the further development and rapid deployment of this family of reactors.

Although the operation of Candu-PHW reactors on the natural uranium fuel cycle has been excellent, preliminary screening analyses performed under NASAP have indicated that heavy water reactors of existing design would fall somewhat short of the goals and objectives of NASAP, and were unlikely to compete successfully with light water reactors currently in operation in the U.S. In particular, it was recognized that in order to meet the objectives of NASAP alternate reactor concepts would have to achieve significantly better uranium utilization than that of LWRs. Although the use of natural uranium fuel has been attractive to nations who wish to develop a nuclear industry without dependence on enrichment capacity, the resource utilization achieved with natural uranium fueling is only moderately better than that of the corresponding LWR once-through fuel cycle. U_3O_8 requirements for the HWR operating with natural uranium fueling are about 20% lower than those of the LWR - about 5,000 ST U_3O_8 for the HWR compared to 6,000 ST U_3O_8 for the LWR are required to operate a 1,000 MW plant for an assumed 30-year lifetime at a 75% capacity factor. Although the 20% reduction in uranium requirements achieved with natural uranium fueling in the HWR is

certainly significant, this reduction in uranium requirements was viewed as insufficient to warrant the development and deployment of a new reactor concept in the U.S. or to significantly alter the time at which reprocessing for plutonium recycle might be required. NASAP objectives also recognized that an alternate reactor concept must compete economically with light water reactors if it is to be widely accepted and deployed in the U.S. Previous studies have indicated that the natural uranium fueled PHWR, with modifications required to meet U.S. licensing criteria and construction practices* (Reference 2-1), would have a significantly higher capital cost than that of the LWR (Reference 2-2), and consequently was unlikely to significantly penetrate into the U.S. market (Reference 2-3).

Consequently, the basic objective of this study has been to develop a conceptual HWR design for use in NASAP evaluations in which careful attention was given to alternatives which can further improve uranium resource utilization or which would reduce the capital cost of the HWR plant. In making design modifications to the basic Candu to accomplish this, it was recognized that significant departures from present Candu designs might open questions of design feasibility, new licensing issues, and necessitate research and development which could delay introduction of the system. It was also recognized that a number of design modifications would have to be made to the Candu design for siting in the U.S. in order to conform to U.S. licensing criteria and construction practices, and that further modifications would be required to increase the capacity of existing Candu designs to the size (approximately 1000 MWe) specified for this study. Also, care was taken to evaluate

*A conservative approach to applying U.S. licensing criteria was employed in this study to obtain an upper limit regarding the cost involved in adopting the PHWR to known U.S. licensing requirements. For this reason, the study did not reflect, in all areas, the compensating cost factors associated with areas where Canadian licensing requirements are more conservative or where different design solutions were followed to meet the same licensing intent.

the feasibility of these modifications and to identify their potential impact on safety and licensability, RD&D requirements, and schedules for commercial deployment. Only those design modifications which resulted in significant improvements in uranium utilization and/or power cost economics were incorporated into the conceptual design.

In order to enhance the prospects for early deployment and to minimize development requirements, the conceptual heavy water reactor design developed for NASAP is based upon the pressurized heavy water Canadian deuterium uranium reactor (Candu-PHWR) and, in particular, on the standard Candu 600 reactor design (References 2-4 and 2-5), currently under construction at the Gentilly 2 and Point Lepreau stations in Quebec and New Brunswick, respectively. The conceptual design differs from the Candu 600 primarily because of an increase in station size to 1260 MW(e), the use of slightly-enriched uranium fuel, modifications to increase station efficiency, and because of design changes made to facilitate conformance to U.S. design or licensing practices.

The subsequent sections of this chapter review briefly the heavy water reactor concept. The results of this evaluation are summarized in Section 3, which provides a description of the conceptual HWR and discusses its fuel utilization and economic performance, safety, licensing and environmental impacts, and commercial introduction considerations. The results of this study are described in detail in Section 4 (Optimization of HWR Design), Section 5 (HWR Reference Design and Performance) and Section 6 (Commercial Introduction of the HWR).

2.3 HWR Concept

2.3.1 Review of Alternate HWR Concepts

The development of heavy water moderated nuclear steam supply systems has been undertaken by a number of nations, and has resulted in a number of alternate HWR concepts. These alternate HWR concepts all employ heavy water as the neutron moderator, but differ in the coolant employed to transport energy from the nuclear fuel. HWRs employing pressurized heavy water, boiling light water, and organic coolants have been designed and successfully operated. HWRs also differ in the types of pressure boundaries employed, with some concepts employing pressure tubes to confine the primary coolant while others employ a pressure vessel similar to that used in LWRs.

The HWR concept which has been most widely deployed is the Candu-PHWR (Canadian, deuterium oxide moderated, natural uranium, pressurized heavy water reactor). The Candu-PHWR utilizes heavy water for both moderator and coolant. The pressurized heavy water coolant flows through horizontally-arranged pressure tubes which are located within a calandria (tank) which contains the heavy water moderator. The moderator is operated at low temperatures and at near atmospheric pressure. Operating Candu plants employ natural uranium fuel and an on-power, or on-line, fueling mode in which individual fuel channels are refueled without interrupting reactor operation. Candu power plants which are in operation, under construction, or planned are listed in Table 2.3-1. The viability of the Candu-PHWR concept was first demonstrated by the design and construction of the 22 MW(e) NPD (Nuclear Power Demonstration) station, which came into operation in 1962. A second larger unit, the 200 MW(e) Douglas Point generating station, designed as a prototype for commercial Candu-PHW reactors, began operation in 1967. The four 514 MW(e) reactor units of the Pickering nuclear generating station, commissioned between 1971 and 1974, represent the first commercial application of Candu-PHW reactors.

The 745 MW(e) units at the Bruce station are the largest capacity Candu units presently in operation. As Table 2.3-1 illustrates, a substantial number of additional units are either under construction or planned in Canada, and in addition, Candu-PHW reactors are operating in Pakistan and India and under construction in Argentina and Korea.

A second heavy water reactor concept is represented by the 320 MW(e) Atucha plant in Argentina (Reference 2-6). Like the Candu-PHWR, the Atucha plant utilizes heavy water both as moderator and coolant, is fueled with natural uranium, and utilizes on-power refueling. However, like pressurized light water reactors, the Atucha plant utilizes a pressure vessel to contain the pressurized coolant, rather than the pressure tubes employed in the Candu design. The 52 MW(e) MZFR reactor, located in Germany and placed into operation in 1962, served as a prototype for the Atucha design. Since the Atucha and Candu-PHW reactor types share the features of natural uranium fueling, pressurized heavy water coolant, and on-line refueling, the resource utilization performance of the two reactor types is roughly equivalent. However, unlike the pressure tube HWR, such as Candu, which can be scaled up to plant sizes comparable to the 1300 MW(e) LWR, HWR's like Atucha are limited to approximately 600 to 900 MW(e) because of size limitations on pressure vessels.*

A third major heavy water reactor concept is characterized by the use of boiling light water as the coolant, rather than the pressurized heavy water of the Candu-PHW or Atucha type reactors. The main incentive for utilizing light water instead of heavy water as a reactor coolant is to reduce the inventory of heavy water in the

*More recent information indicates that it appears possible to construct reactors of this type with capacities of approximately 1100 - 1200 Mwe using present state-of-the-art technology (Reference 2-7).

plant, and hence to reduce plant capital cost. However, since the light water coolant will absorb more neutrons than would heavy water, the resource utilization of this type of reactor is somewhat less than in pressurized heavy water designs; the resource utilization performance is typically somewhere between that of the PHW and the LWR reactor types. The first power reactor of this type was the 92 MW(e) Winfrith SGHWR (Steam Generating Heavy Water Reactor) which was placed into operation in 1968 by the United Kingdom Atomic Energy Authority. The Winfrith SGHWR employs slightly-enriched uranium fuel. An experimental reactor of similar design, the Candu-BLW (Boiling Light Water) was placed into operation in 1971 at Gentilly, Quebec. This reactor has a capacity of 250 MW(e) and is fueled with natural uranium. Reactors of this general type are also under construction in Japan (the 200 MW(e) Fugen Station, now nearing completion) and in Italy (the Cirene Station, a 40 MW(e) plant still in the early stages of construction). Although under development in Canada, Britain, Japan and Italy, reactors of this type are not currently commercially available, nor have any of these countries announced plans for the commercial deployment of this concept.

A fourth heavy water reactor concept is the heavy water moderated organic cooled reactor, represented by the Candu-OCR (Organic Cooled Reactor). The primary incentives for organic cooling are the reduction in heavy water inventory and the higher temperatures and hence channel powers and operating efficiencies possible with organic coolant. The major concern typically raised with this concept is the question of the performance of the organic coolant under irradiation. The principles of the organic cooled reactor were tested in the 60 MW(t) reactor experiment, WR-1, operated by AECL at its Whiteshell Nuclear Research Establishment in Manitoba, Canada. Although the operation of the WR-1 reactor has been successful, the development of the Candu-OCR concept beyond the test stage has not occurred.

Although any of the alternate heavy water reactor concepts described above might well have been chosen for consideration in this evaluation, the conceptual design discussed in this report is based upon pressurized heavy water moderated and cooled reactors of the Candu-PHWR type. The reasons for selecting this concept as the point of departure for this evaluation lie in the proven performance of this reactor type and its commercial status, and the superior fissile material utilization possible with heavy water coolant. As mentioned above, the Candu-PHW is the only type of heavy water reactor which has been extensively deployed: approximately 4,100 MW(e) capacity is currently in operation, while an additional 11,700 MW(e) of generating capacity is currently under construction or planned. As indicated by Table 2.3-2, the performance of operating units (Pickering and Bruce Stations) has generally been excellent. Due to the use of heavy water both as coolant and moderator, reactors of the Candu-PHW type exhibit improved resource utilization compared to alternate concepts which utilize light water or organic coolants. The resource utilization of the Candu-PHW and Atucha type reactors are comparable, but Candu reactors are more widely deployed and are not limited in generating capacity, as are Atucha-type reactors, by pressure vessel size.

These considerations--the commercial status and number of units deployed, the demonstrated reliable performance, and the efficient fuel utilization inherent in the concept--led to the selection of the Candu-PHWR as the point of departure for the conceptual HWR design described in this report. Although one objective of this evaluation has been to minimize the changes to existing Candu-PHWR designs so as to enhance the prospects for early deployment and to minimize R&D requirements, it was necessary to incorporate a number of modifications into the conceptual HWR design in order to improve uranium utilization, enhance the prospects for economic competition with the LWR, and to conform to U.S. licensing criteria and construction practices. The design of the Candu-PHWR plant is reviewed briefly

in the next section along with the design approach used to evolve the conceptual HWR design for U.S. siting. This conceptual HWR design is summarized in Section 3.1, while Section 3.2 provides a summary description of modifications relative to the standard Candu-PHWR plant.

2.3.2 Summary of the Candu-PHWR Concept

The purpose of this section is to summarize briefly the design features of the standardized Candu-PHWR design which has been the point of departure for the development of the conceptual HWR design for U.S. siting described in this report. The Candu pressurized heavy water reactor design has evolved from the 208 MW(e) Douglas Point reactor commissioned in 1967, through the 514 MW(e) Pickering units commissioned between 1971 and 1973 to the most recently operating Candu reactors, the 745 MW(e) Bruce stations commissioned between 1976 and 1979 and the 800 MW(e) units planned for Darlington. However, the point of departure for this study has been the standardized Candu-600 plant because information on this plant was readily available having been used as the basis for previous U.S. HWR capital cost and licensing studies. The standardized Candu-600 plant, which achieves a net power output of 638 MW(e), is represented by the Gentilly 2 and Point Lepreau stations, scheduled for operation in 1980 and 1981, respectively. The standardized Candu-600 design (References 2-4 and 2-5) employs somewhat larger components and achieves somewhat improved performance as compared to its predecessors. Design and operating data for the standardized Candu-600 plant are summarized and compared with previous designs in Table 2.3-3.

The principal design features of the Candu-PHWR reactor concept are illustrated schematically in the simplified diagram Figure 2.3-1. The reactor consists of a large horizontally-orientated cylindrical tank, called a calandria, which contains the heavy water moderator. The moderator is kept at a relatively low temperature (71°C) by the moderator cooling system shown at the bottom of Figure 2.3-1, and is operated at near atmospheric pressures. The moderator is heated

primarily by energy deposited from the neutrons and gamma rays emitted by the fuel; approximately 6% of the total energy produced in the fuel through fission is deposited in the moderator. Since the moderator temperature is low, heat extracted from the moderator circuit is not employed to generate electricity.

Horizontally-orientated pressure tubes, which contain the natural uranium fuel and through which the pressurized high-temperature, heavy-water coolant flows, penetrate the calandria. The standard Candu-600 design contains 380 such fuel channels, divided into two coolant loops (Figure 2.3-1 illustrates one of these two coolant loops). The coolant is pumped through the fuel channels, removing heat from the fuel, and then to the steam generators where heat is given up to produce steam which is fed to the turbine. The steam generators and coolant pumps are located at each end of the reactor so that flow is in one direction through one-half the fuel channels and in the opposite direction through the other half. The standard Candu-600 design has two coolant loops of the type shown in Figure 2.3-1, so that 4 steam generators and coolant pumps are employed, with 2 steam generators and 2 coolant pumps associated with each loop.

Inlet and outlet headers are provided at each end of the reactor for each reactor coolant loop. These headers are large diameter pipes to which smaller pipes providing coolant to each fuel channel are connected. The fuel channel consists of a zirconium-2.5% niobium alloy pressure tube expanded at each end into a stainless steel end fitting (rolled joint). These pressure tubes have a thickness of about 4.34 mm., and serve as a pressure barrier between the high pressure (11 MPa(a)), high temperature (267°C) heavy water coolant and the low-pressure low-temperature moderator; consequently, the pressure tubes serve the same purpose as the pressure vessel utilized in light water reactors. Each pressure tube passes through a thin-walled (1.4 mm) calandria tube. The gap between the pressure tube and calandria tube is gas filled, and serves to provide thermal insulation to minimize the amount of heat transfer to the moderator

from the high-temperature fuel channels. These fuel channels are widely spaced (approximately 28.6 cm. center-to-center spacing) so as to be surrounded with sufficient quantities of heavy water to adequately moderate the neutron spectrum, and as a result the physical dimensions of the Candu-PHWR reactor are large compared to LWRs. The wide spacing in the fuel channels also facilitates access by the refueling machine, allowing the fuel channels to be refueled while the reactor is in operation.

Fuel for the reactor is in the form of cylindrical bundles, approximately 0.5 meters long. Each bundle consists of 37 fuel elements which are comprised of approximately 29 fuel pellets stacked end-to-end and sealed in a Zircaloy-4 sheath. The fuel elements are attached mechanically at their ends to form a cylinder approximately 100 mm in diameter, with a small space being maintained between each element by spacers attached to the element cladding. Twelve of these fuel bundles are stacked end-to-end in each fuel channel.

2.3.3 Design Approach

As mentioned previously, the conceptual HWR design described in this report is based upon the Candu-PHWR. Although the advantages of minimizing the number of changes to the Candu-PHWR in evolving the conceptual HWR design for U.S. siting were recognized, it was also recognized that a number of the features of the Candu plant differ significantly from those of light water reactors for which U.S. licensing and construction experience is centered. Thus, while emulating the design of the Candu plant to the greatest extent possible would allow the most direct application of the highly successful operating experience of operating Candu units and would minimize R&D requirements, this approach would also introduce a number of design features unreviewed by the U.S. licensing authorities and alien to the U.S. construction industry.

In order to avoid introducing a large number of new design features which would require licensing review or which might require extensive design development, a design approach was evolved which employed, to the greatest extent possible, engineered design features developed for light water reactors. This design approach recognizes that while there are many unique features of the Candu plant (for example, the reactor itself), there are many other aspects of plant design which can be similar or identical to light water reactor systems (for example, containment design, design of structures, emergency core cooling system, etc.). This latter class of engineered design features of the Candu plant often differs somewhat from similar systems employed in light water reactors because of differing Canadian licensing criteria and construction practices. While it should be possible, with some development and demonstration, to adopt the Canadian construction practices and to show in most instances that the Canadian design meets U.S. licensing requirements, clearly considerable design development and licensing review would be required, thus increasing the uncertainty that the conceptual design will prove acceptable after detailed U.S. licensing review. In developing the conceptual design, an attempt has been made to minimize the uncertainties associated with licensing acceptance by employing engineered features consistent with, or identical to, designs employed in U.S. light water reactors. Thus, as discussed more fully in Section 3.2, the conceptual design employs a U.S.-type containment design, structures designed to U.S. seismic qualifications, PWR-type equipment supports and constraints, and a high-pressure emergency core cooling system similar to that employed in PWRs.

A number of changes have also been made to the basic Candu-PHWR design to improve uranium utilization or economic performance, the most notable of which are the use of slightly-enriched uranium fueling and an increase in primary and secondary system pressures to increase net electrical generating capacity. Since a detailed safety analysis of the conceptual HWR incorporating these design modifications was outside the scope of this study, an alternate approach was employed to qualitatively assess the impact of design

changes on safety and licensability. In particular, the effect of these design changes on such reactor core characteristics as power distributions and the coefficients of reactivity, which can have significant impact on the safety analysis, were calculated and compared with similar parameters for the standard Candu plant to ascertain the likely consequences of these design changes.

The basic design approach--that of basing the conceptual HWR design upon the demonstrated Candu-PHWR concept, of employing PWR-type engineered features to minimize uncertainties associated with licensing acceptance, and of evaluating changes made to improve uranium utilization or economic performance on a case-by-case basis--serves to minimize the uncertainties in the resulting data base. Nevertheless, it should be recognized that many important areas could not be fully explored in this limited study. In particular, it is felt that significant uncertainties remain in the material and performance characteristics of the pressure tubes at the elevated pressures and temperatures of the conceptual design, in the fuel management and power margin (power peaking factor and margin to dry-out) characteristics of the slightly enriched fuel cycle, in the transient performance of the HWR, and in possible ramifications to plant design and capital cost of applying U.S. licensing criteria. Thus, although this study is now completed, there remains considerable work which would be required to fully resolve these technical issues. While many of these items would require extensive R&D, including the construction of test and demonstration facilities, for their resolution, uncertainties in many areas could be eliminated or reduced by more modest R&D programs. Technical issues which require further R&D are discussed in the body of this report, where an attempt has been made to identify those items which appear to be resolvable with modest additional effort, and those more fundamental problem areas which might require more extensive RD&D.

TABLE 2.3-1
CANDU PHWR PLANTS IN OPERATION, UNDER CONSTRUCTION OR PLANNED

<u>Station</u>	<u>Country</u>	<u>Number of Units</u>	<u>Unit Power MWe</u>	<u>Operation Date</u>
NPD	Canada	1	22	1962
Douglas Point	Canada	1	208	1967
KANUPP	Pakistan	1	125	1971
RAPP 1	India	1	203	1972
RAPP 2	India	1	203	?
Pickering A	Canada	4	514	1971-1973
Bruce	Canada	4	745	1976-1979
Gentilly 2	Canada	1	600	1980
Cordoba	Argentina	1	600	1980
Lepreau	Canada	1	600	1981
Wolsung	Korea	1	600	1982
Pickering B	Canada	4	500	1981-1983
Bruce B	Canada	4	750	1983-1986
Darlington	Canada	4	800	1985-1988

TABLE 2.3-2
CAPACITY FACTORS FOR CANDU PLANTS

Unit	In-service Date	1972 ¹⁾	1973	Capacity Factor, %		1976	1977	1978	Lifetime Average
		1974	1975						
Pickering									
Unit 1	7/71	72.3	92.5	72.0	80.2	92.8	85.6	95.1	84.5
Unit 2	12/71	82.2	69.0	88.4	86.0	93.2	90.9	84.3	84.9
Unit 3	6/72	91.3	85.1	42.7 ²⁾	57.5 ²⁾	93.9	95.6	82.2	76.6
Unit 4	6/73	--	90.1	93.9	23.8 ²⁾	68.4 ²⁾	90.8	89.6	75.0
Bruce									
Unit 1	9/77	--	--	--	--	--	88.1	72.1	76.1
Unit 2	9/77	--	--	--	--	--	80.3	64.5	78.4
Unit 3	2/78	--	--	--	--	--	--	87.6	87.6
Average									80.3

1) Excludes labor dispute

2) Low capacity factor due to pressure tube replacements

TABLE 2.3-3
DESIGN PARAMETERS

	<u>Douglas Point</u>	<u>Pickering Unit 1 & 2</u>	<u>Bruce</u>	<u>Candu-600</u>
Net Power, Mwe	203	4 x 512	4 x 732	638
Fuel Assemblies				
Number per channel	12	12	13	12
Length, mm	495	495	495	495
Maximum diameter, mm	81.7	102.4	102.4	102.4
Number fuel elements/ assembly	19	28	37	37
Pressure Tube				
Pitch, mm	228.6	285.8	285.8	285.8
Thickness (minimum), mm		5.0	4.2	4.2
Material	Zr-2	Zr-2	Zr-2.5% Nb	Zr-2.5% Nb
Number of channels	306	390	480	380
Core Operating Conditions				
Pressure, M Pa(a)	9.8	9.6	9.3	11.0
Maximum Channel Power, Power, Mw	2.7	5.1	5.7	6.5
Exit Steam Quality, %	0	0	3.5	3.0
Inlet temperature, °C	249	249	254	267
Outlet temperature, °C	293	293	299	310
Maximum mass flow/ channel, kg/S	12.6	23.9	23.9	23.9
Discharge burnup, Mwd/kg	7900	7200	8000	7500

STEAM PIPES

HEAVY WATER REACTOR
SIMPLIFIED FLOW DIAGRAM

FIGURE 2.3-1

HEAVY WATER REACTOR SIMPLIFIED FLOW DIAGRAM

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3. Summary

3.1 Summary Description of Conceptual Design

The conceptual Heavy Water Reactor (HWR) Plant described herein is based upon the pressurized heavy water Canadian deuterium uranium reactor (Candu-PHW) and, in particular, upon the standard CANDU 600 reactor currently under construction at the Gentilly II and Point Lepreau stations in Quebec and New Brunswick, respectively. This conceptual design differs from the Canadian Candu 600 design primarily because of the increase in station size to 1260 Mw(e), the use of slightly enriched uranium fuel, modifications to increase station efficiency, and because of design changes made to facilitate conformance to U.S. design or licensing practices; where possible, the design and features of the Candu-PHW reactor have been retained in order to take advantage of the extensive development and proven performance of the Canadian heavy water reactor design.

The layout for this single unit conceptual HWR plant is shown on Figure 3.1-1 and the main station data is given in Table 3.1-1. The arrangement of buildings provides for optimum use of space for the various functions performed in the plant and provides controlled access to vital areas. Mechanical draft evaporative cooling towers are provided to reject heat both from the condenser and from the moderator heat exchanger.

The Reactor Containment Building (RCB) houses the Nuclear Steam Supply System (NSSS). It is a seismic Category I 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 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, 0 inches, 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. A 3/8 inch thick carbon steel liner is applied to the inner surface of the cylinder, 1/2 inch thick liner on the inside of the dome, and a 1/4 inch thick liner on the top of the foundation mat. A 100-ton polar crane, for construction and maintenance operation, is located near the spring line of the containment and is supported from the wall of the containment. The Reactor Service Building (RSB) houses all the auxiliary systems, emergency systems, fuel handling equipment and related systems as well as all the systems associated with heavy water inventory control. It is a seismic Category I building located north-northwest of the Reactor Containment Building with exterior walls, interior walls, floor slabs and roof slab of reinforced concrete. The exterior walls are a minimum of two feet thick. The spent fuel pool is 25x25 ft. and will accommodate one year accumulation of spent fuel plus one core of fuel. The pool is lined with 1/4 inch thick Type 304 austenitic stainless steel plate, which provides a watertight membrane.

The Control, Auxiliaries and Diesel Generator Building (CADB) houses the control and electrical equipment, including support systems required for plant operation. The CADB also houses essential features including the main control room, remote shutdown area, and the emergency diesel-generators. It is a 7 story, reinforced concrete seismic Category I structure, 124 feet wide, 113 feet long, and 91 feet 6 inches high. Two separate and independent cable spreading rooms, one above and one below the control room are provided for power, control, and instrumentation channels. In addition, a network of four separate and independent vertical cable chases and containment penetration rooms are provided.

A Containment Annulus Building (CAB) is provided between the CADB and the containment. This building subtends an arc of 102 degrees, is 16 feet wide, and extends the full height of the CADB and is supported on a common mat with the CADB. Steel structure is provided to support the various cable trays for their routing to the outside of the containment. The structure is designed to seismic Category I requirements. The exterior walls, interior walls, and top slab of the building are reinforced concrete. The exterior walls are a minimum of 2 feet thick. Reinforced concrete walls separate the diesel-generators and support the upper floors.

The Penetration Building houses the main steam and feedwater piping and valves, and portions of the Moderator system. The building is a seismic Category I structure of reinforced concrete. It is on a common foundation mat with the containment and shares a common wall with the containment on one side and has two-foot thick walls on the other three sides and on the roof. The building measures 120 feet long, 32 feet wide from outside of containment wall to edge of turbine building, and is 98 feet high. The main steam and boiler feedwater lines are separated from the moderator system components by reinforced concrete walls. The area containing the main steam piping includes the power operated relief valves and safety relief valves. The relief valves are provided with vent stacks that permit the steam to exit to the environment. Boiler feedwater piping valves are also in this building.

The non-seismic Category I structures contain non-safety related equipment and components that are not required for safety or safe-shutdown. These structures normally are constructed of structural steel framing, metal siding, concrete plank roofing, and founded on concrete spread footings. An example is the turbine building which houses the turbine-generator, condensers and associated equipment, feedwater heaters and pumps, and auxiliary equipment. Seismic Category I components or equipment are not

housed in this building. Other non-seismic Category I buildings that are part of the plant complex, include:

- Administration Building and Warehouse
- Security Building
- Circulating Water and Service Pump Building
- Makeup Water Intake Structure
- Makeup Water Pretreatment Building
- Cooling Tower Switchgear Building
- Fire Water Pump House

The nuclear steam supply system (NSSS) is housed in the containment building and consists of the reactor assembly including fuel, the heat transport system, steam generators, primary system pumps, pressurizer, shutdown and control systems, and instrumentation.

The reactor design is based upon a heavy water moderated, pressurized heavy water cooled, slightly enriched uranium dioxide fueled, horizontal pressure tube reactor. A simplified diagram illustrating the features of this type of reactor is given in Figure 2.3-1.

The reactor is installed in a steel lined concrete vault which is filled with light water for shielding. The main reactor structure is an austenitic stainless steel calandria tank with integral end shields and peripheral internal thermal shield. The calandria tank contains the low temperature (200°F) heavy water moderator, and operates at near atmospheric pressures. The horizontal coolant tubes, in which the fuel resides, are located within calandria tubes which pass through the calandria tank. These calandria tubes are made of thin-walled (1.4 mm minimum wall thickness) Zircaloy and are separated from the coolant (or pressure) tubes by a sealed annulus containing dry nitrogen. The purpose of this annular gas gap is to provide thermal insulation between the high temperature coolant tubes and the low temperature heavy water moderator. The coolant tubes are located within the calandria

tubes and are supported in sliding bearings at the end shields of the calandria. Heat produced in the fuel by fission is removed by pressurized heavy water which flows by the fuel element bundles which reside in these tubes. In heavy water reactors of this type, these coolant tubes serve as the pressure boundary between the high pressure coolant, and the moderator which is operated at near atmospheric pressures. These coolant, or pressure tubes, are made of zirconium-2.5% niobium alloy, approximately 6 meters in length the 10.3 cm in inside diameter. The conceptual HWR plant is designed to operate a somewhat higher primary system pressure than the CANDU 600 (2250 psi vs. 1600 psi for the CANDU 600) in order to allow primary system coolant temperatures and station electrical efficiency to be increased. In order to accommodate this higher system pressure, the thickness of the pressure tubes has been increased from the 4.06 mm thickness of the CANDU 600 design to 5.79 mm; other than this increase in pressure tube thickness, the design features of the standard CANDU 600 pressure tube design have been retained.

Fuel for the reactor is in the form of cylindrical bundles, approximately 0.5 meters long, each bundle consists of 37 fuel elements which are comprised of approximately 29 fuel pellets stacked end-to-end and sealed in a zircaloy-4 sheath. The fuel elements are attached mechanically at their ends to form a cylinder approximately 100 mm in diameter, with a small space being maintained between each element by spacers attached to the element cladding. Twelve of these fuel bundles are stacked end-to-end in each fuel channel.

The heat transport system is diagramed schematically in Figure 2.3-1. The main heat transport system circulates pressurized heavy water through the reactor fuel channels to remove heat produced by the fission process. The heat is carried to the steam generators where it is transferred to ordinary water to

form steam which subsequently drives the turbine generator. The simplified flow diagram of Figure 2.3-1 illustrates a single primary heat transfer loop. Each loop consists of two steam generators and two primary coolant pumps located at each end of the reactor, so that flow is in one direction through one-half of the fuel channels and in the opposite direction through the other half. In the conceptual HWR design, two such circulation loops are provided, each serving one-half of the reactor core. In addition to the main heat transport system, a separate system is provided to maintain the temperature of the heavy water moderator. Because of the low temperature of the moderator, heat extracted by the moderator cooling system is not used to generate electricity, but rather is dissipated directly by the heat rejection system.

As with all pressurized water reactor concepts, steam generators are utilized to transfer heat from the reactor coolant and to raise the temperature of, and boil, light water contained in a secondary circuit. Steam generated in the steam generators is then utilized to drive the turbine generator. Four identical steam generators consisting of inverted vertical U-tube bundles installed within a shell and containing integral pre-heaters are provided; these U-tube steam generators are identical in concept to those currently provided in PHWR and PWR reactors, differing only in sizing.

During normal operation, the main method for reactivity control is the use of on-power refueling (for first cores, soluble boron moderator poison control is also employed until the excess reactivity of the initial core is depleted). With on-power refueling, the reactor operates with essentially no excess reactivity (except for that contained in the normally inserted adjuster units). Fresh fuel bundles are charged into the fuel channel by a remotely operated fueling machine, while spent fuel bundles are simultaneously discharged into another refueling machine at the opposite end of the reactor.

With on-power refueling, fuel is continually charged/ discharged so as not to require reactivity control systems to compensate for fuel depletion. An H_2O zonal control system and vertical absorber rods are provided to control the small reactivity changes which occur between refueling visits, and to adjust the regional power distribution within the reactor core. Vertical adjuster rods, containing a stainless tube varying in thickness lengthwise, are provided for xenon power override and power distribution flattening. These adjuster rods are normally fully inserted during operation, and are withdrawn after shutdown or operation at low power to increase reactor power by providing the reactivity to override the buildup of xenon.

Two diverse reactor shut-down systems are provided. The first system consists of control rods* which are released upon trip to fall into the core by gravity. The second shut-down system consists of the injection of gadolinium poison into the moderator.

The instrumentation present in the plant encompasses a wide variety of equipment, designed to perform a large number of diverse monitoring, control and display functions. In-core instrumentation is provided to allow automatic control of reactor power and flux shape and to monitor local core behavior. Central to the instrumentation and control system are large capacity digital computers which are employed for station control, alarm annunciation, and data display. The computer system also serves to coordinate reactor power level with turbine demand to manipulate the various reactivity control devices so as to maintain a desired flux shape, to initiate slow and fast power cutbacks to keep the plant parameters within limits which will avoid a reactor trip and to control other plant parameters such as steam generator and pressurizer pressure and water levels.

*Throughout this report, "control rods" is used to refer to any moveable mechanical poison rod used for reactivity control. Canadian terminology divides these rods into "shutdown rods" and "control rods", reflecting their intended use.

TABLE 3.1-1
MAIN STATION TECHNICAL DATA

CAPACITY

Number of reactors	1
Output per reactor:	
total fission	4029 MW
gross electrical	1344 MW
net electrical	1260 MW
Thermal efficiency	31.3%
D ₂ O Inventory	0.65 Mg/MWe

REACTOR-PRESSURIZED HEAVY WATER

Pressure tubes	740
material	Zr-Nb
inside diameter	103.3mm
wall thickness	5.79mm
Active core size (h x d)	594 x 8.76m
Lattice pitch	28.58 cm
Fuel load	8880 bundles 163 tonne U 185 tonne UO ₂
Fuel bundle (37 elements)	
length	495.3mm
diameter	102.49mm
Cladding	Zr.4
Max. fuel temperature	1900°C

MODERATOR

Material	D ₂ O (414 te)
Design temperature	93°C

TABLE 3.1-1 (Cont.)

HEAT TRANSPORT

Type	Pressurized D ₂ O
Flow	50,760 te/hr
Reactor inlet/outlet temps.	290/337°C
Pumps	4
Boilers	4
Preheaters	4
Steam pressure	7.72MPa (1120 psia)
Steam quality	99.75%

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

0 50 100 200
SCALE IN FEET.

FIGURE 3.1-1
PLOT PLAN

3.2 Summary Description of Plant Modifications

This section summarizes the significant modifications that have been incorporated into the conceptual 1260 MWe HWR relative to the CANDU-600 and the 1100 MWe HWR design reported in the Capital Cost assessment performed for the US-ERDA (Reference 3.2-1).

It should not be inferred that the modifications incorporated into the conceptual design in any way indicate inadequacies in the design or performance of the currently operating CANDU's. Rather, these modifications were incorporated into this conceptual HWR design on a U.S. site to either improve fuel utilization, significantly reduce the capital cost, conform to U.S. construction practices, or arose from consideration of U.S. licensing requirements.

The major changes incorporated into the conceptual HWR design are summarized below:

- ° A net electrical output of 1260 MWe, as compared to 1100 MWe for the HWR design in the Capital Cost Assessment and 600 MWe for the CANDU-600.
- ° 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%.
- ° A D₂O reflector thickness of 33 cm, about half the thickness employed in the other HWR designs, in order to reduce the D₂O inventory.

- ° A two-loop primary system configuration incorporating four steam generators and four primary coolant pumps, in contrast to a three-loop configuration. The use of the two-loop configuration allows the containment size and equipment costs to be reduced, thereby reducing the plant capital costs.

The following design modifications were either added or replaced comparable features:

- ° Containment

The containment was modified to eliminate the dousing tank located near the top of the containment shell and to incorporate a U.S. PWR-type ECCS and containment spray systems.

- ° Reactor Service and Fuel Handling Building

This structure was redesigned and resulted in a smaller structure, with some functions being transferred to other structures. The spent fuel storage pool was reduced to 4/3 core capacity from the conventional CANDU pool holding an equivalent of 10 years operation worth of spent fuel.

- ° Control, Auxiliary, and Diesel Building

Functions from various locations were combined in this seismic Category I structure, including redundant emergency electric power diesel generators, redundant control rooms, redundant cable spreading rooms, and all major switchgear. Security and control of personnel within the reactor plant is provided from this building.

- Penetrations Building

This seismic Category I reinforced concrete structure has been provided to protect the main steam and boiler feedwater containment isolation valves, and to house the moderator cooling heat exchangers and moderator circulating pumps.

- Annulus Building

An annulus building has been added between the containment and the Control, Auxiliary and Diesel (CAD) building which among other things house the cables for the control and powering of various electrical and instrumentation equipment in the power plant complex.

- Moderator Cooling System

The moderator cooling system was modified to include a refrigeration system for cooling the moderator fluid. Since the conceptual design utilizes cooling towers for main heat rejection, an ammonia absorption intermediate closed circuit cooling loop offers economic advantages compared to the conventional water-cooled heat-exchangers.

- Protective System Functions

This system has been modified to include four channel actuation for protection functions, as is frequently employed in U.S. PWR's. Two trips of the four are required for actuation of a protective function vs. two of three for CANDU. Two of four channel actuation has the advantage of improved reliability since the failure of one channel will not result in a spurious trip even when a second channel is out of service for testing or maintenance.

- Reactor Coolant System

Stainless steel wetted surfaces have been used in place of carbon steel wetted surfaces.* The advantage of stainless steel is its corrosion and erosion resistance. The feeder tubes from the headers to the pressure channels are stainless steel and the major pipe sections, headers, pump casings and steam generator primary head and nozzles are clad on the interior with austenitic stainless steel. Any connecting piping will be austenitic stainless steel.

In accordance with U.S. General Design Criteria-4, a complete system of supports for the Reactor Coolant System (RCS) have been incorporated. 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 inventory control, or charging and letdown, was changed to include positive displacement pumps as presently used on PWR light water NSSS. Overpressure protection was modified to include power operated relief valves. So that the surge line isolation valves meet single failure criteria, redundant isolation valves were included in each line to the RCS loops.

*Candu plants presently employ carbon steel wetted surfaces and have demonstrated relatively low corrosion in the primary system. There thus appears to be no compelling technical reason for changing wetted surface material, and further, Canadian experience would seem to indicate that carbon steel wetted surfaces would also be suitable for PWR application. Stainless steel wetted surfaces, however, were assumed for consistency with current U.S. PWR construction practices.

The modifications indicated in this section represent the significant changes that have been incorporated in the conceptual HWR design. These and other less significant changes are discussed in Section 5.2. Modifications that resulted primarily from size increase of the plant and the arrangement of the primary coolant system have not been included since these modifications do not represent deviations from the basic CANDU concept.

Reference

- 3.2-1 C00-2477-13, "Capital Cost, Pressurized Heavy Water Reactor Plant,"
United Engineers and Constructors, Inc., June 1977.

3.3 Fuel Management and Resource Utilization

3.3.1 Optimization of the HWR for U.S. Siting

The overall purpose of the Heavy Water Reactor Characterization Program has been to develop a reliable data base for HWRs for use by DOE in comparative assessments and evaluations as part of the Non-proliferation Alternative Systems Assessment Program (NASAP). The primary reason for considering alternatives to the light water reactor in the NASAP program is to identify reactor concepts and fuel cycles which achieve, consistent with non-proliferation goals, better uranium resource utilization. At the same time, it is recognized that such alternate reactor concepts must be economically and commercially feasible to gain acceptance and widespread deployment; power cost economics of the conceptual HWR must be comparable to those of other alternatives if the concept is to be commercially viable and the potential for improved uranium resource utilization realized.

As has been noted previously, heavy water reactors of the Candu-PHWR type have been commercially deployed in Canada and in a number of other nations, and the performance of these units has generally been outstanding. One option, then, of developing a heavy water reactor data base for DOE evaluation would have been to adopt the Candu design, unmodified except as required to meet U.S. licensing and construction practices. Such an approach might enhance the acceptance and deployment of the HWR in the U.S. by most fully utilizing developed technology and proven operating experience.

However, preliminary assessment performed as part of the NASAP program indicated that Candu reactors of existing design would not fully meet the goals and objectives of NASAP. In particular, although the operation of heavy water reactors on the natural

uranium cycle has been quite satisfactory, the resource utilization achieved with natural uranium fueling is only moderately better than in the corresponding LWR once-through cycle. U_3O_8 requirements for the HWR operating with natural uranium fueling are about 20% less than those of currently operating LWRs; although this improvement in uranium utilization is certainly significant, it is questionable whether the reduction in uranium requirements is sufficiently large to warrant development of a new reactor type for U.S. deployment. The capital cost (excluding D_2O) of the HWR is also evaluated to be significantly higher than the capital cost of the LWR when designed to U.S. licensing criteria and construction practices, and results in higher HWR power cost. Consequently, the prospects for commercial deployment of the unmodified Candu-PHWR in the U.S. appear unlikely.

In recognition of these past evaluations, one of the basic objectives of this study has been to develop a conceptual HWR design for use in NASAP evaluations in which careful attention has been given to alternatives which can further improve uranium resource utilization and which would reduce the capital cost of the HWR plant. This design optimization evaluation has led to the incorporation of several significant changes into the conceptual HWR design. In particular, station size was increased to 1260 MW(e) while retaining the two-loop configuration of operating Candu plants (so that the size of each loop is considerably larger), the design is predicated upon the use of slightly-enriched uranium fuel, and modifications have been made to increase station efficiency; in addition, a number of less significant changes were also incorporated into the conceptual HWR design. Summarized in this section are the various design changes which were explored to optimize the fuel utilization and/or economic performance of the HWR for U.S. siting. A more detailed discussion of the optimization evaluation can be found in Chapter 4.

3.3.1.1 Initial Fuel Enrichment

It is generally recognized that, for the once through cycle, significant improvements in uranium resource utilization can be achieved by the use of slightly enriched uranium fuel (References 3.3-1 and 3.3-2). Slightly-enriched fueling also offers the potential for reduced fuel cycle costs, both because of the reduced uranium requirements and because of reduced fuel fabrication and disposal costs which result from the lower fuel through-put at higher discharge exposure. The use of slightly-enriched fuel also makes more attractive a number of the design options summarized in subsequent sections of this chapter which serve to reduce plant capital cost; most of these design alternatives reduce cycle length somewhat and would be unattractive for natural uranium fueling because of the already low discharge burnup.

In order to evaluate the effect of initial enrichment on uranium utilization, a series of calculations parametric in initial enrichment were performed for the equilibrium cycle of operation. U_3O_8 utilization, relative to that of natural uranium fueling, is illustrated as a function of initial enrichment in Figure 3.3-1; this figure also illustrates that the achievable discharge burnup will increase approximately linearly with initial enrichment. U_3O_8 requirements decrease rapidly as the initial enrichment is increased from that of natural uranium to about 1 w/o, and the uranium utilization is rather insensitive to initial enrichment for initial enrichments greater than 1.2 w/o. For an initial enrichment of 1.2 w/o, equilibrium cycle U_3O_8 requirements are about 35% lower than those of the natural uranium fueled system.

Figure 3.3-2 presents the change in fuel cycle costs as a function of initial enrichment. Fuel cycle costs decrease significantly as the initial enrichment is increased from natural uranium to 1.2 w/o as a result of the reduced U_3O_8 requirements and because smaller quantities of fuel must be fabricated and disposed of each year as

a result of the higher discharge exposure.* A rather broad minima in fuel cycle costs occur for enrichments between 1.0 and 2.0 w/o; the optimal enrichment from a fuel cycle standpoint appears to lie between initial enrichments of 1.2 to 1.4 w/o, but an initial enrichment of 1.2 w/o achieves nearly equal fuel cycle cost. The optimal enrichment is rather insensitive to the price of U_3O_8 .

Thus, an initial enrichment of about 1.2 w/o appear optimal both from the standpoint of fuel utilization and economic performance, and there appears to be little incentive to increase the initial enrichment beyond this value. There are a number of incentives for keeping the initial enrichment as low as possible, consistent with obtaining the majority of the reduction U_3O_8 requirements and economic benefits, since slight-enriched fueling does introduce a number of potential design problems which become more severe as the initial enrichment is increased. Consequently, an initial enrichment of 1.2 w/o was selected as the reference value for the conceptual HWR design.

The primary design problems associated with the use of slightly-enriched fueling relate to fuel performance and core peaking factors. The fuel assembly design currently employed in Candu reactors has been developed for the low burnups (7,500 MWd/t) obtained with natural uranium fueling, and if the enrichment/burnup is increased to too high a value, fuel assembly redesign would be required. However, the fuel design assessment indicated that it should be possible to retain the existing Candu fuel design if the burnup is limited to the 20,000 MWd/t value obtained with an initial enrichment of 1.2 w/o; higher burnups would probably require modification of the fuel design to include internal fission gas plenums. The performance of the Candu fuel design at the higher discharge burnups concomitant with slightly enriched fueling, of course, will have to be demonstrated by in-reactor irradiation programs.

*Figure 3.3-2 employs constant fabrication and spent fuel disposal costs, independent of initial enrichment. Fueling costs for the natural uranium cycle would be lower than indicated since somewhat lower fabrication and spent fuel disposal costs are anticipated for this fueling mode.

A second reason for choosing as low an enrichment as is practical is the increase in core peaking factor which results as initial enrichment is increased. The use of slightly-enriched fuel increases both the local peaking factor (within a fuel assembly) as well as channel-to-channel and axial peaking factors. The local peaking factor increases monotonically with initial enrichment because of the increased flux depression which occurs at the center of the fuel assembly. An increase in local peaking factor of about 4% occurs for a 1.2 w/o initial enrichment (compared to natural uranium), and still larger peaking penalties occur for higher enrichments. Increases in both channel-to-channel and axial peaking factors occur as the initial enrichment is increased because of the greater disparity in reactivity between highly exposed and fresh fuel; the magnitude of this peaking penalty depends upon the detailed fuel management employed.

There are a number of design options available to reduce core peaking: local enrichment zoning, fuel management, and a reduction in core-average power density achieved by increasing the number of fuel channels. With local enrichment zoning, the enrichment of the outer ring of fuel in which the maximum power density occurs is reduced, while the enrichment of the interior fuel rods is increased so as to maintain an average enrichment of 1.2 w/o. The enrichment zoning evaluation indicated that the local peaking factor can be reduced from about 12% for the uniform 1.2 w/o case to about 4%, i.e., enrichment zoning can provide about an 8% improvement in power peaking. In fact, the use of local enrichment zoning reduces power peaking by about 4% with respect to natural uranium fueling, thus providing some margin to compensate for the increase in channel-to-channel peaking factor.

The increase in channel-to-channel and axial peaking factors can be minimized through fuel management. The HWR employs partial channel refueling in which only part of the fuel which resides in each fuel channel is replaced during a refueling visit. With natural uranium fueling, a substantial fraction of the fuel assemblies is replaced each visit (typically 8 or 10 of the 12 fuel assemblies which reside in each channel). Because of the higher burn-up of slightly-enriched fuel, it becomes feasible to replace a smaller number of fuel assemblies at each visit. For example, if two of the twelve fuel bundles are replaced at each refueling visit, the three-dimensional peak is reduced to a value within about 20% of that achieved using the normal refueling scheme with natural uranium fuel.

Although more detailed fuel management calculations, in which the full HWR core is explicitly represented, will be required to definitively assess the power peaking implications of slightly-enriched fueling, it appears that the increase in power peaking can be held to within about 15% of that of natural uranium fueling, provided local enrichment zoning is employed and careful attention is paid to the fuel management. The remaining increase in power peaking necessitates that the number of fuel channels be increased so as to preserve thermal margin. The effect on capital cost of increasing the number of channels to accommodate increased peaking is shown in Figure 3.3-3. This change in capital cost results primarily from the change in heavy water inventory, since the effect on calandria cost is very small and since increasing the number of pressure tubes does not significantly affect containment sizing. As Figure 3.3-3 illustrates, the effect on increasing the number of pressure tubes is rather modest, so that the effect of higher peaking can be accommodated by increasing the number of pressure tubes without substantially impacting capital costs.

3.3.1.2 Primary System Pressure Evaluation

The capital cost contribution to total power cost depends both upon the total plant cost (in dollars) and upon the net electrical power output which can be generated from that plant. Consequently, power cost will depend upon the efficiency with which thermal power can be converted into electrical power, and hence power cost can be reduced by improving the net electrical efficiency. The standard Candu-600 reactor operates at a net electrical efficiency of about 29.3%, compared to 33% efficiency achieved by the PWR.* The Candu reactor will, therefore, generate about 12% less electrical energy than a PWR designed to have an identical thermal output. Since the total capital cost for the two plant types are expected to be similar (since both the PWR and HWR have similar or equivalent structures and components), one would expect the Candu reactor to have a 12% higher capital cost contribution to the total power cost than the PWR due to this lower net operating efficiency.

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 of the HWR. The loss of energy to the moderator, about 6% of the total energy produced, is an inherent feature of pressure tube reactor design and little can be done to reduce this loss of efficiency. The reduced efficiency resulting from the lower operating pressures and temperatures of the HWR can, however, be eliminated by increasing the 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 comparable to that of the PWR.

The efficiency with which thermal energy deposited in the coolant can be converted to electrical energy depends primarily upon the

*Net electrical efficiencies of Candu plants range from 29 to 31.7% depending on secondary side design and cooling water conditions at the site. Likewise, PWR efficiencies vary for the same reasons. The PWR and HWR efficiencies quoted above are for roughly equivalent secondary designs and site coolant conditions.

operating pressure of the secondary system, and hence equivalent efficiencies can be obtained by increasing the secondary pressure from the 4.7 MPa (700 psia) currently employed in Candu to a value comparable to the 7.7 MPa (1100 psia) of modern PWRs. Such increase in secondary pressure, however, necessitates increasing the primary coolant inlet temperatures from 266°C currently employed to 299°C, and necessitates an increase in primary system pressure to a value comparable to that of the PWR, 15.5 MPa (2,250 psia) in order to preserve core thermal margins. The effect on power cost of increasing primary and secondary system pressures and temperatures is illustrated in Figure 3.3-4 (this figure includes only the effect of plant effective capital cost on power cost, and neglects the impact of potential changes in fuel cycle cost). As this figure illustrates, increasing coolant conditions to those comparable to that of the PWR results in a significant reduction in power costs.

As noted above, the primary system design must be modified to accommodate the increase in primary system pressures and temperatures. In particular, it will be necessary to increase the thickness of the pressure tubes in order to maintain the same degree of conservatism in pressure tube design under the modified primary operating conditions. This increase in pressure-tube thickness will reduce the achievable burnup and could potentially decrease uranium utilization and increase fuel cycle costs. The achievable discharge burnup and the resulting relative U_3O_8 utilization as a function of initial enrichment is shown in Figure 3.3-5 for both the standard Candu primary system pressure of 1600 psia and for the 2250 psia pressure utilized in PWRs. As this figure illustrates, the achievable discharge burnup is reduced by 1,000 - 1,200 Mwd/t as a result of the 33% increase in pressure tube thickness necessary to accommodate the higher pressure; the decrease in achievable burnup is approximately independent of initial enrichment, although slightly larger reductions in achievable burnup do occur at the lower enrichments.

For natural uranium fueling, the increase in pressure-tube thickness increases U_3O_8 requirements slightly in excess of 10%. This penalty results from the 16% reduction in achievable burnup, which is only partially compensated for by the increase in net electrical efficiency. As the initial enrichment and discharge burnup increases, the burnup penalty becomes a smaller and smaller fraction of the achievable cycle length. For example, for an initial enrichment of 1.2 w/o, the 1000 MWd/t penalty which results from the increase in pressure tube thickness represents only a 4.4% reduction in achievable burnup. For enrichments above 1.1 w/o, the improvement in electrical efficiency proves to outweigh the burnup penalty, and the uranium utilization is consequently somewhat better than that obtained using the standard Candu system pressure and net electrical efficiency. Thus, modifying coolant conditions to obtain efficiencies comparable to that of the PWR appears attractive both from a power cost and uranium utilization standpoint, although the effect on the latter is relatively small.

A review of pressure tube design considerations associated with the use of higher coolant temperatures and pressures indicates a likelihood that the Zr-Nb pressure tubes can operate successfully under these conditions. However, the increased primary system pressure and temperatures do represent significant departure from present Candu design, and the data base on the performance of irradiated Zr-Nb pressure tubes under these operating conditions is quite limited. Consequently, a pressure tube development and demonstration program to develop information on the performance of pressure tubes at higher operating temperatures and pressures will be required prior to the detailed design, development and licensing of an HWR incorporating this modification.

3.3.1.3 Lattice Spacing Evaluation

One method for reducing the capital cost of heavy water reactors is to reduce lattice spacing, i.e., to reduce the distance between fuel channels. Reductions in lattice spacing will reduce the heavy water inventory present in the plant, and hence the contribution to power costs attributable to heavy water charges. Decreasing the lattice spacing will also result in a reduction in achievable burnup, and hence in resource utilization. Improvements in power costs resulting from the reduced D_2O inventory charges will thus be compensated by higher fuel cycle costs resulting from the decrease in achievable discharge exposure.

The evaluation indicated that reductions in lattice pitch result in a significant decrease in achievable cycle length. This reduction in achievable cycle length results in a large degradation of resource utilization for initial enrichments less than 1 w/o, but becomes less significant when higher enrichments are employed. When both the decrease in uranium utilization and the reduction in D_2O inventory are considered, only small overall reductions in power costs are achieved by reducing lattice spacing to an optimum value. Reducing lattice spacing would require a careful evaluation of whether the weakened end-shield structure remains adequate under seismic events and some redesign of the refueling machines so as to allow them to fit into the smaller spacing between channels.

In view of the degradation in uranium utilization, the marginal improvement in power cost and the likely necessity of rather major design changes, it was concluded that there appears to be little incentive for reducing lattice pitch and hence this option has not been incorporated into the reference conceptual HWR design.

3.3.1.4 Reflector Thickness Evaluation

The standard Candu reactor employs a reflector of D_2O having a thickness of approximately 66 cm. This D_2O reflector surrounds the fuel channels and is located between the fuel channels and the calandria tank wall. Reductions in D_2O inventory cost can be realized by reducing the thickness of the D_2O reflector.

As the thickness of the D_2O reflector is decreased, some reduction in achievable cycle length will result due to an increase in core leakage. While this decrease in achievable burnup may prove significant for natural uranium-fueled systems where the achievable burnup is already low, the evaluation of resource utilization as a function of reflector thickness indicated that only a small and acceptable reduction in achievable burnup occurs for the slightly enriched uranium fueled system. Also, the conceptual design, with a net generating capacity of 1260 MWe, is physically larger than the standard Candu-600 design and consequently has an inherently lower neutron leakage. As a result, the conceptual design is less sensitive to the thickness of the D_2O reflector than is the smaller standard Candu plant.

A D_2O reflector thickness of about 31 cm. was selected for the conceptual HWR design, since reducing the reflector thickness below this value would introduce mechanical design problems. This reduction in reflector thickness reduces the D_2O inventory by about 10%, effecting a reduction in power cost of about 0.25 mills per kw-hr.

3.3.1.5 Primary System Arrangement Evaluation

The standardized Candu-600 employs two coolant loops, with each coolant loop containing two steam generators and two primary coolant pumps. The conceptual HWR attains a net power output of

1260 MW(e). The higher electrical capacity of the conceptual design could have been obtained by either increasing the size of each coolant loop or 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 have been, or which shortly will be demonstrated. Thus, this is the simplest approach for increasing station capacity since it makes the greatest use of developed components, and may consequently be the preferred approach for the initial HWR units introduced to the U.S. This three-loop approach was employed in previous conceptual design evaluations (Reference 3.3-3), and has also been considered in Canadian large plant conceptual design studies.

As part of the optimization evaluation, the alternative of employing two large capacity loops was explored, and eventually adopted as the reference configuration for the conceptual HWR design since this configuration was found to result in lower plant capital costs. Reductions in capital cost occur partially from economies of scale (i.e., the four larger steam generators and pumps of the two-loop configuration are less expensive than the six smaller steam generators and pumps which would be required for the three-loop configuration) and partially because the two-loop configuration resulted in a more compact arrangement allowing containment diameter to be reduced to 146 feet, as compared to the 168-foot diameter containment required with a three-loop configuration (Reference 3.3-3).

The primary disadvantage of employing the two-loop configuration is that new and larger steam generators and pumps must be developed. The four steam generators employed in the conceptual HWR design each have evaporation rates of about 2,000 Mg/Hr. Although these steam generators are not large compared to some PWR units (some PWR units 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-600 plant will have somewhat larger steam generators, 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. The primary coolant pumps employed in the conceptual HWR design also are larger than similar pumps employed to date in Candu or light water reactors. The conceptual design employs four pumps, each having an operating load of 9275 kw; the largest pumps employed to date in Candu units are the 8200 kw Bruce units, while the pump size for a large PWR is 6600 kw. Consequently, some development will be necessary to design the slightly-larger pumps required in the conceptual design, but this development is thought to be straightforward, requiring some RD&D, but offering no fundamental technical problems.

As mentioned above, the primary incentive for employing a three-loop configuration would be the use of the standardized components. However, some change in component design would be required at any rate because of the increased temperatures and pressures employed in the primary and secondary systems of the conceptual HWR design. The lower capital cost of the two-loop configuration coupled with the need to redesign loop components to accommodate the higher operating temperatures and pressures led to the adoption of the two-loop configuration in the conceptual HWR design.

3.3.1.6 Moderator Temperature Evaluation

Approximately 6% of the total energy produced in the HWR is deposited in the moderator, primarily from the interaction of the

moderator with neutron and gamma radiation. A moderator heat exchanger must be provided to remove this deposited energy. The standard Candu-600 reactor employs once-through cooling and maintains a moderator temperature of about 71°C. The maintenance of this moderator temperature is facilitated by the use of once-through cooling, which provides ample quantities of low temperature water to the moderator heat exchanger.

In the conceptual HWR design, mechanical draft cooling towers are used in place of once-through cooling and as a result the temperature of the water available to the moderator heat exchanger is higher. Because of the lower temperature differential between the operating temperature of the moderator and the temperature of the closed-cycle cooling water, the moderator heat exchange system would be rather large.

In order to reduce the size and cost of the moderator heat exchange system, the temperature of the moderator was increased to 93°C in the conceptual HWR design. The uranium resource evaluation indicated that this increase in moderator temperature has an insignificantly small effect on uranium utilization, decreasing the cycle length by only about 0.6% in the case of slightly enriched (1.2 w/o) fueling.

3.3.2 Fuel Management and Resource Utilization Characterization

3.3.2.1 Introduction

The purpose of this section is to summarize the fuel management and resource utilization of the conceptual HWR design. Discussed are the slightly enriched once-through uranium fuel cycle, the uranium fuel cycle with plutonium recycle, and the denatured thorium fuel cycles. Economic evaluations are presented in

Section 3.4. A more detailed discussion of the fuel management and resource utilization of the conceptual HWR can be found in Section 5.4.

Table 3.3.2-1 summarizes the lifetime (30-year) U_3O_8 and separative work (SWU) requirements and the annual fissile plutonium production for the various fuel cycles in the conceptual HWR plant. Also shown for comparison are values for the standard PWR; this standard PWR is typical of light water reactors currently in operation, and does not include improvements which are currently under development by DOE 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. Separative work requirements for the slightly-enriched HWR are about 1/4 as large as those of the LWR. 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 less than half the discharge of the natural uranium fueled HWR.

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 once-through fuel cycle), or by the use of the denatured U-235/thorium cycle where U_3O_8 requirements are about 70% of the slightly-enriched once-through fuel cycle. 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. The use of denatured thorium fueling increases separative work requirements

by about a factor of 2.5 as compared to the slightly enriched once-through fuel cycle; separative work requirements, however, are still well below those of the light water reactor. The annual discharge of fissile plutonium in spent fuel is also decreased significantly, to about 10% that of the slightly-enriched once-through fuel cycle.

Each of these fueling modes is summarized in somewhat greater detail in the sections which follow.

3.3.2.2 Reference Slightly Enriched Uranium Once-Through Fuel Cycle

An initial enrichment of 1.2 w/o was selected as the reference value for the fuel management characterization of the slightly enriched uranium once-through fuel cycle. This value for the initial enrichment was selected based upon the fuel enrichment optimization evaluation which 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 peaking problems.

Fuel management information for the reference once-through cycle is summarized in Table 3.3.2-2. The reactor is initially loaded with 166.1 metric tons of natural uranium fuel contained in 8,880 fuel assemblies. Natural uranium, rather than 1.2 w/o enriched fuel, is utilized to fuel the initial core to avoid the necessity of excess reactivity hold-down until an equilibrium burnup distribution is obtained; the use of natural uranium fuel also improves fuel cycle costs by reducing head-end inventory requirements. This initial loading allows the reactor to operate for approximately 1/2 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. At 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.

As Table 3.3.2-2 indicates, the annual U_3O_8 requirements (assuming a 75% capacity factor, 0.2 w/o diffusion plant tails, and a 1.5% fissile loss during fabrication and conversion) are about 114 ST/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 cycle. Separative work requirements are zero for the initial core because of the use of natural uranium fuel and are 39.0 and 1152 MT SWU/GWe for the equilibrium and 30-year cumulative requirements, respectively; these requirements are roughly 1/4 of those of currently operating LWRs. As Table 3.3.2-2 indicates, approximately 149 kg/GWe of fissile plutonium are discharged each year, a value somewhat lower than that of currently operating LWRs. This contrasts with natural uranium fueling where considerably higher annual fissile plutonium discharge occurs, 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 concentration of fissile plutonium in the spent fuel is also rather low (0.3 w/o) as is the fissile content (the fissile content is about 58% of the total plutonium content), making recovery of this plutonium for use in nuclear weapons unattractive.

Although the conceptual HWR design is based upon the use of on-line refueling, an evaluation was also performed to ascertain the effect on uranium utilization of employing a batch-fueling mode (such as currently employed in light-water reactors). Since on-line refueling maximizes the achievable discharge burnup obtainable

from a given initial enrichment and eliminates neutron losses to control poisons, some degradation in uranium utilization is anticipated. Although the containment/surveillance safeguards technique developed for on-line refueling are generally thought to be quite adequate, batch refueling, has been suggested as a method for potentially simplifying safeguards since it would limit fuel handling operations to well-defined intervals.

Lifetime (30-year cumulative) U_3O_8 requirements for batch refueling are compared with uranium requirements for on-line refueling in Table 3.3.2-3. Calculations were performed for both the natural uranium fueled system and for the 1.2 w/o slightly-enriched system assuming either annual or semiannual refueling. In the case of the natural uranium fueled system, an annual refueling interval is impossible, since natural uranium provides insufficient reactivity to allow operation in the conceptual design for a period of a year. Semiannual refueling is possible, but results in high U_3O_8 requirements, nearly twice that of on-line refueling and well in excess of that required to operate a LWR. With an initial enrichment of 1.2 w/o, the conceptual HWR could be operated in a batch-fueled mode in which one-half of the fuel loading was replaced at each annual refueling outage (two-batch fuel management). However, the resulting 30-year U_3O_8 requirement, 5,240 ST/GWe, is nearly 50% greater than when on-line refueling is employed. Semiannual refueling reduces the 30-year U_3O_8 requirements to 4,220 ST/GWe, a value about 20% higher than the U_3O_8 requirements with on-line refueling.

In summary, batch refueling is impractical for the natural uranium-fueled HWR. For slightly-enriched fueling, batch refueling might be considered, but it results in a very significant degradation of uranium utilization. Further, 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. The design modifications which are required in order to perform batch fueling in a fashion which would provide an increase in security against diversion as

compared to on-line refueling would necessitate a major modification to proven Candu-PHWR design.

3.3.2.3 Plutonium Recycle

Although the HWR operating on the slightly enriched uranium once-through fuel cycle utilizes fissile material relatively efficiently, further improvement in uranium utilization can be obtained by the recycle of 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. 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. In this recycle mode, all fresh fuel assemblies would contain natural uranium enriched to 0.3-0.4 w/o fissile plutonium.

Plutonium recycle in the HWR, of course, is analogous to plutonium recycle in the LWR, and is, in general, subject to the same proliferation concerns and considerations. In particular, plutonium is present in a concentrated form during reprocessing (if conventional reprocessing is employed) and is also present in unirradiated fresh fuel assemblies where it could be separated using relatively simple chemical technology. The only advantage of plutonium recycle in the HWR is the relatively low plutonium concentration in the fresh fuel, 0.3-0.4 w/o Pu fissile, compared to the fissile enrichments of about 3.0 w/o contemplated in the LWR. The plutonium recycle mode was evaluated in this report since it is an option which has been considered for eventual deployment in HWRs, and it is of interest to know whether significant improvements in uranium utilization can be obtained and to determine if there is sufficient incentive to explore technical and institutional alternatives which might allow this fueling alternative to be employed consistent with nonproliferation goals.

Fuel cycle information for the uranium cycle with self-generated plutonium recycle in the conceptual HWR is summarized in Table 3.3.2-4. As was the case for once-through fueling, the initial core would be comprised of natural uranium fuel elements. After the first half year of operation, on-line refueling would be initiated in which fresh uranium fuel assemblies are loaded into the reactor and spent fuel discharged. Refueling with uranium fuel continues for a two-year period until recycle can begin. During this period before recycle is initiated, either natural uranium or slightly-enriched fuel can be utilized; in the particular case evaluated, natural uranium fueling was assumed in the interval before recycle was initiated. The plutonium content of the spent fuel is recovered by reprocessing and used to supplement the fissile content of natural uranium fuel. No further utilization of the uranium recovered from the spent fuel is assumed because of the low U-235 content (< 0.2 w/o).

In the self-generated plutonium recycle scheme evaluation, plutonium recovered from the spent fuel is added to an equal volume of (natural) uranium as had been present in the spent fuel. This "topping" with fissile plutonium increases the fissile content of natural uranium to slightly more than 1 w/o fissile, and as a result the achievable discharge exposure increases. The plutonium-enriched natural uranium fuel assemblies are capable of achieving a discharge exposure of about 14,700 MWd/t, while unenriched natural uranium fuel assemblies achieve a discharge exposure of 6,300 MWd/t in the conceptual design; the recycle of plutonium consequently increases the energy which can be extracted from natural uranium fuel by more than a factor of two. As indicated in Table 3.3.2-4, annual U_3O_8 requirements are 77.2 ST/GWe, or about 2/3 of the U_3O_8 requirements of the conceptual HWR operating on the slightly-enriched once-through uranium fuel cycle. The 30-year cumulative U_3O_8 requirements, 2,671 ST/GWe, are about 25% less than that of slightly-enriched fueling. The quantity of fissile plutonium discharged each year, 218 kg/GWe, is only slightly larger than the 149 kg/GWe discharged in the slightly-enriched once-through cycle, and considerably

smaller than the 350 kg/GWe discharge with natural uranium once-through fueling.

3.3.2.4 Denatured Thorium Fuel Cycles

As supplies of reasonably-priced U_3O_8 are consumed, it may be desirable to shift to denatured thorium fueling in the HWR, since significant further reductions in uranium requirements can be realized with this fueling mode. Denatured thorium fueling consists of a mixture of thorium and uranium oxides. The uranium component is enriched to up to 20 w/o U-235 or 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. 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. The second denatured thorium 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.

Since U-238 is present in limited quantities in denatured thorium fuel, some plutonium is produced. This plutonium could either be stored or disposed of as waste, or utilized to fuel a variety of converter or breeder reactors. 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 subjected 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 exogeneous U-233 requirements of the denatured U-233/thorium fuel cycle.

Fuel management information for the denatured U-235/thorium cycle is summarized in Table 3.3.2-5. This table illustrates two examples of denatured thorium fueling, in which the enrichment (w/o fissile in total heavy metal) has been adjusted to obtain discharge exposures of either 13,700 or 37,000 Mwd/t. The lower burnup case is an example of fueling options which seek to minimize U_3O_8 requirements; low burnup enhances fuel utilization since fewer neutrons are lost to parasitic captures in fission products and consequently the conversion ratio is larger. The higher discharge exposure case is representative of fuel management schemes which seek to minimize fuel cycle cost. Fuel cycle costs are minimized by the higher discharge exposure because smaller quantities of fuel have to be reprocessed and fabricated each year (approximately 30 MT/yr of fuel must be fabricated and reprocessed in the high burnup case, compared to 80 MT/yr in the low-burnup example). The savings in reprocessing and fabrication costs typically proved to outweigh the additional costs incurred for the purchase of the larger quantities of U-235 makeup required because of the decrease in conversion ratio which occurs at higher burnups.

As Table 3.3.2-5 illustrates, the annual U_3O_8 requirements for the U-235/Th cycle are less than half as large as the uranium requirements for slightly-enriched once-through fueling. The 30-year cumulative requirements for the U-235/Th cycle 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 someplace in the fuel reprocessing-refabrication stream awaiting recycle). It is interesting to note that although the annual reload U_{308} requirements are significantly lower for the low-burnup fuel-management schedule as compared to the high burnup schedule (20%), the 30-year cumulative requirements are comparable. This is because the annual throughput of fuel is much higher in the low burnup case, and consequently the ex-core inventory is larger (the ex-core inventory is assumed to be equal to a 2-year fuel throughput), as is the loss of fissile material during reprocessing and refabrication (reprocessing and refabrication losses are assumed to total 2% of the fuel throughput). 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_{308} 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 spent between fuel discharge and the reloading of recycle assemblies (ex-core time).

The 30-year cumulative separative work requirements for the denatured U-235/Th fuel cycle are about twice that required for slightly-enriched uranium fueling in the HWR, but still are about 30% lower than the separative work requirements of light water reactors operating on the once-through fuel cycle. As Table 3.3.2-5 illustrates, separative work requirements are higher because of the initial core loading and because of the 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 (6.5 to 12%, for the high and low burnup examples, respectively).

Comparison of the 30-year cumulative U_3O_8 requirements for the U-235/Th cycle with those of the uranium cycle with Pu recycle (cf. Table 3.3.2-4) indicates, from a uranium utilization standpoint, little reason for preferring one or the other fueling mode, since either will achieve a 20-25% reduction in uranium requirements as compared to the slightly-enriched once-through fuel cycle. However, the two fuel cycles are fundamentally different in that the initial loading requirements for U_3O_8 are considerably higher for thorium fueling (about 4 times greater) while the annual reload requirements for thorium fueling are 30-45% lower than those of the Pu recycle case. This suggests that the uranium cycle with plutonium recycle may be particularly attractive during times of rapid growth in nuclear capacity, so as to minimize uranium requirements to inventory new plants coming on line. The denatured U-235/thorium cycle is likely to be particularly attractive during periods of slow nuclear growth, where uranium demand will be dominated by the U_3O_8 requirements for reloading operating plants. In this instance, the low annual U_3O_8 requirements of the denatured U-235/thorium cycle are particularly important; for a given annual U_3O_8 production capacity, the use of the denatured U-235/thorium cycle would allow 1.5 to 1.8 times as many reactors to be operated as would be possible with plutonium recycle, or 2 to 3 times as many reactors as could be operated with slightly enriched once-through fueling.

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. As was the case for the U-235/Th cycle, annual fissile makeup requirements increase as the discharge exposure is increased, as is illustrated in Figure 3.3-6. This figure also illustrates the effect of discharge exposure on 30-year cumulative U-233 requirements; 30-year cumulative requirements can actually increase at very low exposures as a result of the increase in ex-core inventory and of fissile losses during reprocessing and refabrication. The optimal burnup from the standpoint of U-233 utilization appears to occur at a discharge burnup of approximately 14,000 Mwd/t, although U-233 requirements

are seen to increase rather slowly for higher discharge exposures (higher discharge exposures are advantageous from a fuel-cycle cost standpoint since the lower fuel throughput achieved with higher discharge exposures reduces fabrication and reprocessing requirements; savings in refabrication and reprocessing costs generally outweigh the slight increase in U-233 requirements).

Fuel cycle information for the denatured U-233/Th cycle are summarized in Table 3.3.2-6. The particular fuel management schedule summarized in this table has a discharge exposure of 14,000 MWd/t, which is near optimal from the standpoint of minimizing U-233 requirements. As the table indicates, approximately 6,850 kg/GWe of U-233 are required to inventory and fuel the conceptual HWR for 30 years of operation; approximately half this quantity is consumed during operation, while the remaining half is comprised of in-core and ex-core inventories. The 30-year cumulative U-233 requirements are nearly a factor of two lower than the 11,800 kg U-233/GWe required to fuel a LWR, while annual U-233 requirements are approximately a factor of three lower than the 316 kg U-233/GWe required in LWRs. The fissile plutonium production, about 25 kg/GWe, is approximately a factor of 7.5 lower than the fissile plutonium content of spent fuel in the slightly enriched uranium fuel cycle.

As mentioned above, both the denatured U-235/Th and U-233/Th produce some plutonium from the U-233 denaturant. The fuel management information presented for these denatured thorium fuel cycles have assumed that this plutonium was stored and that only the uranium recovered from spent fuel was recycled. An alternative 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 colocated 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/Th cycle at dispersed sited reactors.

Fuel cycle information for the plutonium-enriched thorium cycle (Pu/Th cycle) is summarized in Table 3.3.2-7. Illustrated here is one particular fuel management schedule having a discharge exposure of 23,400 MWd/t; other fuel management schedules can, of course, be employed having either higher or lower discharge exposures. 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. The particular case illustrated in Table 3.3.2-7 consumes approximately 578 kg/GWe of fissile plutonium per year, while producing about 308 kg/GWe of U-233/yr; a conversion efficiency of about 0.53 is thus obtained.

The effect of utilizing the plutonium produced from the denatured U-235/thorium cycle in the Pu/Th-fueled reactor is illustrated in Table 3.3.2-8. Table 3.3.2-9 provides similar information on the effect of plutonium utilization on the U-233 requirements of the denatured U-233/Th cycle. As these tables illustrate, utilization of plutonium in the Pu/Th-fueled reactor reduces fissile requirements by 4-8%, depending upon the particular denatured cycle employed (U-235 or U-233) and upon the burnup schedule. One or two plutonium fueled HWRs would serve to consume the plutonium produced by about 50 HWRs operating on the denatured thorium cycle.

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- 3.3-3 COO-2477-13, "Commercial Electric Power Cost Studies--Pressurized Heavy Water Reactor Plant," United Engineers and Constructors, Inc., June 1977.

TABLE 3.3.2-1
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	1.18
HWR - Natural U Fueling (Once-Through)	5000	1.41	0	0	350	2.35
SEU HWR - 1.2 w/o (Once-Through)	3550	1.00	927	1.0	149	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	0.15
37,000 MWD/MT	2500	0.71	2360	2.55	12.0	0.08
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	-	-

TABLE 3.3.2-2
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 3.3.2-3
URANIUM REQUIREMENTS IN THE BATCH FUELING MODE

	Discharge Burnup (Mwd/MT)	Number of Batches	30-Year Cumulative U_{308} 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 3.3.2-4
FUEL CYCLE INFORMATION FOR THE
URANIUM CYCLE WITH PLUTONIUM RECYCLE

<u>Initial Core</u>	
Enrichment, w/o	0.711 (Nat. U)
Core Loading, MT	166.1
Fabrication Loss, %	1.0
<u>Reload Fuel (First 2.0 yrs)</u>	
Enrichment, w/o	0.711 (Nat. U)
Discharge Exposure, MWd/MT	6300
Fuel Throughput, MT/yr	175.4
Fabrication Loss, %	1.0
<u>Reload Fuel (Remaining Years)</u>	
Enrichment, w/o	Nat. U + Self-Generated Pu Recycle
Discharge Exposure, MWd/MT	14,740
Fuel Throughput, MT/yr	74.8
Fissile Losses, %	
Fabrication	1.0
Reprocessing	1.0
Ex-Core Time, yrs	2
<u>U₃O₈ Requirements (ST/GWe)*</u>	
Initial Core	170.9
Annual Equilibrium	77.2
30-year Cumulative	2,671
30-year Cumulative, Net**	2,315
<u>SWU Requirements</u>	
Annual Fissile Pu Discharge, kg/GWe	218

* 75% Capacity Factor

**30-year cumulative, net requirements are 30-year cumulative requirements less credit for in-core and ex-core fissile inventories at the end of the assumed 30-year reactor lifetime.

TABLE 3.3.2-5
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 3.3.2-6
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 3.3.2-7
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 3.3.2-8
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-233/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

RELATIVE URANIUM UTILIZATION
AS A FUNCTION OF INITIAL ENRICHMENT

FIGURE 3.3-1

Figure 3.3-2
CHANGE IN FUEL CYCLE COST WITH INITIAL ENRICHMENT

Figure 3.3-3
CHANGE IN CAPITAL COST vs NUMBER OF PRESSURE TUBES

Figure 3.3-4
EFFECT OF PRIMARY SYSTEM PRESSURE ON POWER COST

FIGURE 3.3-5

RESOURCE UTILIZATION AT 1600 AND 2250 PSIA
 PRIMARY SYSTEM PRESSURE

FIGURE 3.3-6

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

3.4 Economic Evaluations

3.4.1 Capital Cost Estimates

Plant capital costs for the conceptual HWR design are summarized in Table 3.4-1. These capital costs are based upon the prices in effect as of January 1, 1978 and are quoted in January 1, 1978 dollars. These costs were estimated for a "nth-of-a-kind" plant and hence exclude any allowances for contingency. Also excluded are the cost of the initial heavy water inventory, interest and escalation during construction, and owner's cost.

The resulting plant capital cost (excluding heavy water) is \$740,124,000 or 587\$/kWe. As is the practice throughout this report, the cost of the initial heavy water inventory has been excluded from the tabulation of plant capital cost to facilitate intercomparison with the LWR, and treated instead as a component of the fuel cycle cost. At a D_2O price of 97\$/lb, the cost of the heavy water inventory would add about 139\$/kWe to the capital cost of the conceptual HWR design. Treating the cost of this initial D_2O inventory as a component of plant capital cost or, as has been the practice in this report, as a component of fuel cycle cost yields the same total power cost.

The capital cost of the HWR design (excluding the cost of heavy water) is about 8% higher than that of the LWR, or about 34% higher when the cost of heavy water is included. This capital cost differential appears reasonable in view of the lower net electrical efficiency of the HWR (31.3 vs. 33% for the LWR) which necessitates a physically larger plant to generate an equivalent electrical output. HWR costs are also larger because of such extra equipment as the refueling machines, D_2O upgrader, and moderator cooling system. The cost differences between the HWR and LWR are primarily in the nuclear steam supply system and related equipment.

The capital costs of the conceptual HWR design are considerably lower than previous estimates (Reference 3.4-1) for a U.S.-sited HWR, which indicated capital costs approximately 17% higher than those of the LWR. The reduced capital costs of the conceptual HWR are due primarily to increased electrical efficiency (resulting from increasing primary and secondary system pressures and temperatures) and the use of a two-loop configuration (in contrast to the three-loop configuration used in the referenced study) which allows a more compact containment arrangement. Cost reductions also resulted from the consolidation of structures, the elimination of the dousing tank and from a reduction in spent fuel storage capacity (to that specified as the NASAP design basis).

Although the HWR capital costs developed in this study are felt to be reasonably accurate, it must nevertheless be recognized that design changes which could significantly impact capital cost may be required as a result of licensing review. Uncertainties in plant capital cost are also introduced by the conceptual nature of this study (design details could influence component and construction costs) and by the limited experience in the U.S. of constructing and fabricating HWR systems.

TABLE 3.4-1
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.

3.4.2 Fuel Cycle and Power Cost

Busbar power costs for the various fueling modes explored are summarized in Table 3.4.2-1. 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 any of the plutonium recycle or the denatured U-235/thorium fuel cycles for both the near and long-term economic assumptions. The same conclusion appears to hold for the denatured U-233/thorium 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 refueling 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 3.4.2-1, the Pu/Th cycle, cannot be viewed in isolation since it utilizes plutonium produced from the slightly enriched HWR and hence is dependent upon the cost of plutonium recovered from this system. The 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 (i.e., the power costs for the HWR on the uranium cycle with Pu recycle are lower), but 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 either operating on the once-through cycle or on the uranium cycle with plutonium recycle.

Under the reference economic assumptions employed in the power cost analysis (see Table 3.4.2-2), slightly-enriched fueling in the HWR becomes economically competitive with the once-through fuel cycle in the (standard) LWR when 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. The break-even U_3O_8 price is most sensitive to the capital cost differential between the HWR and LWR (an evaluated capital cost differential of 8.3% was used in the analysis) and 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).^{*} The economic analyses indicate that the recycle of plutonium in the self-generative mode evaluated in this study does not become economically competitive with the once-through slightly-enriched fueling mode until the price of U_3O_8 rises to about \$200/lb; alternate recycle modes which employ higher plutonium enrichments and increased discharge exposures, however, are expected to prove more economical. The break-even U_3O_8 price for the denatured U-235/Th fuel cycle is similar (\$180/lb) if high discharge burnup (37,000 MWd/t) schedules are employed; lower discharge exposures are unlikely to prove economic unless it is possible to develop less expensive refabrication and/or reprocessing methods. A similar conclusion can be reached on the U-233/Th cycle, although the economic performance of this cycle is less easy to ascertain because it is dependent upon the price for which U-233 can be purchased.

The power cost economics of each of the refueling options is reviewed in slightly greater detail in the sections below. A more detailed discussion can be found in Section 5.4.

^{*}Although the differences in LWR and HWR design would seem to justify only the 2.5 percentage point difference in capacity factor cited above, 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 3.4-2). This difference in capacity is not fully understood, and is believed to be due, at least in part, to factors other than inherent differences in HWR and LWR reactor plant design. However, if this larger differential in plant capacity factor could be realized in a U.S.-sited HWR of the conceptual plant design, the breakeven U_3O_8 price would decrease to below 10\$/lb.

3.4.2.1 Slightly-Enriched Uranium Fueling

Total busbar power costs for the slightly-enriched (1.2 w/o) conceptual HWR are summarized in Table 3.4.2-3, along with similar information for the standard LWR for comparison. The busbar power costs are comprised of three components: capital cost, O&M, and fuel cycle cost. The capital cost component is based upon a total capital cost (in January 1, 1978 dollars) for the HWR of 764.8\$/kWe (exclusive of heavy water inventory) and of 706.5\$/kWe for the LWR; these capital costs are obtained by adding interest during construction and owner's cost to the reference plant capital cost summarized in Section 3.4.1. The total capital cost of the HWR (exclusive of heavy water inventory) is thus evaluated to be approximately 8.3% greater than that of the LWR. The cost of the heavy water inventory in the HWR is included as a component of the fuel cycle cost. In order to convert the capital cost to mills/kw-hr, it is necessary to assume a plant capacity factor. For the LWR, a capacity factor of 75% was 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. The operating and maintenance cost for the HWR is slightly higher than that of the LWR, with the difference attributable to the cost of D₂O makeup and upgrading due to irrecoverable and recoverable losses, respectively.

As Table 3.4.2-3 illustrates, the 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 and separative work. The cost of heavy water represents a substantial fraction of the HWR fuel cost and is partially, but not completely, compensated by the lower separative work requirements.

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 , as Figure 3.4.2-1 illustrates, and as a result the total busbar power costs for the HWR will be lower than that of the LWR for higher U_3O_8 prices. Figure 3.4.2-1 also illustrates that HWR and LWR power costs are equal at a uranium price of \$55/lb (breakeven price) assuming an HWR capacity factor of 77.5%, and that the breakeven price would increase to about \$75/lb if the capacity factor for both units were assumed to be equal.

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 breakeven U_3O_8 price to various economic assumptions was explored and was found, as Figure 3.4.2-2 illustrates, to be particularly sensitive to the HWR capital cost penalty and to the assumed difference between HWR and LWR capacity factors. As Figure 3.4.2-2 illustrates, the conceptual HWR would be competitive today (i.e., for uranium prices below \$30/lb) if it could be constructed at essentially the same cost as an LWR (zero capital cost penalty). On the other hand, if the modifications incorporated into the conceptual design to reduce plant capital cost (higher operating pressure, use of a two-loop configuration and larger size components, and a higher net station capacity) are not realized, a capital cost penalty greater than the 8.3% reference value should be anticipated, and breakeven U_3O_8 prices could increase to 90-100\$/lb or higher. As this figure also illustrates, the difference in capacity factor between the HWR and LWR can also significantly affect the breakeven U_3O_8 price. The reference value assumes that the HWR will operate with a capacity factor 2.5 percentage points higher than that of the LWR because of on-line refueling, and results in a breakeven U_3O_8 price of \$55/lb. The breakeven U_3O_8 price would increase to about 75\$/lb

if the capacity factor of the conceptual HWR proved equal to that of the LWR. On the other hand, operating Candu plants have demonstrated capacity factors approximately 10 percentage points higher than those of LWRs operating in the U.S.; should the conceptual HWR achieve similar excellent performance, the breakeven U_3O_8 price would decrease to below 30\$/lb.

In summary, although the conceptual HWR operated on the slightly-enriched once-through fueling mode would have slightly higher power costs under the reference economic assumptions than an LWR operated in the same fueling mode, the difference in power cost is relatively small and will decrease as the price of U_3O_8 increases; for U_3O_8 prices above 55\$/lb lower power costs would be achieved by the conceptual HWR. However, the economic competition between the two reactor types is sensitive to a number of parameters such as HWR capital cost and capacity factor. In addition, improvements which are under development to reduce uranium consumption in the LWR could also significantly alter this economic balance. Consequently, it is not possible to draw a definitive conclusion as to whether or not the HWR is economically competitive with the LWR. The construction and operation of a U.S. sited HWR will be required to reduce the uncertainty in important economic and performance parameters before a definitive conclusion can be reached.

3.4.2.2 Uranium Cycle with Plutonium Recycle

Fuel cycle costs for the uranium cycle with plutonium recycle are compared with those of the slightly enriched once-through uranium fueling mode in Table 3.4.2-4. Only fuel cycle costs are compared here since the capital and O&M costs are assumed to be equal in both cases. As this table indicates, the fuel cycle cost for the plutonium recycle mode is higher than that for slightly-enriched once-through fueling for both the near and long-term cost assumptions (see Table 3.4.2-2). Inspection of this table indicates that the plutonium recycle mode is less economic because of the higher contributions from fuel fabrication and reprocessing/disposal

which proved to outweigh the savings in uranium and separative work.

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

3.4.2.3 Thorium Fuel Cycle

Estimates of the fuel cycle cost for the denatured U-235/thorium fuel cycle are compared with the slightly-enriched once-through cycle in Table 3.4.2-5. As this table illustrates, fuel cycle costs for the denatured U-235/thorium cycle are highly dependent upon 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 the denatured U-235/thorium and slightly-enriched uranium fuel cycles are shown in Figure 3.4.2-4 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 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 U-235/thorium fuel cycle would

become economically competitive with slightly-enriched once-through fueling for the high burnup case; uranium prices would have to be quite high before the low burnup case became economically competitive under the economic assumptions of this analysis. It would appear that if the denatured thorium fuel cycle is to become economically competitive with once-through fueling for reasonable U_3O_8 prices, less expensive methods of reprocessing and/or of refabricating U-233-bearing fuel must be developed.

Estimates of the fuel cycle cost for the denatured U-233/thorium cycle are compared with those for slightly-enriched once-through uranium fueling in Table 3.4.2-6. 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\$/gm.

As Table 3.4.2-6 illustrates, the denatured U-233/thorium cycle is unlikely to be economically competitive with slightly enriched uranium fueling in the foreseeable future, for even under the long-term assumptions (which are characterized by a U_3O_8 price of 100\$/lb), fuel cycle costs would be higher for the denatured U-233/thorium cycle even if U-233 could be obtained free of charge. As this table illustrates, the main reason for the higher fuel cycle cost of the denatured U-233/thorium cycle 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), and the higher cost of fuel reprocessing as compared to spent fuel disposal (271\$/kg vs. 145\$/kg for SEU fuel disposal). In Figure 3.4.2-5, the breakeven U_3O_8 price is plotted as a function of U-233 price. As this figure illustrates, the U-233/thorium cycle would not be economically competitive with the slightly-enriched once-through uranium cycle until U_3O_8

prices reached about 120\$/lb, even if U-233 could be obtained at no cost; for more reasonable U-233 prices, the breakeven U_3O_8 price begins to approach 200\$/lb.

Fuel cycle costs for the denatured U-233/thorium cycle could, however, be reduced 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 is reduced. The effect of discharge exposure (initial enrichment) on the fuel cycle cost for the denatured U-233/Th cycle is illustrated in Figure 3.4.2-6. The fuel cycle costs illustrated here are only approximate, since 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 employed a U-233 price of 30\$/gm and 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.

Fuel cycle costs for the Pu/Th cycle are compared to those of once-through uranium fueling in Table 3.4.2-7 under both near and long-term economic assumptions. Pu/Th fuel cycle costs will, of course, depend both upon the price at which fissile material can be purchased and the price for which U-233 can be sold; hence, the fuel cycle costs shown in this table are tabulated parametric in these parameters. As this table illustrates, fuel cycle costs are higher than those of the slightly-enriched once-through fuel cycle for all values of Pu and U-233 prices explored for the near-term economic assumptions. Fuel cycle costs are higher than those of slightly-enriched fueling because of the increased fuel fabrication and reprocessing cost components, which prove to outweigh the savings in U_3O_8 and separative work. 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). The Pu/Th cycle becomes more competitive under the long-term economic assumptions partially because of the increased cost of U_3O_8 (which increases the fuel cycle cost for slightly-enriched once-through fueling) and partially because of the reduced fuel reprocessing and fabrication costs which characterize the long-term economic cost assumptions.

The breakeven U_3O_8 price as a function of U-233 sale value is shown in Figure 3.4.2-7 (plutonium value is based upon cost of recovery in the HWR, c.f. Section 5.4.4.3). As this figure illustrates, the Pu/Th cycle is unlikely to become economically competitive with once-through fueling under the near-term economic assumptions. 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.

TABLE 3.4.2-1
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 3.4.2-2
ECONOMIC ASSUMPTIONS

	<u>Near Term</u>	<u>Long Term</u>
U ₃ O ₈ , \$/lb	40	100
Conversion, \$/kg	4	4
Separative Work	100	100
Fabrication, \$/kg		
SEU-LWR	106	106
SEU-HWR	72	72
Denatured U-235/ThO ₂	99	80
Denatured U-233/ThO ₂	567	487
Pu/ThO ₂	370	318
Pu/UO ₂	361	310
Reprocessing, \$/kg		
UO ₂	280	197
ThO ₂	291	204
Spent Fuel Disposal, \$/kg		
LWR	150	150
HWR	145	145
Capacity Factor, %		
LWR	75	75
HWR	77.5	77.5
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
Tails Assay, w/o	0.2	0.2

TABLE 3.4.2-3
 BUSBAR POWER COST
 (mills/kw-hr)

	1.2 w/o HWR		Standard LWR	
U ₃ O ₈ Price, \$/lb	<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>	-	-
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 3.4.2-4
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 3.4.2-5

**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 3.4.2-6
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	<u>-</u>	<u>-</u>
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 3.4.2-7

**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

FIGURE 3.4.2-1

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

FIGURE 3.4.2-2

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

FIGURE 3.4.2-3

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

FIGURE 3.4.2-4

FUEL CYCLE COST AS A FUNCTION
OF U_3O_8 PRICE

FIGURE 3.4.2-5

BREAKEVEN U_3O_8 PRICE AS A FUNCTION
OF U-233 COST

FIGURE 3.4.2-6

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

FIGURE 3.4.2-7

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

Reference

- 3.4-1 C00-2477-13, "Commercial Electric Power Cost Studies--Pressurized Heavy Water Reactor Plant," United Engineers and Constructors, Inc., June 1977.
- 3.4-2 "Comparison of Ontario Hydro with World Power Reactors 1978," Ontario Hydro.

3.5 Safety, Licensing, Environmental Impact

3.5.1 Approach to the Safety and Licensing Review

A qualitative assessment of safety and licensability was performed as part of this evaluation to identify safety-related and environmental issues which would require significant licensing review, or which represent potential problem areas impacting research and development needs. 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 the Canadian licensing and acceptance criteria against which Candu plants are licensed, the design basis of the conceptual HWR, its inherent operating characteristics, and the likely safety and licensing implications of the design changes which have been incorporated into the conceptual HWR design to improve uranium utilization or economic performance.

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 design 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. licensing criteria. Thus, such aspects of the engineered features as containment design and liner, emergency core cooling system, and seismic constraints have been replaced with U.S.-type systems. Since basically similar engineered design features are employed and licensed in

U.S. pressurized water reactors, there is a high degree of assurance that the engineered features of the conceptual HWR conform, or can be made to conform, to U.S. licensing requirements without engendering fundamental questions of licensability or of significantly impacting research and development needs.

Safety and licensing questions consequently center upon the inherent characteristics of the HWR, the acceptability of postulated transients and other accidents, and upon the safety implications of design changes which have been incorporated into the conceptual HWR to enhance uranium utilization and to improve power cost economics. The inherent features of the conceptual HWR are reviewed in Section 3.5.2 with the intent of identifying aspects which may require extensive licensing review. Section 3.5.3 attempts to provide some insight into the consequences of various accidents by reviewing Canadian licensing acceptance criteria and comparing these criteria with those employed in the U.S. Lastly, Section 3.5.4 reviews the safety and licensing implications of the design changes which have been incorporated into the reference conceptual HWR to improve uranium utilization or economic performance.

3.5.2 Inherent Characteristics

The inherent characteristics of a concept are the design features and core transient performance characteristics which are fundamental to the design of the reactor or 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 on safety and licensability when all the unique attributes of the system, both positive and negative, are considered.

The prime inherent characteristics of the pressurized heavy water reactor which contribute to increased plant safety are considered to be the use of the pressure tube configuration, the on-line refueling feature, and the separate low-pressure moderator system. The use of the pressure tube configuration in the HWR, coupled with the injection of emergency core cooling water into both inlet and outlet headers, facilitates the supply of coolant 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, and also facilitates incorporating a second and independent shutdown 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 potentially available for release 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, the positive void coefficient of reactivity, and the increased tritium production which is inherent in heavy water systems. These inherent characteristics are reviewed in somewhat greater detail below.

° Pressure Tube Configuration

In HWRs of the Candu type, zirconium alloy pressure tubes (4-6 mm., depending upon system pressure) serve as the primary system pressure boundary. The possibility and degree of pressure tube failure, along with the potential consequence (if any) of credible failures, must be considered as part of the safety analysis. 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 tubes. Pre-service and in-service inspections should be sufficient to ensure that large cracks are not present. Under these circumstances, crack propagation, should it occur, would result in a penetration of the clad wall and coolant leakage, rather than a catastrophic pressure-tube failure. A leak of a pressure tube is readily detectable by means of the surveillance system provided to sample 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.

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 fitting. The use of these rolled joints has 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.

The fact that the 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 the Pickering units where cracking occurred in the vicinity of the rolled joints of a number of pressure tubes (Reference 3.5-1). 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 plant. 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 (References 3.5-2 and 3.5-3).

The pressure tube configuration also has important implications for 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. As a result, consequences of the loss-of-coolant accident are much less dependent on ECC 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.

Since pressure tubes and the rolled joints which are used to connect these pressure tubes to their end fittings are not employed in light water reactors, these aspects of HWR design are expected to be the subject of considerable review and discussion during the licensing of a U.S. sited plant. As noted in Section 3.5.4, R&D on pressure tube performance and integrity will be required for licensing in order to extend the data base on pressure tube performance to the higher operating pressures and temperatures of the conceptual HWR design.

- Independent Moderator System

Reactors of the Candu type employ a separate heavy water moderator which operates at relatively low temperatures 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. 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. Likewise, control (shutdown) rods are inserted into the moderator, avoiding the necessity of penetrating the pressure boundary as required in LWRs.

° Coefficients of Reactivity

The power coefficient of reactivity at operating power levels is near zero, and the coolant void coefficient of reactivity 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 coefficient of reactivity is thus viewed as an inherent characteristic of the HWR which is less satisfactory than that of light water reactors.

The plant protection system is designed to terminate power transients well before design limits are approached. A second independent shutdown system is provided to ensure that transients are terminated in the event of failure of the main (control rods) shutdown system. Also partially mitigating the effect of these positive coefficients is a longer mean neutron lifetime (approximately 30 times larger than for LWRs), which results in much slower prompt critical power transients than would occur in an LWR for an equal amount of reactivity insertion.

Canadian accident analyses, of course, include the effect of the positive coefficients of reactivity and indicate that these coefficients have little impact upon the consequences of various transients and postulated accidents. Although Canadian analyses indicate that these positive coefficients of reactivity do not significantly impact the safety analysis, it is noted that U.S. licensing criteria explicitly require that the plant be designed so that (in the power operating range) prompt inherent nuclear feedback characteristics tend to compensate for rapid increases in reactivity (General Design Criteria 11). The positive reactivity coefficients of the HWR are consequently considered to be a likely licensing issue. Licensing discussions are expected to center on the consequences of these coefficients during various transients and postulated

accidents, and upon whether such attributes of the HWR which tend to mitigate the effects of these positive coefficients (in particular, the longer neutron lifetime and a second independent shutdown system) are sufficient to justify changing the design criteria currently imposed on light water reactors.

- ° 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 amount of excess reactivity held up in control poisons is considerably less in the HWR because of its on-line refueling feature. Both the ability to detect and remove damaged fuel and the reduced reactivity holdup in control poisons are considered to be positive features of on-line refueling.

On-line refueling, however, does result in the frequent breach of the pressure boundary, potentially increasing the probability of small-scale loss-of-coolant accidents. The effect of on-line refueling on the probability of such small-scale LOCAs and of the consequence of these postulated accidents is expected to be the subject of licensing evaluations for a U.S.-sited plant.

- ° 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 tritium-bearing heavy water. The majority of the tritium inventory is contained in the moderator region and the isolation of the moderator from the environment is simplified by the low pressure at which it operates.

Even though the design of the HWR seeks to minimize heavy water losses, tritium release from the HWR is expected to be several times that of LWRs, and hence could become a licensing issue. However, it is expected that HWR tritium release will be judged acceptable; also, tritium removal has been demonstrated on a small scale, and it appears that this technology could be applied, if required, to the HWR for tritium level control.

3.5.3 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 acceptance criteria.

Canadian licensing practice requires that all accidents (process failures) be analyzed under both the assumption that safety systems function properly* (single failure) and under the assumption that any one of the safety systems was in a failed state when the process failure occurred (dual failure). Under U.S. licensing requirements, only the single failure must be addressed. Canadian acceptance criteria are based solely upon site boundary doses. Prevention of centerline fuel melting is established as a conservative secondary criteria for maintaining the integrity of the primary systems pressure boundary, thus precluding the release of coolant activity and ensuring that site boundary limits are not exceeded. Different site boundary doses 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 such limiting faults under U.S. acceptance criteria. In this respect, Canadian licensing criteria are much more restrictive than are U.S. criteria.

*For single failures, credit is taken for the operation of only one of the two shutdown systems.

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 place explicit limits on fuel damage; however, in practice, Candu is designed to prevent fuel damage during expected events because of the economic consequences of such fuel damage, and limitations are also placed on coolant activity levels. It is impossible to conclude whether the imposition of fuel failure criteria on the HWR design will necessitate a reduction in core power density, or whether the margins provided to prevent the economic consequences of fuel damage would be sufficient to satisfy U.S. regulations; a detailed quantitative evaluation would be required to resolve this question. Peak clad temperatures during the limiting loss-of-coolant event also exceed that specified by U.S. regulation (Appendix K); some decrease in linear heat rate may be necessary to conform to U.S. LOCA criteria.

In summary, Candu reactors are designed to acceptance criteria which are, in places, much more stringent than U.S. requirements, and in places somewhat less stringent. To the extent that Canadian acceptance criteria are more stringent than those employed in the U.S., there is a high level of assurance that those aspects of Candu design which conform to these criteria will be found acceptable under U.S. licensing. For those areas where the Candu reactor has been designed to acceptance criteria which are viewed to be less stringent than those employed in the U.S., a more detailed quantitative safety analysis is required in order to definitively assess whether the Candu system conforms to U.S. licensing criteria or to identify design changes which may be necessary to provide such conformance.

3.5.4 Safety and Licensing Implications of Design Changes

Candu-PHWR plants, such as those currently in operation or under construction in Canada, have been subject to extensive safety evaluations to assure that established safety criteria are met.

Although differences between Canadian and U.S. licensing criteria may necessitate some modification to HWR design, the licensing review of the Candu concept performed in this study and the highly successful performance of operating Candu units lends support to the contention that such systems will be found licensable under U.S. regulations. However, 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 MWe, the use of slightly-enriched (1.2 w/o) uranium fuel, an increase in primary coolant pressure to 2250 psi, a reduction in D_2O reflector thickness, and an increase in moderator temperature. Of concern is the possibility that, as a result of these modifications, the consequences of various transients and postulated accidents may no longer prove satisfactory, thereby compromising licensability of the conceptual HWR design.

Although detailed design evaluations will be necessary before each of these modifications can be judged licensable on a commercial HWR, the potential safety and licensing implications of these design changes were reviewed to ensure that they did not introduce fundamental questions of licensability, and to establish the R&D requirements necessary for licensing. The safety and licensing implications of the various design changes which have been incorporated into the conceptual HWR design to reduce U_3O_8 requirements and/or plant capital costs are reviewed briefly below.

- ° Large Plant Size

The conceptual HWR is predicated upon a plant having a net electrical output of 1260 MWe. The largest Candu plants currently in operation are the 745 MWe units of the Bruce Station; 850 MWe plants are also under construction at the Darlington Station.

From the standpoint of the reactor itself, it is anticipated that the increase in electrical output to 1260 MWe can be

readily attained. In the pressure tube HWR, the increase in capacity is obtained by adding additional pressure tubes (fuel channels); 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. From the standpoint of overall station design, the increase in net electrical capacity could most easily be accomplished by the addition of a third coolant loop (for example, the addition of a third coolant loop of equal size to those employed in the two-loop Darlington Station would increase net electrical output from 850 MWe to 1275 MWe). With this option, components, such as steam generators and main coolant pumps, of standard size and proven performance, could be utilized. A somewhat different approach was adopted in the conceptual HWR design. A two-loop configuration, having somewhat larger components, was employed, since this configuration was found to minimize plant capital costs.

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, Candu reactors are provided with an automatic system (via the in-core instrumentation, plant computer, and zonal control system) to monitor and adjust local power distributions. Although increasing plant capacity to 1260 MWe may place additional demands on the design of the control system, it should introduce no fundamentally new considerations to control system design. The use of slightly enriched fueling will also increase stability. The use of larger size components is not expected to significantly impact the loss-of-coolant accident. This conclusion is based upon Canadian analyses which indicate that the limiting

size break for LOCA (with respect to peak clad temperature) is not the large break, but rather a break about 30% of the largest postulated break area. The more rapid rate of coolant discharge during large breaks has been considered in containment design. Consequently, the use of larger primary system components for which larger break areas can be postulated should not adversely affect the LOCA analysis.

- Slightly Enriched Uranium Fueling

The conceptual HWR design employs a 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, the most significant of which appear to be the effect of the higher discharge burnup on fuel integrity, changes in the coefficients of reactivity, and power peaking.

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 currently being obtained in natural uranium fueled Candu reactors, raising the question of whether fuel performance will remain satisfactory at these higher discharge exposures. Evaluations of fuel rod design, reported in Section 4.8, indicate that fuel of the present Candu design should be capable of achieving the discharge burnups anticipated for slightly-enriched fueling. Test irradiations to burnups approaching 20,000 MWd/t have also been performed in Canadian test reactors, lending further support to the supposition that fuel performance at these burnups will be acceptable. Nevertheless, 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 should be initiated in Canadian reactors as part of the development program to commercialize the conceptual HWR design.

The use of slightly-enriched fueling (or of thorium fueling) could change the coefficients of reactivity and hence the consequences of various postulated accidents. However, the evaluation performed in Section 5.5.8 found these coefficients to be comparable to those of natural uranium fueling, and hence no new safety and licensing issues are anticipated.

Power peaking will increase as a result of slightly-enriched uranium fueling, potentially reducing core thermal margins. This increase in core peaking has been accommodated by employing local enrichment zoning, by modifying the number of fuel assemblies replaced each fueling visit, and by increasing the number of fuel channels (i.e., of reducing the average power density so as to maintain equivalent peak power densities). Because of these design changes, 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.

° Increased Primary System Pressure

The conceptual HWR design employs a higher primary coolant pressure (2250 psia) than does the conventional Candu design (1600 psia). Primary temperatures are also increased. Of the design modifications incorporated into the conceptual HWR design, this increase in primary system pressure and temperature is thought to be the most significant, having the greatest potential impact on safety and licensability and necessitating significant R&D. Of concern is the effect of the change of primary coolant pressure and temperatures on pressure tube integrity, and upon the potential for pressure-tube-to-pressure-tube failure propagation during LOCA.

In order to accommodate the increase in primary system pressure and temperature, the thickness of the pressure tube has been increased by about 33%. The modified pressure tube wall thickness of the conceptual HWR was selected so as to maintain the same strength, or resistance to creep, as is present in the standard Candu design. Although this modification should be sufficient to accommodate the higher operating temperatures and pressures, it is recognized that the materials data base for irradiated Zr-Nb tubing at the temperatures and pressures of the conceptual HWR design is inadequate. Consequently, additional experimental information will be required to verify whether the selected pressure tube thickness does indeed provide comparable strength.

The possibility of pressure-tube-to-pressure-tube failure propagation is considered highly improbable 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 a thin-walled pressure tube, and since experiment has demonstrated that failure propagation will not occur even in the highly unlikely event of a catastrophic pressure tube rupture. 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 a detectable coolant leakage, rather than catastrophic pressure tube failure, thereby eliminating the mechanism which might be postulated to lead to failure propagation. Although the "leak-before-break" characteristics of the pressure tube should be sufficient to preclude the possibility of pressure tube failure propagation, it may be desirable to extend the experiments which have been performed to demonstrate that failure propagation will not occur to the primary system pressures and temperatures proposed for the conceptual HWR design.

The change in primary system operating temperatures and pressures might also affect the coefficients of reactivity, and hence change the consequences of various transients and postulated accidents. However, the effect of the higher primary temperature and pressure on the coefficients of reactivity were evaluated and found to have no adverse impact.

- ° Moderator Temperature

The moderator temperature of the conceptual HWR design was 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 design. No adverse safety implication of this change has been identified.

- ° D₂O Reflector Thickness

The conceptual HWR employs a D₂O reflector having about half the thickness of that employed in the standard Candu design, in order to reduce D₂O inventory. There is no obvious safety and licensing implication of this design change.

References

- 3.5-1 P. A. Ross-Ross, et al., "Performance and Reliability of Primary Circuit Components in Candu Reactors," AECL-5092.
- 3.5-2 P. A. Ross-Ross, "Experiments on the Consequence of Bursting Pressure Tubes in a Simulated Power Reactor Arrangement," AECL-4877.
- 3.5-3 F. Dallavalle, et al., "Explosive Rupture of a Power Channel Pressure Tube in a D₂O Moderated Reactor," 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, California, 15-19 Aug. 1977, Paper F8/2.

This section reviews briefly the considerations of commercial introduction which are discussed in greater detail in Chapter 6. In order for a concept to be commercially introduced, there must be a manufacturer willing to develop and market the concept, and a customer willing to purchase the marketed reactor design. There are significant risks associated with introducing a new concept: the possibility that capital costs will be higher than expected, that performance will not meet expectations, that significant technical problems will be encountered, that licensing problems and delays will occur, and that as a result a market sufficient to allow a manufacturer to recover investment in HWR design will not develop. Commercialization of the HWR in the U.S. thus necessitates that the perceived incentive for this concept be sufficiently high (as compared to the risks) to motivate utility consideration of this alternative and to provide reasonable assurances that the market for an HWR in the U.S. will be sufficiently large and long in duration to allow manufacturers to obtain a reasonable return on invested capital. The prospects for commercialization are consequently enhanced if the incentives for the HWR are large, or if the risks and uncertainties associated with introducing the HWR into the U.S. can be minimized.

At the present time, there does not appear to be a large incentive for utilities to adopt the HWR concept. The most attractive attribute of the HWR is its ability to utilize U_3O_8 more efficiently. However, while this improved uranium utilization may be attractive from a national policy standpoint, it is viewed to be of secondary importance with respect to motivating commercialization. Of more direct interest is the anticipated operational and economic performance of the HWR. Although the operational performance of operating Candu units has been excellent, the performance of the conceptual design is uncertain due to the design changes (slightly

enriched fueling/higher burnup, larger net electrical capacity, and modifications to improve net electrical efficiency) which were incorporated into the conceptual design in order to obtain capital costs more comparable to those of light water reactors. Expectations of improved performance with respect to the LWR are thus very speculative and hence unlikely to motivate adoption of this concept. Although also subject to numerous uncertainties, it does not appear that the conceptual HWR design will be economically competitive with light water reactors until the price of uranium rises to above 55\$/lb.

The economic evaluations thus indicate no current or near-term incentive for the HWR. 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 development of a new concept.

In view of the limited commercial incentives for HWR deployment, it appears clear that the HWR will not be commercially viable in the U.S. until the uncertainties which now cloud its acceptance -- uncertainties in the HWR plant capital cost, 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, performance, and reliability -- are significantly reduced. Consequently, it is concluded that the construction and operation of a large U.S.-sited demonstration plant is a prerequisite to commercialization. Such a demonstration plant would not necessarily cause commercialization to occur. It would, however, reduce many of the uncertainties associated with licensing, plant capital cost, and plant performance, thereby allowing a more informed assessment

of the incentives for HWR commercialization. Some further government involvement, however, may still be required (i.e., a pre-commercial deployment period) until uranium prices rise sufficiently to provide a commercial incentive for HWR deployment.

A number of possible introduction scenarios are discussed in Chapter 6. These scenarios assume that the HWR is introduced into the U.S. either under Canadian license or as a joint U.S./Canadian construction venture. Under this assumption, the experimental and analytical data base developed for the Candu-PWR would be available, as would be critical manufacturing capability for those plant components which may not be available in the U.S. Only a limited amount of attention has been devoted to the possibility of introducing the HWR independently of the Canadian program, since without U.S./Canadian cooperation the commercial introduction of the HWR would be delayed significantly, perhaps by 10 years or longer, to a point in time probably too late to be of real interest.

The most rapid path for demonstrating the HWR in the U.S. would be to construct, as the first plant in a sequence, a plant similar to Candu reactors currently in operation. This option would minimize the R&D requirements for this initial unit, and also 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. Although not strictly necessary, it would seem reasonable to employ slightly enriched uranium fueling in this initial unit, based upon the supposition that such fuel could be demonstrated by prior test irradiations in operating Canadian

plants and would not complicate licensability. Although the increase in station capacity from that of current Canadian designs is thought to be a straightforward engineering development problem, the lead plant might most prudently be of the 600 MW(e) size, both to avoid introducing additional questions of licensability and to minimize capital cost.

The introduction scenario for this case is illustrated in Figure 3.6-1. The time spans are based upon the assumption that it would require 10 years to license and construct a plant in the U.S. of established and accepted design such as the LWR. For the initial HWR unit, this time period has been increased by two years to allow additional time for U.S. licensing and construction. R&D activities are initiated two years prior to the beginning of this 12-year licensing and construction period, during which initial design activities and the preparation of a preliminary safety analysis report are prepared. The second unit in the sequence could be a larger plant incorporating all of the features of the conceptual HWR design. In Figure 3.6-1, construction of this second plant is not initiated until the initial demonstration unit has been in operation for a period of two years. This approach minimizes risk, since it allows the performance of the initial unit to be demonstrated prior to committing the second unit of the series. However, this deployment schedule can be accelerated somewhat by the early initiation of licensing reviews on the second unit (however, construction is not initiated until the first unit demonstrates successful performance, because of the high capital commitment), as illustrated in Figure 3.6-2. These accelerated deployment schedules are discussed in greater detail in Chapter 6.

Figure 3.6-3 illustrates an alternate approach which seeks to demonstrate both U.S. licensing and the technical viability of the modifications incorporated into the conceptual HWR design in

the first plant. Clearly the risks and uncertainties of adopting this approach are greater than for the scenarios discussed above since this demonstration plant now differs significantly 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. As illustrated in Figure 3.6-3, an additional two years has been added to the licensing and construction schedule, primarily to allow additional time for NRC review of design changes. The R&D period prior to submittal of the PSAR has also been increased by two years 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. The scenario illustrated in Figure 3.6-3 assumes that this initial unit operates for a period of two years to demonstrate successful performance before subsequent units are committed. Similar to the case previously discussed, some acceleration of this schedule can be obtained by committing to further units at an earlier date, should the incentives or urgency warrant the increased risk. Accelerated deployment schedules for this case are also discussed in Chapter 6.

Estimates of R&D costs are given in Table 3.6-1. These R&D costs are divided into two components: generic technology programs (which are independent of the deployment schedule selected) and the reactor plant demonstration program. For the latter, R&D costs have been developed for each of the two basic introduction scenarios discussed above. As Table 3.6-1 indicates, the majority of the costs are associated with the reactor plant demonstration. The initial plant demonstration costs for the first introduction scenario are lower than those for the alternate scenario primarily because of the assumption that a unit of smaller size (600 vs.

1260 MW(e)) is constructed; safety and licensing and first-of-a-kind engineering costs are also lower since this initial plant is assumed to be of basically similar design to operating Candu plants. Costs for the second unit, however, are higher with this scenario since the design and licensing development required for the conceptual HWR design must be performed. As noted previously, some funding may also be required for subsequent units, until such time as the HWR can successfully compete economically with light water reactors.

TABLE 3.6-1

RD&D COSTS (MILLIONS)

A. Generic Technology Programs

1. Base Research	3.0
2. Core and Fuel Development	6.5
3. Component Development	1.5
4. Safety and Generic Licensing	3.0
	<u>14.0</u>

B. Reactor Plant Demonstration

Scenario AScenario B1. Demonstration Plant¹⁾

Safety and Licensing	25	--
First-of-a-Kind Engineering	15	--
Plant Cost	<u>637</u>	--
	677	

2. Large Plant/Lead Plant²⁾

Safety and Licensing	30	50
First-of-a-Kind Engineering	35	40
Plant Cost	<u>910</u>	<u>910</u>
	975	1,000

3. First Commercial Plant²⁾

Safety and Licensing	0	5
First-of-a-Kind Engineering	0	5
Plant Cost	<u>910</u>	<u>910</u>
	910	920

1) 600 MWe, Similar to Candu Plants in Operation

2) 1260 MWe Conceptual HWR

Figure 3.6-1
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 3.6-2
CASE A.2 - GRADUAL INTRODUCTION OF DESIGN CHANGES
MODERATE RISK

Figure 3.6-3
CASE B.1 - RAPID INTRODUCTION OF DESIGN CHANGES
LOW RISK

(1) 1260 MW(e) CONCEPTUAL HWR

3.7 Conclusions and Recommendations

The primary conclusions and recommendations reached in this evaluation are summarized below. The reader is referred to the body of the text (Chapters 4 through 6) for a more detailed discussion of the basis for the various conclusions and recommendations.

3.7.1 Design Approach

- It was concluded that the conceptual HWR design should be based upon Candu pressurized heavy water reactor technology. This technology was selected based on the commercial status of the Candu concept, its proven performance, its inherently efficient uranium utilization, and the relative ease with which Candu plants can be expanded to the 1000 MW(e) size specified for this NASAP evaluation.
- Existing Candu designs and fueling practices do not fully meet the objectives of NASAP. It was concluded that design modifications should be entertained to further improve uranium utilization and/or power costs economics.

3.7.2 Optimization of HWR Design

3.7.2.1 Slightly Enriched Uranium Fueling

- The conceptual HWR design should be predicated upon the use of slightly enriched uranium fueling. The use of slightly enriched fueling reduces uranium requirements, results in a reduction in fuel cycle costs, and makes practical a number of modifications to the plant design which serve to reduce net capital cost.

- U_3O_8 requirements can be reduced to about 65% of that of the natural uranium fueled HWR by employing slightly enriched uranium fuel. An initial enrichment of 1.2 w/o appears to be near optimal from a uranium resource and fuel cycle economics standpoint. Limiting the initial enrichment to 1.2 w/o or less is desirable, since it minimizes the likelihood that fuel rod design modifications will be required. Limiting the initial enrichment to 1.2 w/o also minimizes the increase in core power peaking.
- The fuel rod design assessment indicates that the higher (20,000 MWd/t) discharge burnup concomitant with an initial enrichment of 1.2 w/o can probably be accommodated in Candu fuel of existing design.
- The use of slightly enriched fueling increases core peaking factors. Local enrichment zoning of the HWR fuel assembly is an attractive method for reducing this power peaking. Local enrichment zoning can decrease the peak power density observed in a uniformly enriched 1.2 w/o fuel assembly by about 8%, or to a value approximately 4% lower than that observed for the natural uranium fueled assembly; local enrichment zoning has a negligible effect on achievable cycle length. Core power peaking can also be reduced by fuel management procedures which replace a smaller number of fresh fuel assemblies at each refueling visit.
- It is recommended that more detailed fuel management calculations be performed in order to more precisely evaluate the increase in core peaking anticipated with slightly enriched uranium fueling. Explicit three-dimensional calculations are required in which each fuel channel and fuel assembly are represented.

Prior to a decision to commercially deploy slightly enriched fueling in HWRs, this fueling mode should be demonstrated by irradiation programs in existing Candu test reactors to demonstrate that acceptable fuel performance can be obtained at the higher discharge exposures implicit in slightly enriched fueling and to establish definitively whether or not design changes to the current Candu assembly are required.

3.7.2.2 Primary System Pressure Evaluation

- The lower net electrical efficiency of the standard Candu reactor is one of the major reasons why its capital cost is evaluated to be higher than that of the LWR. Part of the lower net electrical efficiency of the HWR is attributable to energy losses to the moderator and part to lower primary and secondary pressures. If primary and secondary pressures are increased to those of the PWR, HWR efficiency increases to 31.3%, as compared to 29.3% for the standard Candu reactor. This 6.8% improvement in net electrical efficiency is reflected in a similar reduction in effective plant capital cost.
- In order to accommodate the higher primary system temperatures and pressure, the pressure tube thickness must be increased. The increase in pressure tube thickness results in a decrease in achievable burnup. For natural uranium fueling, this decrease in achievable burnup is large, decreasing the discharge burnup to about 6,300 MWd/t, and results in about a 10% increase in uranium requirements. However, for slightly-enriched fueling, the decrease in achievable burnup becomes a small fraction of the total cycle length and is more than compensated by the increase in electrical efficiency; as a result, uranium utilization is improved slightly.

- Review of pressure tube design considerations associated with the use of higher coolant temperatures and pressures indicates a likelihood that the Zr-Nb pressure tubes can operate successfully under these conditions. However, it is recognized that increased primary system pressure represents a significant departure from present Candu design, which will require pressure tube development and demonstration prior to licensing.

3.7.2.3 Lattice Spacing Evaluation

It was concluded that there appears to be little incentive for reducing lattice pitch. Reductions in lattice pitch result in a significant decrease in achievable cycle length. This reduction in achievable cycle length results in a large degradation of resource utilization which adversely impacts fuel cycle cost and largely negates the benefits of reduced D_2O inventory. Reducing lattice spacing also introduces mechanical problems in end shield and refueling machine design.

3.7.2.4 Reflector Thickness Evaluation

- Reducing reflector thickness results in a modest (0.25 mill/kw-hr) reduction in power costs, due to the reduction in D_2O inventory. The reduction in D_2O reflector thickness has little impact on the achievable cycle length for slightly enriched fuel and appears to introduce no unusual mechanical design problems, provided the reflector thickness is not reduced below about 31 centimeters. Consequently, it was concluded that a reduced reflector thickness should be included in the conceptual HWR design.

3.7.2.5 Primary System Arrangement Evaluation

- The use of a two-loop primary system arrangement, containing four steam generators and four primary coolant pumps (2 per loop) results in a significantly lower capital cost than a three-loop configuration. Savings result partially from the use of larger components (steam generators and pumps) and partially because the two-loop configuration allows a smaller containment diameter.
- Although the size of the steam generators and primary pumps required for the two-loop configuration are larger than components currently in use in operating Candu plants or in plants under construction, extensive component R&D should not be required. The steam generators, though larger than those in operation on Candu plants, are smaller than units in operation on pressurized light water reactors. The primary system pumps, which are larger than any currently in operation, would require a modest development program.
- The consequences of a loss-of-coolant accident remain acceptable with the two-loop primary system arrangement.

3.7.2.6 Moderator Temperature Evaluation

- The cost of the moderator heat exchange system can be reduced by increasing the operating temperature of the moderator slightly to provide a larger temperature differential between the moderator and the mechanical draft cooling tower heat sink. Increasing moderator temperature to 93°C results in only an insignificant decrease in achievable burnup and hence fuel cycle costs.

3.7.3 Plant Design and Capital Cost

- ° It was concluded that the increase in net generating capacity from the 600-750 MWe of current operating Candu units to the 1260 MWe of the conceptual HWR design can be accomplished in a straightforward fashion, i.e., can be accomplished in the course of normal engineering design development without necessitating extensive R&D efforts. Capacity of the reactor core itself is increased by simply increasing the number of fuel channels present in the reactor; unlike the PWR, the development of massive pressure vessels of larger size is not required.
- ° Increasing the capacity of the nuclear steam supply system can most easily be accomplished by the addition of a third coolant loop equal in size to the loops currently employed in the standardized Candu-600 design or the 800 MW(e) Darlington design. This alternative would make greatest use of the components (primarily pumps and steam generators) already developed.
- ° However, the alternate approach of increasing loop size to retain the two-loop configuration of operating Candu units was employed in the conceptual HWR design. This alternative was selected because of capital cost savings which were found to result from the use of a fewer number of larger components and from the smaller containment structure made possible by the two-loop configuration. The fact that some modification to the pump and steam generator design would have been required at any rate in order to accommodate higher operating pressures and temperatures of the conceptual design also weighed in this decision. Although the larger components of the two-loop configuration is a larger departure from proven Candu design than would have been necessary had the three-loop design been selected, extensive component R&D should not be required.

- ° It was concluded that the balance of plant design should be as similar as possible to those of PWRs, rather than attempting to replicate the features of operating Candu units. A U.S. type containment structure and balance-of-plant design was selected because of the belief the U.S. architect engineers would elect to utilize the more familiar PWR concepts in constructing a U.S.-sited HWR. Also, questions of the acceptability of the Candu design under U.S. licensing criteria frequently center on these engineered features of the plant, and consequently the use of a U.S.-type containment and balance-of-plant design simplifies the licensing evaluation and allows attention to be focused on the more fundamental characteristics of the HWR plant. The estimate of plant capital cost is also simplified by this design approach.
- ° The estimated capital cost of the conceptual HWR design is \$587/kWe, exclusive of heavy water. This plant capital cost is in January 1, 1978 U.S. dollars, and is exclusive of interest during construction, escalation during construction, contingency allowance and owner's cost. The capital cost of the conceptual HWR is approximately 8.2% higher than the capital cost of the LWR. When the cost of the initial inventory of D₂O is included, the conceptual HWR has a capital cost about 34% higher than that of the LWR.
- ° The 8.2% higher capital cost of the conceptual HWR designed to the same licensing criteria as the LWR appears reasonable in view of the lower net electrical efficiency of the HWR (31.3 vs. 33% for the LWR) which necessitates a physically larger plant to generate an equivalent electrical output. HWR costs are also larger because of such extra equipment as the refueling machines, D₂O upgrader, and moderator cooling system. The cost differences between the HWR and LWR are primarily in the nuclear steam supply system and related equipment.

- The capital cost of the conceptual HWR is considerably lower than previous estimates (Reference 3.7-1) for a U.S.-sited HWR, which indicated capital costs approximately 17% higher than those of the LWR. The reduced capital costs of the conceptual HWR are due primarily to increased electrical efficiency (resulting from increasing primary and secondary system temperatures and pressures) and the use of the two-loop configuration.
- Although the HWR capital cost developed in this study is felt to be reasonably accurate, it must nevertheless be recognized that design changes which could significantly impact capital cost may be required as a result of more detailed assessments of safety and licensing against U.S. design criteria and regulations. Consequently, it is recommended that a more detailed and quantitative evaluation of safety and licensing issues be performed to establish the effect of potential licensing issues on plant capital cost.

3.7.4 Uranium Utilization and Economic Performance

3.7.4.1 Slightly-Enriched Uranium Once-Through Cycle

- An initial enrichment of 1.2 w/o was found to be near optimal from the standpoint of both uranium utilization and fuel cycle costs. Limiting the initial enrichment of 1.2 w/o minimizes fuel performance and core peaking problems.
- Thirty-year cumulative U_3O_8 requirements for the conceptual HWR enriched to 1.2 w/o are 3,550 ST/GWe, as compared to U_3O_8 requirements of about 5,000 and 6,000 ST/GWe for natural uranium fueling in the HWR and for the standard LWR, respectively. Thus, U_3O_8 requirements for the slightly-enriched HWR are about 30% lower than those of the natural uranium-fueled

HWR, or about 40% lower than those of the LWR; alternately, about 40% more energy could be generated from a given resource base if the slightly-enriched HWR was employed in place of the natural uranium fueled HWR, or about 70% additional energy if the slightly-enriched HWR was used instead of the standard LWR.

- Thirty-year cumulative separative work requirements for the slightly-enriched HWR are about 927 MT/GWe, about 25% of those of the standard LWR.
- It is imprudent to draw definitive conclusions as to the economic competitiveness of the conceptual HWR with respect to the LWR, since these conclusions are affected significantly by the uncertainties in plant capital cost and reactor availability, and by the uncertainties in the projection of economic parameters into the future. However, it can be concluded that HWR power costs are less sensitive to the price of U_3O_8 (due to the lower U-238 requirements of the conceptual HWR) and that consequently the HWR will eventually achieve lower power costs as the price of U_3O_8 increases.
- Under the reference economic assumptions employed in the power cost analysis, the slightly-enriched conceptual HWR does not become economically competitive with the once-through fuel cycle in the standard LWR until the price of uranium rises to about 55\$/lb. This breakeven U_3O_8 price, however, is very sensitive to a number of economic parameters. It is most sensitive to the capital cost and capacity factor differentials assumed between the HWR and LWR. It is also significantly affected by the cost of heavy water and by the degree to which improvements can be incorporated into the LWR to reduce LWR uranium requirements and fuel cycle costs.

- ° Slightly-enriched once-through fueling is more economic than other fueling modes in the HWR for uranium prices of up to 100\$/lb or higher.
- ° Batch fueling of the HWR, in place of the on-line refueling employed in operating Candu units, is unattractive. With an annual refueling interval, the uranium requirements of the slightly-enriched HWR are increased by about 30%, while annual refueling of the natural uranium HWR is impossible. If semiannual refueling is employed, U_3O_8 requirements are still increased by about 16% for the slightly-enriched case, while uranium requirements for the natural uranium fueled system are nearly doubled.

3.7.4.2 Plutonium Recycle

- ° Plutonium recycle in the conceptual 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 fuel cycle.
- ° Although not evaluated in detail, it does not appear that design modifications to the conceptual HWR will be necessary to operate in the plutonium recycle fueling mode. However, more detailed evaluations of this question should be performed if operation on this fuel cycle is contemplated.
- ° The self-generated plutonium recycle mode evaluated in this report does not appear to be economically attractive because of the high fuel cycle cost contributions for 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 would become more economic than slightly-enriched once-through fueling. Improved fuel cycle economies are anticipated, however, for recycle modes employing high plutonium enrichments and increased discharge exposures.

3.7.4.3 Denatured U-235/Th Cycle

- ° Significant reductions in U_3O_8 requirements can 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 fueling mode. Separative work requirements are about twice those of once-through fueling.
- ° Annual equilibrium U_3O_8 requirements are dependent upon the discharge burnup employed, varying between 42 and 53 ST/GWe as discharge burnup is varied between 13.7 and 37 MWd/kg (compared to an annual U_3O_8 requirement of 114 ST/GWe for slightly-enriched once-through fueling). Thirty-year cumulative U_3O_8 requirements are rather insensitive to discharge burnup, with higher ex-core inventory requirements compensating for the lower annual requirements as discharged burnup is decreased.
- ° The economic performance of the denatured U-235/Th cycle is very sensitive to discharge exposure. For low discharge exposures, this 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 as the burnup is increased, since the lower annual throughput of fuel results in decreased fabrication and reprocessing costs.
- ° Under the reference fuel cycle economic assumptions, the denatured U-235/Th fuel cycle employing a discharge burnup of 37,000 MWd/t does not become economically competitive with the slightly-enriched once-through cycle until the price of uranium rises to about 180\$/lb.

- It is concluded 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.
- The use of thorium fueling may place increased demand 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 and of the reactor operational performance with thorium fueling be undertaken.

3.7.4.4 Denatured U-233/Thorium Fuel Cycle

- Because of the inherent neutron economy of the HWR concept, makeup U-233 requirements for the denatured U-233/Th cycle are also considerably lower than the U-233 requirements of the LWR. Annual U-233 makeup requirements of the conceptual HWR depend upon the discharge burnup employed, varying between about 80 and 140 kg/GWe compared to the 220 kg/GWe annual requirements of the LWR. As was observed in the denatured U-235/Th cycle, 30-year cumulative U-233 requirements prove insensitive to discharge exposure (for discharge exposures greater than about 10,000 MWd/t, because of the compensating ex-core inventory requirements).
- Self-sufficient operation, in which no makeup U-233 feed is required for the equilibrium cycles of operation, is not possible with denatured thorium fueling. This mode of operation is possible only when pure U-233 (i.e., without U-238 additions) is used to enrich thorium.
- It appears unlikely that the denatured U-233/Th fuel cycle will prove economically competitive with once-through fueling

unless high discharge exposures are employed. At a discharge exposure of 14,000 MWd/t, fuel cycle costs are about 1.5 mills/kw-hr greater than that of the once-through uranium cycle (100\$/lb U_3O_8), even assuming a rather low, 10\$/gm, U-233 price.

- ° As was the case for the denatured U-235/Th fuel cycle, it is concluded that denatured U-233/Th fueling is unlikely to prove economic unless it is possible to develop less expensive refabrication and/or reprocessing methods.

3.7.4.5 Pu/Th Cycle

- ° Plutonium is produced in both the denatured U-235 and U-233/Th fuel cycles from the U-238 denaturant (the quantity of plutonium produced is about 10% of that of the slightly-enriched fueling mode). Uranium utilization can be improved slightly (by about 4%) by using this plutonium to enrich thorium; the U-233 bred in this Pu/Th cycle can then be denatured and used to supply part of the U-233 required for the denatured U-233/Th cycle.
- ° The use of fissile plutonium in the Pu/Th cycle appears to be more economically attractive than the recycle of plutonium into uranium fuel. The Pu-Th fuel cycle appears economically competitive with once-through fueling if the price of uranium is greater than 75-105\$/lb, depending upon the cost of reprocessing and mixed-oxide fuel fabrication.

3.7.5 Safety, Licensing and Environmental Impact

- ° An assessment of the safety and licensing of the conceptual HWR design can be divided into considerations of the basic HWR concept and of the modifications to this concept incorporated to improve uranium utilization and/or economic performance. The safety and licensability of the basic HWR concept can in turn be divided into considerations of the engineered features and of the 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, it was concluded that it would be more appropriate to replace these engineered features with PWR type designs since it is expected that U.S. architect engineers would elect to employ the more familiar PWR type designs in a U.S.-sited plant. 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. Consequently, there is little question that these engineered features are acceptable under U.S. licensing criteria, or at least can be made to be so as a result of detailed engineering design.
- ° 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. It is concluded that, on balance, the heavy water reactor concept provides an equivalent degree of protection to the health and safety of the public as do other alternatives.

- ° The prime inherent attributes of the pressurized heavy water reactor which contribute 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 system provided in the HWR. The inherent attributes of the HWR which might be viewed as being less satisfactory than those of the LWR are the use of pressure tubes as the primary pressure boundary and the positive void coefficient of reactivity.
- ° 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, a qualitative assessment was performed by comparing Canadian licensing and acceptance criteria to those employed in the U.S. This evaluation identified two areas where U.S. acceptance criteria appear more restrictive than those employed in Canada, and consequently which might necessitate minor design changes (e.g., a slightly lower power density) to meet U.S. acceptance criteria. In particular, under Canadian licensing a time at temperature criterion is employed to determine the acceptability of clad temperatures during the loss-of-coolant transient; this criterion allows peak clad temperatures to exceed the limits specified by U.S. regulations (Appendix K) for brief periods of time. Some decrease in peak linear heat rate may be necessary to conform to U.S. licensing criteria. U.S. licensing criteria also expressly precludes clad damage during moderate frequency faults and places limitations on clad damage during infrequent faults. Canadian licensing imposes no explicit limits on clad damage. The imposition of fuel failure criteria on the

HWR may necessitate a reduction in core power density. Neither of these differences is considered to be fundamental, since they can be accommodated by slightly increasing the number of fuel channels, should more detailed evaluations indicate that this is actually required. In other respects, Canadian design and acceptance criteria are viewed to be comparable or more conservative than those employed in the U.S.

- ° 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 should be performed to demonstrate this supposition. The primary operational constraint placed on the HWR is the necessity of returning to power within about 30 minutes after a reactor trip. If the reactor is not returned to power within this period, it becomes necessary to wait 36 (natural uranium fueling) or 30 (slightly enriched fueling) hours (for xenon decay) before a return to criticality is possible.
- ° 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, a quantitative analysis should be performed to demonstrate the adequacy of the HWR design for U.S. seismic conditions.
- ° Because of HWR is cooled and moderated with heavy water, production of radioactive tritium is much higher than in the LWR. Although careful attention is paid in the design of HWRs to minimize D₂O/tritium losses and the measured releases of tritium from operating Candu plants have been quite low,

only a fraction of the allowable release rate, tritium releases will be significantly greater (by a factor of about 10) than those from LWRs. Although tritium releases are considered to be the most significant difference in environmental impact between the HWR and LWR concepts, it is expected that HWR releases will be judged acceptable. Also, tritium removal has been demonstrated on a small scale, and it appears that this technology could be applied, if required, to the HWR for tritium level control. As a result of the lower efficiency of the conceptual HWR (31.3% vs. 33-34% of the PWR), 5 to 8% more heat will also be rejected to the atmosphere. Because of the pressure-tube design and on-line refueling feature of the HWR, failed fuel can be more easily detected and removed from the reactor, thus reducing the potential for release of radionuclides to the environment.

- ° In arriving at the conceptual HWR design, a number of changes to the standard Candu plant have been made in order to improve uranium utilization or to reduce plant capital cost. An assessment of the effect of these design changes on safety-related core parameters was performed with the conclusion that core transient characteristics are not significantly changed and that the consequences of various postulated accidents should remain acceptable.
- ° The design change which engenders the most significant safety and licensing questions is the increase in primary system pressures and temperatures. In order to accommodate these increases, 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. However, it is recognized that the data base on pressure tube material performance at the postulated pressures and temperatures of the conceptual

design are inadequate, and consequently R&D to determine pressure tube material strength and corrosion resistance will be necessary.

- ° A second major departure from operating Candu reactor experience is the use of slightly-enriched fuel with a concomitant increase in discharge exposure. Although the design evaluation indicated that the performance of Candu-type fuel at these extended burnups should be acceptable, irradiation demonstration programs would be desirable. Such irradiation programs could probably be performed in existing Candu reactors.
- ° An evaluation of the safety-related core parameters for denatured thorium fueling indicates that these parameters are comparable to those obtained for natural uranium fueling. Consequently, it appears that denatured thorium fueling could be employed without engendering fundamental questions on safety and licensability. However, the use of thorium fueling may place increased demands on the reactor control system and may alter the transient performance characteristics. Consequently, it is recommended that a more detailed evaluation of the transient and safety and licensing implications of thorium fueling be performed.
- ° The Canadian experience with the construction and operation of Candu plants and the brief licensing review formed as part of this evaluation provides a favorable outlook on the eventual acceptance under U.S. licensing. However, there are a number of aspects, or potential licensing issues, that are likely to require more detailed evaluation and/or the development of a suitable data base prior to the licensing of plants based upon the conceptual HWR design in the U.S. These areas are summarized 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.

2. R&D programs are required 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.
3. The probability and consequences of pressure tube failure and of the potential for damage propagation to adjacent tubes is likely to become a licensing issue, particularly as it relates to the higher system pressure of the conceptual design.

Tests to demonstrate the low probability of pressure-tube failure propagation at the elevated pressures of the conceptual design are desirable.

4. 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 fuel results in excessive core peaking and to establish the transient and control characteristics of the larger core of the conceptual HWR design.
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.

7. More detailed fuel management and core performance analyses should be performed for denatured thorium fueling.
8. A seismic evaluation should be performed to demonstrate the suitability of the HWR design under U.S. geology and seismic requirements.
9. 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.
10. A more extensive evaluation should be made of the implications of the positive coolant void coefficient of reactivity on transient undercooling and underpressurization events. Although from a technical standpoint this positive coefficient of reactivity can be accommodated within the design, the implications of this positive coefficient should be thoroughly evaluated and discussed at an early date with licensing authorities.
11. In order to obtain a more complete licensing review of the conceptual HWR design, a preliminary safety information document, closely paralleling the contents of the PSAR, should be developed and pursued through discussions with licensing authorities.

3.7.6 Commercial Introduction

- Although commercialization of a wholly U.S. developed HWR might be considered, it is concluded that the only practical approach to commercialization is through the deployment of Candu-type reactors with Canadian cooperation or license.

The independent development of a new HWR concept would defer commercialization too far into the future to be a viable alternative.

- One or more U.S.-sited demonstration units will be required prior to commercialization. Demonstration units are required to establish plant performance and capital cost in the U.S., and to demonstrate U.S. licensability.
- The most rapid path for introducing the HWR into the U.S. would be to deploy as the lead demonstration unit a reactor as similar in design as possible to those currently in operation in Canada. This approach would allow the earliest introduction of an HWR by making greatest use of developed and proven technology, so as to provide an early indication of plant cost and licensability. This plant might be predicated upon slightly-enriched fueling, since this mode of fueling could be demonstrated during the construction period in operating Candu plants. R&D on such potential design modifications as higher primary system temperature and pressure could be performed simultaneously for implementation on subsequent demonstration units. It is estimated that such a demonstration unit could begin operation approximately 14 years after commitment, under the assumption of a two-year R&D and PSAR preparation period, a five-year period for PSAR review, and a seven-year construction schedule. Subsequent units might be initiated after PSAR approval, as construction progresses, or after the reactor has demonstrated successful operation for several years, depending upon perceived commercial incentives and urgency for rapid deployment.

- An alternate approach would be to attempt to license and demonstrate the conceptual HWR design, including all design modifications, in the initial lead plant. Clearly the risks and uncertainties in adopting this approach are greater, but this alternative would demonstrate more rapidly the licensability and performance of the reference conceptual design upon which the assessments of commercial viability are based. It is estimated that the lead unit would come into operation two to four years later than the case previously mentioned, because of additional time required for R&D and licensing review.
- The conceptual HWR is not likely to be introduced in the U.S. today. While the conceptual HWR meets the goals and objectives of NASAP, as viewed from a U.S. utility viewpoint there is little difference between the economics of present light water reactor plants and the proposed reactor. The risks inherent in introducing a new reactor type (including one that is commercially successful elsewhere) to their utilities systems suggests that commercial introduction would have to be preceded by a demonstration project.
- 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. Strong government support, however, will be required if the HWR is to be developed and demonstrated in the near term before uranium prices have risen sufficiently to warrant commercial deployment.

- In view of the highly satisfactory operation of Candu plants on the natural uranium cycle, the major technical problem areas which might influence prospects for commercialization are considered to be the changes incorporated into the conceptual design to improve uranium utilization and power cost economics. In particular, development and demonstration of slightly enriched fueling and of pressure tubes capable of withstanding the elevated temperatures and pressures are required to obtain the economic performance upon which commercialization of the HWR in the U.S. has been predicated.
- No technical problem has been identified which should preclude licensability of the conceptual HWR design in the U.S. However, questions of licensability and of the impact of licensing review on plant design and capital cost are considered to be one of the major barriers to commercialization. Thus, uncertainties associated with plant licensability must be resolved through the licensing review of a demonstration plant before the HWR can be considered a viable commercial alternative in the U.S.
- There are a number of possible scenarios for the introduction of the HWR, depending upon the availability of Canadian technology and upon whether large or small financial risks are thought to be warranted, based on the incentives for HWR deployment. Assuming a joint U.S.-Canadian program and incentives sufficiently high to warrant significant financial risk, the first commercial HWR unit could begin operation as early as 1997. However, scenarios which attempt to minimize risk would defer the operation of the first commercial unit to about 2005. 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 longer.

REFERENCES

- 3.7-1 C00-2427-13, "Capital Cost, Pressurized Heavy Water Reactor Plant," United Engineers and Constructors, Inc., June 1977.