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HEAVY WATER REACTORS

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INTRODUCTION

The International Fuel Cycle Evaluation (INFCE) is directed toward the identification of nuclear fuel cycle options which pose inherently low risk of nuclear weapons proliferation while retaining the major benefits of nuclear energy. In order to limit the availability of nuclear materials from which weapons can be constructed, it is desirable to fuel the reactors located throughout the world with nuclear fuels which are inherently unsuitable for use in weapons--either low enriched uranium fuel or Uranium-233 isotopically denatured with Uranium-238. At the same time, it is recognized that the benefits of nuclear power are limited by the size of the uranium resource base and by the efficiency with which that base can be utilized when the once-through uranium fuel cycle is employed, or by the efficiency with which U-233 can be generated and utilized when denatured U-233 fuel is utilized.

In this context, the Pressurized Heavy Water Reactor (PHWR) is worthy of special attention because of a unique combination of characteristics. The pressure-tube version of the PHWR with on-power refueling of natural uranium in a once-through cycle, known as the Candu-PHW, 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. Because of the superior neutron economy inherent in the use of heavy water for both reactor moderator and coolant and inherent in the use of on-power refueling, the PHWR can, in principle, surpass all other reactors in energy production per unit of U_3O_8 and separative work when the once-through uranium fuel cycle is employed. The requirements for U-235 or U-233 are likewise minimized on the denatured Uranium-Thorium cycle because of the inherent neutron economy of the PHWR.

This paper presents preliminary technical and economic data to INFCE on the denatured U-233/Thorium fuel cycle for use in early comparisons of alternate nuclear systems. The once-through uranium fuel cycle is discussed in a companion paper. In presenting this preliminary information at this time, it is recognized that there are several other denatured thorium fuel cycles of potential interest, such as the U-235/thorium cycle which could be implemented at an earlier date. Information on these alternate cycles is currently being developed, and will be provided to INFCE when available.

Although a number of alternate heavy water reactor concepts, such as the Steam Generating Heavy Water Reactor (SGHWR) which employs light water as coolant or the heavy water cooled and moderated pressure vessel reactor (ATUCHA), could well have been considered for evaluation, this paper addresses heavy water reactors of the Candu-PHWR type. The reasons for selecting this concept for 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. Due to the use of heavy water both as coolant and moderator, reactors of the CANDU type exhibit improved resource utilization compared to the SGHWR because of the use of light water as a coolant. The resource utilization of CANDU and ATUCHA type reactors are comparable, but CANDUs are more widely deployed and are not limited in generating capacity, as are ATUCHA type reactors, by pressure vessel size. Although one objective of this evaluation has been to minimize the changes to existing Candu-PHW designs so as to enhance the prospects for early deployment and to minimize R&D requirements, several modifications to the existing paper, in order to enhance the prospects for economic competition with the LWR. The conceptual design utilizes an increased primary system pressure so as to increase system efficiency and reduce the effective capital cost of the plant. Tne design of this conceptual HWR, its fuel cycle and economic performance, technological status, safety and accident considerations, and environmental information are discussed in subsequent sections of this paper.

I. SYSTEM DESIGN AND PERFORMANCE DATA

A. Nuclear Steam Supply System

The conceptual Heavy Water Reactor (HWR) design employed in the evaluation of the denatured thorium fuel cycles is identical to that developed for the once-through uranium cycle. As discussed in the companion paper on the once-through cycle, this conceptual reactor differs from the Canadian Candu 600 design primarily because of the increase in station size to 3800 Mw(th), 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 conceptual 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 I-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 shields. 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 coolant, which is operated at a pressure of 2200 psi, 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 and 10.3 cm in inside diameter. The conceptual HWR plant is designed to operate at a somewhat higher primary system pressure than the Candu 600 (2200 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 5.59 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 diagrammed schematically in Figure I-2. 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 I-2 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.

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 initial 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.

In developing the reference conceptual NSSS design described in the preceding paragraphs, no attempt has been made to optimize the design for thorium fueling. Rather, the conceptual design has been optimized for the once-through cycle employing slightly enriched uranium fuel, and the evaluation limited to assessing the performance of denatured thorium fuels in this design. This approach has the advantage of deliniating the performance of thorium fuels in a HWR design which might be initially deployed at an earlier date using uranium fueling. However, it is recognized that in the longer term improved performance on the denatured thorium cycles might be obtained by optimizing the HWR explicitly for thorium fueling.

B. Balance of Plant

A plot plan for the conceptual HWR plant is shown in Figure I-3. A diagram of the reactor containment and auxiliary structures are shown in Figure I-4. The plot plan shown in Figure I-3 is arranged for a single HWR unit but contains provisions for incorporating a second plant at the same site. Mechanical draft evaporated cooling towers are provided to reject heat both from the condenser and from the moderator heat exchanger. The reactor containment building is a prestressed concrete structure with a steel liner which houses the reactor and primary heat transport system. In contrast to multi-unit Canadian designs which employ a separate vacuum building, the containment building for the conceptual design follows U.S. construction practices and is designed to withstand the differential in pressures encountered during postulated loss of coolant or steam-line break accidents. A vertical cross section of the containment building is shown in Figure I-5. The reactor control building is a multi-storied steel beam and concrete structure which houses the reactor control room electronic cabinets, emergency diesel generators, and other equipment closely associated with actual plant operation. The control building is entered through a security building which controls the access to the reactor and sensitive equipment.

The turbine generator, located in a turbine hall of reinforced concrete, is of conventional design. Also contained in this hall are condensers, main steam piping, and feed water heaters, feed water pumps, electrical switch gear, and water treatment plant.

In developing the reference balance of plant design, both two- and three-loop configurations were considered. The three-loop configuration has the advantage of employing similar size components as is currently utilized in the Candu-600 plant. However, the two-loop configuration results in a smaller and less costly containment building and results in economy of scale in the larger size components. Because of the lower capital cost of the two-loop configuration, this option was employed in the conceptual design. Efforts were also made to reduce the number of buildings, as reflected in the plot plan shown in Figure I-3, to further reduce plant capital cost.

C. General Reactor Performance Specifications

General reactor performance characteristics are specified in Table I. The conceptual HWR plant is designed to produce a reactor thermal power of 3800 Mw(t) (coolant power) consistent with the maximum power level currently allowed in the United States. This increase in thermal power was accomplished by increasing the number of coolant channels from the 380 employed in the Candu-600 design to 740. The estimated thermal efficiency of the conceptual design is 31.5%, corresponding to gross and net electrical power ratings of 1343 and 1260 Mw(e) (tentative, subject to detailed evaluations of electrical efficiency and house load). This increase in overall efficiency, compared to the Candu-600 design which has an efficiency of 29%, was achieved by increasing secondary system pressure (from 680 to 1100 psia). The reactor core volume, dimensions and coolant flow rate are shown in Table I-1.

II. FUEL MANAGEMENT AND HANDLING INFORMATIC.

The denatured uranium-thorium fuel cycle employs thorium fuel which is enriched with a mixture of fissile uranium (U-235 or U-233) and fertile U-238; such denatured fuel is typically composed of about 85 to 88 percent thorium with the remaining percentage being uranium. The enrichment of the uranium fraction is limited to 20 w/o fissile U-235 or 12 w/o fissile U-233 or an appropriate weighted average if both U-235 and U-233 are present in the fissile component of the uranium with the remaining uranium bearing U-238 so that the uranium, even if chemically separated from the thorium, is unsuitable for use in nuclear weapons without further isotopic upgrading.

The fuel cycle involving use of thorium fuels enriched with denatured U-235 will be submitted to INFCE at a later date. The remainder of this paper considers the fuel cycle which involves the use of thorium fuels enriched with denatured U-233. In this cycle U-233 is the fissile component of the fuel in the initial core loading and in the reload fuel. Unlike U-235, the U-233 isotope is not found in nature so the U-233 required for the initial core loading and the U-233 required for the makeup component of the reload fuel will have to be obtained from some exogenous source, such as from the blanket region of a thorium-blanketed LMFBR, or such as an HWR fueled with thorium enriched with plutonium. Once operation of the denatured U-233 thorium

HWR has been initiated and the fissile content of the spent fuel recovered by reprocessing, the recovered U-233 will supply the major source of U-233, but some makeup U-233 will be required from the exogenous source; the amount of exogenous makeup U-233 will depend upon the fuel management (burnup).

Fuel cycle facilities required for the denatured U-233 thorium fuel cycle are shown in Figure II-1. This figure also illustrates the equilibrium cycle mass flows for a specific fuel management scheme in which a discharge fuel exposure of 15,000 Mwd/T is achieved. Denatured U-233 thorium fuel will have to be fabricated in specially designed hot-fuel fabrication facilities because of the gamma activity of the daughters of U-232 which is produced in trace quantities along with U-233 in thorium-bearing fuels. This facility will probably have to employ both remote fabrication and remote maintenance. The denatured thorium fuel is utilized in a HWR whose design is assumed to be identical to that of the conceptual reference design described in previous sections of this paper. To recover the U-233 content of the spent fuel from the HWR this fuel would be reprocessed in a Thorex reprocessing facility, although some modification of the Thorex process may be necessary because of the significant uranium and plutonium content of the spent fuel. A waste disposal facility, of course, will also be required and some provision must be made for plutonium storage or alternately for the use of plutonium in reactors which can generate additional quantities of U-233.

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

As the discharge exposure is decreased, the conversion ratio will be improved because the fission product burden is reduced and hence fewer neutrons are lost to parasitic captures in fission products. Consequently, less exogeneous U-233 will be required for makeup when a low discharge burnup is employed. However, when such low burnup fuel management schemes are utilized, fuel must be frequently reprocessed and refabricated and the cost for these services will adversely affect fuel cycle cost economics.

As the discharge burnup is increased, larger quantities of U-233 makeup will be required because of the decrease in conversion ratio which results from increased neutron losses to the larger quantities of fission products produced as a result of the higher burnup. In this case, additional costs will be incurred for the purchase of makeup U-233, but the cost for reprocessing and fuel fabrication will be reduced since smaller quantities of fuel would be reprocessed and fabricated each year. The optimal performance of the denatured thorium fuel cycle with respect to U-233 utilization, consequently, occurs at very low fuel discharge exposures, while the optimal from the standpoint of economic performance typically occurs at somewhat higher fuel discharge exposures and occurs as a result of the economic trade-off between the increasing cost for makeup U-233 and the decreasing cost of fuel fabrication and reprocessing services as the discharge fuel burnup is increased.

In this paper, a fuel management scheme has been characterized in which a discharge burnup of 15,000 Mwd/T is employed. As noted above, this discharge fuel exposure is not a unique characteristic of the HWR. Although an optimization study on discharge burnup has not as yet been performed, it is clear that superior U-233 utilization would be achieved by lower discharge exposures, but at the expense of fuel cycle cost economic, while the optimum burnup from the standpoint of fuel cycle economics is significantly greater due to the high cost of fabricating U-233 bearing fuels. This particular fuel discharge exposure was selected since it appears to be a reasonable compromise between U-233 utilization and fuel cycle cost economics; also, as discussed in the companion paper on once-through uranium fuel, the significantly high burnup necessary to reach an economic optimum (30-45 Mwd/kg or greater) would necessitate significant modifications to fuel design, as well as increase core peaking factors.

Fuel cycle information for the denatured Uranium-233 thorium cycle employing a fuel management scheme which achieves the discharge exposure of 15,000 Mwd/T is summarized in Tables II-1 and II-2. Mass flows for the equilibrium cycle of operation are also shown in Figure VII-1. Particularly noteworthy are the rather small annual makeup and 30-year cumulative net U-233 requirements, 125.4 and 5,230 kg/gwe, respectively. This contrasts to the LWR where 316 and 11795 kg/gwe of U-233 is required for annual makeup and for 30 years of operation, respectively. Also note that the quantity of fuel which must be fabricated and reprocessed each year is over twice as large as the corresponding quantities in the LWR, and so the cost for these services for the HWR would be proportionately higher.

III. TECHNOLOGICAL STATUS AND R&D REQUIREMENTS

The technological status of the HWR and the R&D requirements necessary to develop the conceptual HWR design are discussed in the companion paper on the once-through uranium fuel cycle, and are not repeated here. This section is limited to a discussion of the technological status and R&D requirements of the denatured U-233/Thorium fuel cycle.

The technology for the utilization of thorium-bearing fuels and for the recycle of U-233 is much less well developed than is the technology of the uranium fuel cycle. The use of the denatured thorium fuel cycle will therefore necessitate significant R&D efforts in the areas of fuel fabrication, reprocessing, and for reactor related data base and verification-type development. In general, the R&D requirements are also applicable to other water reactor types and could be developed as part of an integrated program; a number of items such as those relating to in-reactor performance are, however, specific to the HWR, and as noted above the HWR fuel cycle does place a particular premium on developing cost effective methods of reprocessing and refabrication.

The fabrication of U-233-bearing fuels is significantly different from the fabrication of uranium or plutonium-bearing fuels due to the radioactivity resulting from trace quantities of Uranium-232 which is produced along with the fissile material Uranium-233 from thorium fuels. Since the decay of Uranium-232 leads to daughter products which emit highly energetic gamma rays, the fabrication of U-233-bearing fuels necessitates remote operations and shielded facilities. Although such remote and shielded facilities are easy to visualize conceptually, the fabrication process is complex and such facilities have yet to be engineered for reactor grade U-233. One significant problem which must be addressed is the maintenance of such remote equipment; equipment must be quickly maintainable to avoid long downtimes for repair which would compromise the economics of the fabrication process. Because of the complexity of the pelletization process, it may be desirable to fabricate U-233-bearing fuels using VIPAC or SPHERE-PAC technologies, technologies which appear more amenable to remote operations. The use of VIPAC or SPHERE-PAC fabrication would, of course, necessitate additional R&D for process development and also for in-reactor performance qualifications, since neither of these alternative fabrication technologies are employed for the manufacture of commercial reactor fuels.

The denatured thorium fuel cycle also introduces significant new requirements for fuel reprocessing and waste treatment R&D. Reprocessing of thorium-based fuels is based upon the Thorex process. Although this process has been demonstrated for low-irradiation-exposure fuel, it is much less developed than the Purex process utilized for reprocessing uranium-based fuels. Since spent denatured thorium fuels will contain significant quantities of plutonium, as well as uranium and thorium, a modified version of the Thorex process will have to be developed and tested. Reprocessing of thorium-based fuels is also complicated by the fact that, unlike urania, thoria dissolves very slowly in nitric acid unless fluoride ion is added. The introduction of fluoride complicates the treatment of waste from the fuel dissolving process, and will necessitate additional R&D in this area. The fluoride ion also complexes with the zirconium cladding so that thoria dissolution is severely retarded unless excess fluoride is added (which would severely increase equipment corrosion). A more acceptable approach may be to remove the cladding before dissolving the thoria by some chemical or mechanical method. Here again, additional R&D will be required both to develop the dissolution process itself and for the treatment of waste which results from this process.

Although there has been some experience with the irradiation of thorium-based fuels in HWRs, additional R&D will also be required in this area. The major areas requiring R&D consist of data-base development and fuel performance qualification. This R&D is necessary to ensure that fuel performance meets licensing requirements, and to develop the information required for licensing thorium-fueled cores. Such information as in-reactor densification and swelling behavior, fission gas release, thermal conductivity of the fuel, pellet cladding interaction, and coefficients of reactivity must be established. Subsequent R&D would consist of in-pile irradiation demonstrations where significant quantities of thorium-based fuels, fabricated with processes and equipment representative of commercial fabrication technology, would be irradiated to provide a demonstration of in-reactor fuel performance.

The cost of the R&D program designed to implement the denatured U-233 thorium fuel cycle in HWRs will depend upon whether similar programs are instituted for light-water reactors since many elements of the R&D program are similar. Cost estimates have, however, been developed*, assuming that such R&D was performed only for HWRs. The estimated R&D costs for reprocessing, refabrication, waste treatment, and fuel qualification development are

* ORNL-5388, "Interim Assessment of the Denatured U-233 Fuel Cycle: Feasibility and Nonproliferation Characteristics (3-13-78 Draft)."

about $1 billion. In addition, the costs for the design, construction and operation of demonstration fabrication and reprocessing facilities has been estimated to be approximately $500 million in capital outlays and $300 million in operating expenses. The R&D program is estimated to require about 14 years to complete (1992 completion), while the reprocessing demonstration facility could begin operation in about the year 2000, assuming that design and construction were initiated in 1989.

IV. SAFETY AND LICENSING CONSIDERATIONS

The extensive analyses performed for the Candu-PHWR by the Canadians and the highly satisfactory performance of Candu reactors is a testimony to the overall inherent safety and licensability of the concept.

Although the individual licensing traditions of various nations must certainly be considered when commercial deployment of a concept is anticipated, consideration of unique or inherent safety features is much more productive at this stage, for it is reasonable to expect that if these inherent features are satisfactory, the various licensing criteria and practices can be accommodated by engineering design.

The ensuing discussion has therefore been limited to an identification of the unique safety considerations inherent in the HWR design previously described in this paper.

The primary inherent attributes of the pressurized heavy water reactor which contributes to increased plant safety are 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 use of redundant shutdown systems. The use of the pressure-tube configuration coupled with the injection of emergency core cooling water into both the inlet and outlet headers, facilitates the supply of coolant water to the fuel channels in the event of a loss of coolant accident. The presence of large quantities of relatively cool moderator surrounding the fuel channels provides a backup heat sink in the event of a failure of the ECCS system. The location of fuel within separate fuel channels, coupled with on-power refueling, also allows failed fuel to be readily detected and replaced. As a result of this ability to detect and replace failed fuel, the radionuclide inventory of the primary circuit can be minimized thus reducing the quantities of radioactivity which can be released to the environment during certain postulated accidents. Candu reactors are also provided with a diverse shutdown system in the form of gadolinium poison injection into the moderator tank.

Further work needs to be done to more fully assess the overall safety and licensing considerations of the HWR design described in this paper. This is not intended to imply that these aspects where applicable have not been fully and satisfactorily considered in the CANDU design.

This work includes an assessment of the adequacy of the zirconium niobium alloy pressure tubes as pressure boundaries with the higher pressures and temperatures employed in the HWR reference design in this paper, considering seismic factors which may be more demanding than Canadian criteria because of differing geology. This assessment would include an analysis of the probability and consequences of pressure tube failure and the potential for damage propagation to adjacent tubes.

Also, the impact of the small positive power coefficient in the operating range would require further evaluation; however, no difficulty in accommodating this inherent characteristic is anticipated, based on Canadian analyses of transient under-cooling and loss of coolant accidents. The impact of tritium levels on personnel exposures and upon inspection and maintenance would be considered, since the concentrations of tritium produced in the HWR are inherently larger than that of the LWR due to the presence of heavy water. Lastly, consideration would be given to potential refueling accidents since the use of on-line refueling could potentially contribute to more frequent refueling events.

An evaluation of the impact of thorium fueling on the safety and licensing of the HWR has not as yet been performed. However, preliminary assessments indicate that such important attributes as the power and void coefficients of reactivity are similar to those of uranium fueling, indicating that the consequence of postulated accident and operating transients should be acceptable.

V. ECONOMIC INFORMATION

A. Capital Cost

In the previous section, which addressed safety and accident considerations, we distinguished between the inherent safety features of the HWR concept and those which were motivated by local licensing traditions. The logic for doing so was that the latter aspect could be accommodated by suitable engineering design. These nonintrinsic safety and licensing considerations, however, can have a significant impact on plant design and on the resulting plant capital cost; estimates of plant capital cost can therefore not be divorced from the particular licensing traditions of various nations. Consequently, it is anticipated that capital cost development performed by various nations or to varying sets of licensing criteria, will differ. The estimates of capital cost which are discussed below are predicated on what is thought to be a reasonable application of U.S. licensing criteria, where criteria and practices seem to apply equally well to the HWR and LWR; however, consideration has been given to the unique attributes of the HWR when applying this criteria.

The capital costs of the HWR conceptual design are summarized by cost category in Table V-1. The resulting total capital costs (exclusive of heavy water) are approximately \$588/kw(e) (tentative, subject to detailed conceptual design development and costing), or 8.7% higher than that of the LWR costed on the same basis. As noted in the discussion of capital cost ground rules discussed below, this cost of \$588/kwe excluded interest during construction, escalation during construction, contingency allowance, and owners cost. The contingency allowance has been excluded since it is assumed that the unit is one of an established standard plant design. In order to establish power costs, interest during construction must be added to the base capital cost of \$588/kwe. Interest during construction will vary depending upon the length of time required for construction, the schedule of payments during this period, and the cost of money (interest). Using the 10-year total construction period and schedule of payment of Reference 4 and a 4.525% cost of money (deflated effective interest rate) results in an interest during construction cost of 21% of the base capital cost, or a total capital cost of $711/kwe in January 1978 U.S. dollars. This value is utilized to establish the capital cost contribution to the total power costs, discussed in Section V-C. The capital cost of the plant delivered at some future date will, of course, be higher when viewed in terms of the then current dollars because of escalation. For example, the cost of a plant for 1990 operation would be about $1603/kwe in 1990 dollars assuming a 8% escalation rate; however a 1990 dollar would be worth only about 0.40 1978 dollars if there were an 8% inflation rate.

The ground rules for this capital cost development are sumamrized briefly as follows:

1. Cost data are for plants deliverable on January 1, 1978, and are in January 1, 1978 U.S. dollars.

2. As discussed above, U.S. licensing and design criteria, safety classifications, seismic categories, and design codes were employed to the greatest extent possible.

3. The cost estimate was developed for a single unit on a typical new site with sufficient land area to accommodate a second unit. The site employed was the hypothetical "Middletown, U.S." site, which has also been used to develop capital costs for other concepts.

4. The main heat rejection system and the moderator heat rejection system is based on mechanical draft evaporative cooling towers.

5. The plant has on-site nuclear reactor core storage capacity for four-thirds core.

6. The plant design lifetime is 30 years for base-loaded operation.

7. The following items have not been included in the plant capital cost estimate:

   a. Main transformer, switchyard, and transmission facility costs.

   b. Owner's cost, including consultants, site selection, spare parts, etc.

   c. Off-site waste disposal cost.

   d. Nuclear liability insurance.

   e. Initial fuel loading.

   f. Interest during construction.

   g. Escalation during construction.

   h. Contingency allowance.

In addition, the cost of the heavy water inventory has been excluded from the tabulation of plant capital cost, and included as a component of the fuel cycle cost.

As noted above, it is anticipated that the perceived capital cost of an HWR plant will vary, depending upon the local licensing traditions which are employed, as well as upon other considerations, such as the cost of labor and labor productivity. However, although the total capital cost may vary, it is felt that the capital cost differential with respect to the LWR is reasonable and relatively insensitive to these considerations. Both the light water reactors, as embodied by the pressurized water reactor concept, and the pressurized heavy water reactor have many elements in common. Both employ steam generators, utilize primary system pumps, and utilize similar containment structures; consequently, one would anticipate roughly equivalent capital costs for reactors of equal thermal capacity. Of course, there are a number of cost savings inherent in the HWR design, particularly the absence of the relatively costly pressure vessel utilized in PWRs; on the other hand equipment not present in the PWR, such as the pressure tube assemblies, refueling machines, and separate moderator heat exchange system, will incur costs in excess of the savings on the pressure vessel. The overall cost differential between the HWR and PWR is explainable in terms of the overall lower net electrical efficiency of the HWR plant, the larger containment structure necessitated by the physically larger HWR NSSS, and by the additional equipment necessary to minimize the irrecoverable losses of heavy water.

B. Operational and Maintenance Costs

Detailed O&M costs have not been developed, but are estimated to be 0.98 mills/kw-hr in January 1978 dollars, or 20% higher than the equivalent O&M costs for the PWR. The primary difference between LWR and HWR O&M costs is the cost of D_2O makeup and upgrading, and the cost of maintenance of the refueling machines. Other components of the O&M cost, such as allowance for staff requirements, maintenance materials, supplies and expenses, nuclear liability insurance and operating fees, and administrative and general overhead expenses are expected to be similar for the two reactor types.

C. Fuel Cycle and Power Costs

Estimates of fuel cycle cost for the denatured Uranium-233 thorium fuel cycle are compared with the slightly enriched once-through uranium cycle in Table V-3. Pertinent economic parameters are given in Table V-2. 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 a U-233 price, the fuel cycle cost evaluation was performed parametric in this quantity, with U-233 prices varying between 10 and $40/gm. As Table V-3 indicates, this particular denatured Uranium-233 thorium cycle (i.e., with fuel discharge exposure of 15,000 Mwd/T) will not be economically competitive with the slightly enriched uranium once-through cycle until the price of U_3O_8 increases significantly from the current $40/lb value, even if U-233 can be obtained cheaply (the break-even U_3O_8 price for a U-233 cost of 0\$/gm is about \$75/lb). For 100 \$/lb U_3O_8, the fuel cycle costs of the once-through uranium would equal that of the denatured U-233 thorium cycle if the U-233 cost were about 15 \$/gm.

As Table V-3 indicates, the rather high fuel cycle cost for the denatured U-233 thorium cycle is due primarily to the cost for fabrication services and to a lesser extent to the cost of fuel reprocessing. Although the lower discharge burnup assumed for the denatured thorium cycle (15,000 Mwd/T for the denatured thorium cycle vs. 19,750 for the slightly-enriched once-through cycle) contributes to the higher fuel cycle service cost for the denatured cycle, the main reason for these increased costs are the higher costs of the fuel cycle services themselves: as Table V-2 indicates, a fabrication cost for U-233 bearing fuels of \$350/kg was employed vs. a fabrication cost of \$60/kg for slightly enriched uranium fuel, while reprocessing incurs a cost of \$180/kg in contrast to spent fuel disposal costs of \$115/kg. The fabrication and reprocessing components could, of course, be reduced by employing higher discharge exposures (at the expense of increase U-233 requirements and cost), and the alternative of higher discharge exposures may well be the preferred option until more innovative and cost-effective methods for fabricating U-233 bearing fuels are developed.

VI. ENVIRONMENTAL INFORMATION

The environmental information associated with the use of HWR is discussed in the companion paper on the once-through uranium fuel cycle, and will not be repeated here. The main area where the denatured U-233/thorium fuel cycle differs significantly from the uranium cycle is in the need for reprocessing and the potential radiological hazard of U-233 bearing fuels because of trace quantities of U-232, the daughters of which emit fairly high energy gamma rays. This activity will require increased shielding for fresh fuel handling and/or revised fuel handling procedures to minimize the occupational exposure of plant personnel. The increased activity of thorium fuels has a much greater impact, however, on the fuel refabrication industry and, as a consequence, results in significantly higher fabrication costs for denatured U-233/thorium fuel than for uranium fuel. Reprocessing will require the use of a modified Thorex process; the environmental impact of this process has yet to be fully evaluated. A thorium mining industry with its attendant environmental impact will also be required.

However, the use of the denatured U-233/thorium cycle would reduce the requirements for uranium ore mining and milling; this segment of the fuel cycle industry typically has the largest environmental impact. The need for long-term spent fuel storage would also be eliminated.

TABLE I-1

GENERAL REACTOR PERFORMANCE SPECIFICATIONS

A. Power Plant Performance	
Thermal Power (Mw)	
Total Nuclear	4029
Coolant	3800
Electrical Power (Mwe)	
Gross	1343
Net	1260
Thermal Efficiency	31.5
B. Reactor Parameters	
Core Volume (liters)	321,000
Core Dimensions (m)	
Equivalent Diameters	8.8
Active Length	5.9
Core Power Density (Mw/ℓ)	0.012
Coolant Flow Rate (Mg/sec)	14.2
Coolant Inlet Temperature (°C)	299
Coolant Outlet Temperature (°C)	337
Primary System Pressure (psia)	2200
C. Fuel Parameters	
Average Fuel Temperature (°C)	936
Maximum Fuel Temperature (°C)	1900
Cladding Temperature (°C)	350
Core Fuel Loading	
Total Heavy Metal	151,131
Fissile Material (initial)	2235
Discharge Exposure (MWD/Mg)	
Average	15,000

TABLE II-1

FUEL CYCLE INFORMATION DENATURED U-233 CYCLE

Discharge Exposure (MWD/Mg)	15,000
Fraction of Core Replaced, %/yr	53.9
Fuel Residency Time, yrs	1.9
Fissile Fabrication and Reprocessing Loss, %	2.5
Requirements for U-233 (kg/GWe)	
Initial Core	1774
Annual Equilibrium (net)	126.6
30-Year Cumulative (net)	5455

TABLE II-2

Reactor Fresh and Spent Fuel Characterization

TYPE (General Description) Zr clad Denatured U-233 pellets for HWR

Refueling Method: On-line X ; Batch ___; (Refueling frequency ______)

Fuel Assembly Characteristics: (where applicable)

a) type: Oxide X ; Metal___; Carbide___;

b) weight: 19.3 kg

c) length: 0.5 m

d) core load: 151,131 mass (kg HM):

e) annual reload: 73530 mass (kg HM):

Design burnup: 15000 (Mwd/MT) discharge batch average

Discharge fuel radiation level: NA r/hour @ 1 meter (also provide a curve of radiation level versus cooling time following discharge)

Discharge fuel energy generation rate as a function of cooling time.

W/hr/element) - provide curve. NA

Heavy Element Isotopic Content (kg/fuel element) at discharge

ISOTOPE	Fresh Fuel Element		Discharged Fuel Element	
	initial	equilibrium	initial	equilibrium
Th-232	14.96	14.82	14.71	14.58
U-232	-	-	-	-
U-233	0.25	0.25	0.22	0.22
U-234	0	0.09	0.022	0.094
U-235	0.003	0.021	0.005	0.020
U-236	0	0.012	0.003	0.012
U-238	1.81	1.81	1.79	1.79
Np-237			-	-
Pu-238			-	-
Pu-239			0.006	0.006
Pu-240			0.003	0.003
Pu-241			0.0005	0.0005
Pu-242			0.0001	0.0001
Am-241				
Cm-242				

*Also provide graphs of fissile content (Pu and U) vs Burnup (GWD/MT)

TABLE V-2

ECONOMIC PARAMETERS

U_3O_8 (\$/kg)	88
Separative Work (\$/SWU)	80
Enrichment Plant Tails (w/o)	0.2
D_2O Cost (\$/kg)	213
Plant Factor (%)	75
Plant Life (Years)	30
UO_2 Fabrication (\$/kg)*	60
Spent Fuel Disposal (\$/kg)*	115
U-233 Fabrication Costs (\$/kg)*	350
Reprocessing Costs (\$/kg)*	
Reprocessing	150-250
Spent Fuel Shipping	10
Waste Shipping	5
Waste Storage	15
Total Reprocessing	180-280

*TC-1064, "Thorium Assessment Program Systems Studies," Hanford Engineering Development Laboratory, February 1978. U-233 fabrication costs are based upon a CANDU 37 pin assembly. Comparable denatured U-233/Th fabrication costs for a PWR or SSCR in the same time frame (year 2001 conditions) would be 550 $/kg.

TABLE V-3

FUEL CYCLE COST
(mills/kw-hr)

		Denatured U-233 Cycle
Fabrication		2.78
Reprocessing or Fuel Disposal[4)]		1.27
U_3O_8		-
SWU		-
Carrying Charges[1)]		0.86
D_2O		2.38
	Subtotal	7.29
Plutonium Credit at 34 $/gm		-0.12
U-233:[2)]	10 $/gm	.87
	15 $/gm	1.31
	20 $/gm	1.74
	40 $/gm	3.48
Total Fuel Cycle Cost[3)]:		-
U-233 @	10 $/gm	8.04
	15 $/gm	8.48
	20 $/gm	8.91
	40 $/gm	10.65

1) Excludes Carrying Charge on U-233.
2) Includes Carrying Charge on U-233.
3) To obtain total busbar power costs add 11.8 mills/kw-hr.
4) Based upon reprocessing cost at lower end of range of 150 $/kg-HM

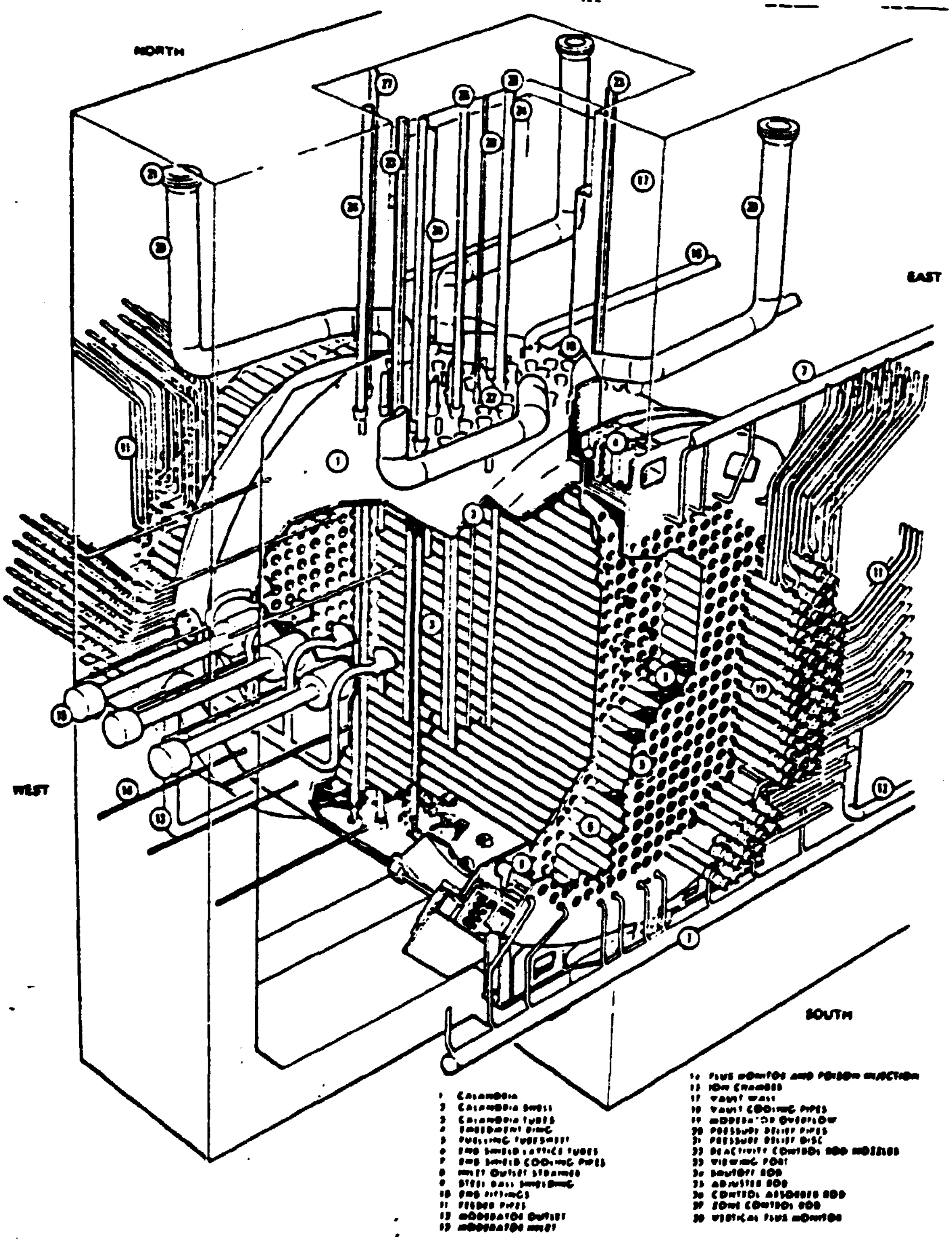


Fig.I-1 CANDU Reactor Assembly.
ANL Neg. No. 116-77-39.

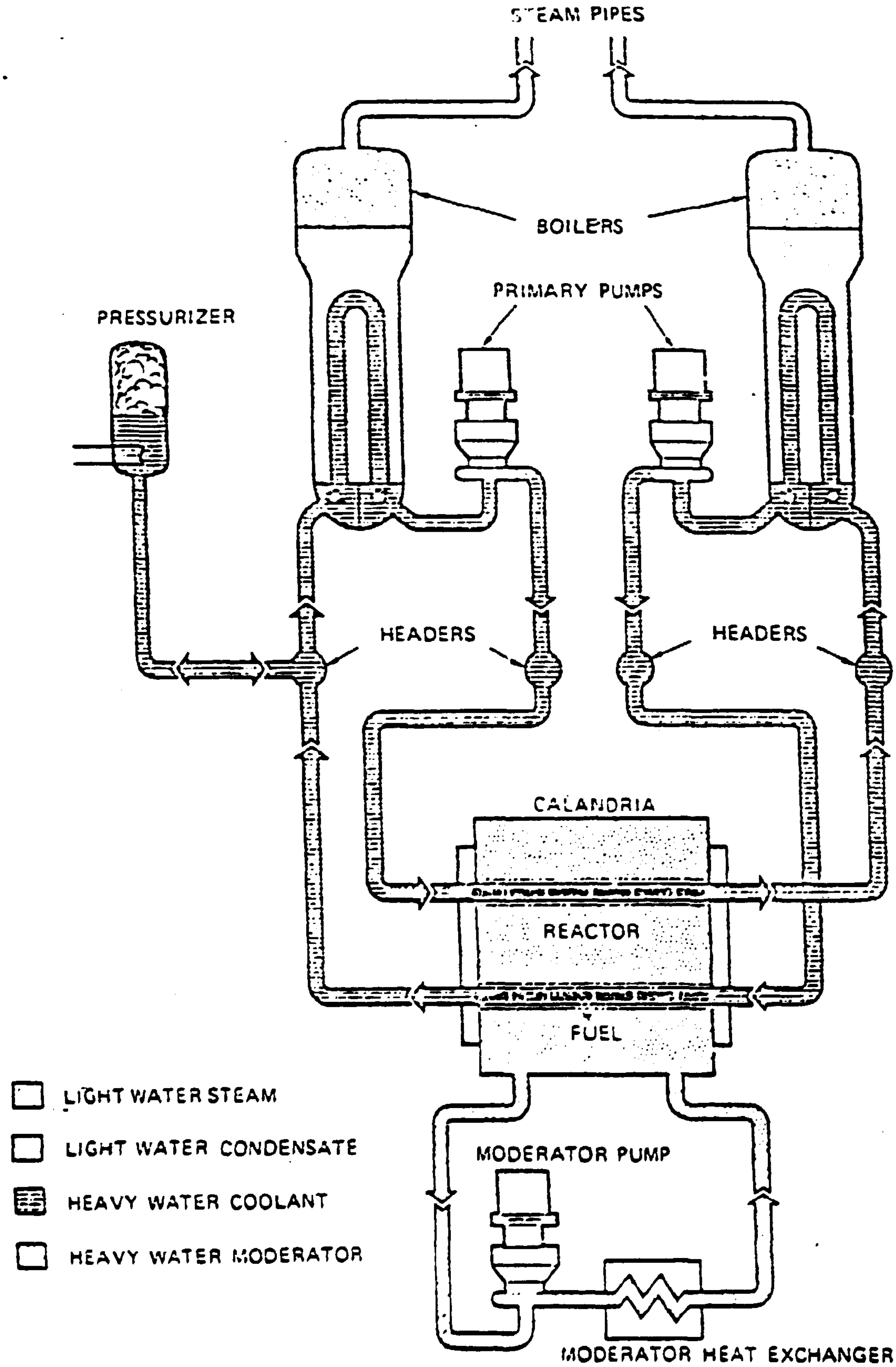
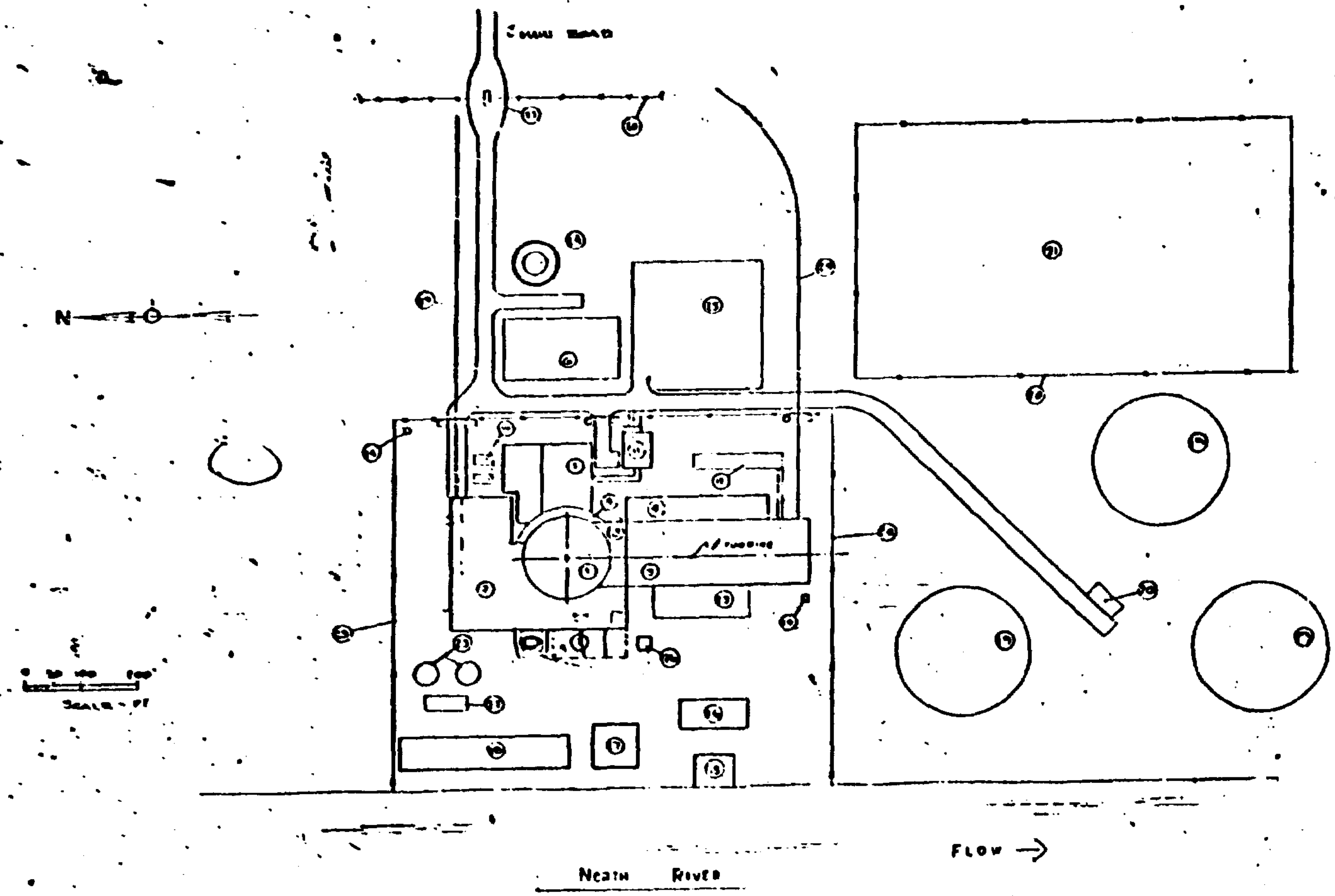


Figure I-2

Heavy Water Reactor Simplified Flow Diagram

Figure I-3



Plant Nomenclature

1 - Reactor Containment Bldg
2 - Reactor Service & Fuel Handling Bldg
3 - Turbine Bldg
4 - Heater Bay
5 - Control & Diesel Bldg
6 - Administration Bldg
7 - Underground Storage Tanks, 1) Condensate, 2) Emergency Water, 3) Demineralized Water.
8 - Refrigeration Bldg
9 - Auxiliary Bldg
10 - Diesel Storage Tanks
11 - Security Bldg
12 - Transformer Area
13 - Circ Water & Service Rad Bldg
14 - Air Intake Structure
15 - Intake Structure
16 - Make-up Water Pretreatment Bldg
17 - Holding Pond Plant Effluent
18 - Ultimate Heat Sink Cooling Tower Structure
19 - Cooling Tower
20 - Cooling Tower Switchgear Bldg
21 - Switchyard
22 - Fire Water Pump House
23 - Fire Water Storage Tanks
24 - Fuel Oil Storage Tank
25 - Parking Area
26 - D_2O Upgrading Tower
27 - Guardhouse
28 - Security Fence
29 - Railroad

Preliminary Rot Ran PHWR Plant (1978)

D.P. 005

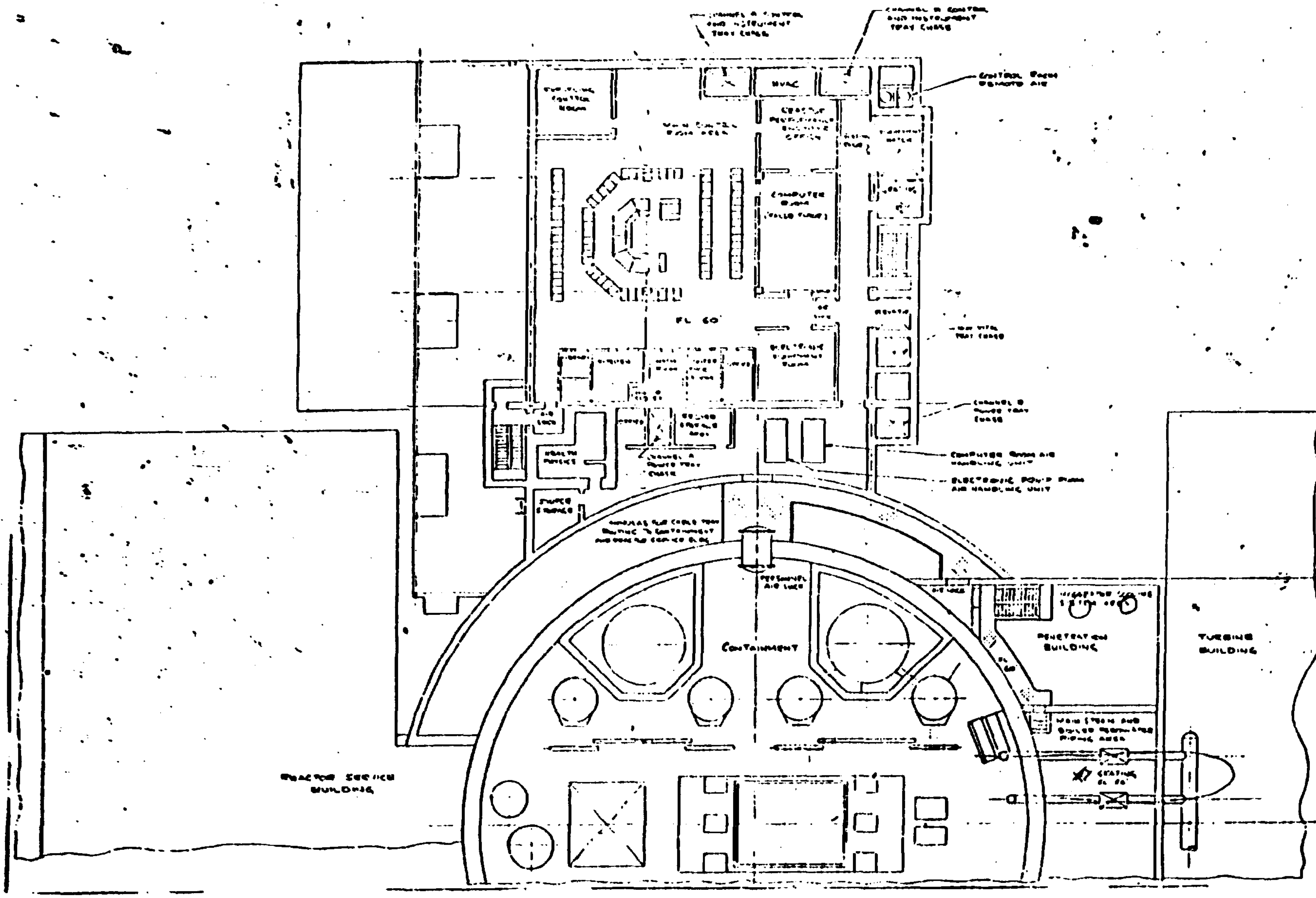

HVAC
COMPUTER ROOM
EL 60'
CONTAINMENT
PENETRATION BUILDING
TURBINE BUILDING
REACTOR SERVICE BUILDING

Figure H-4

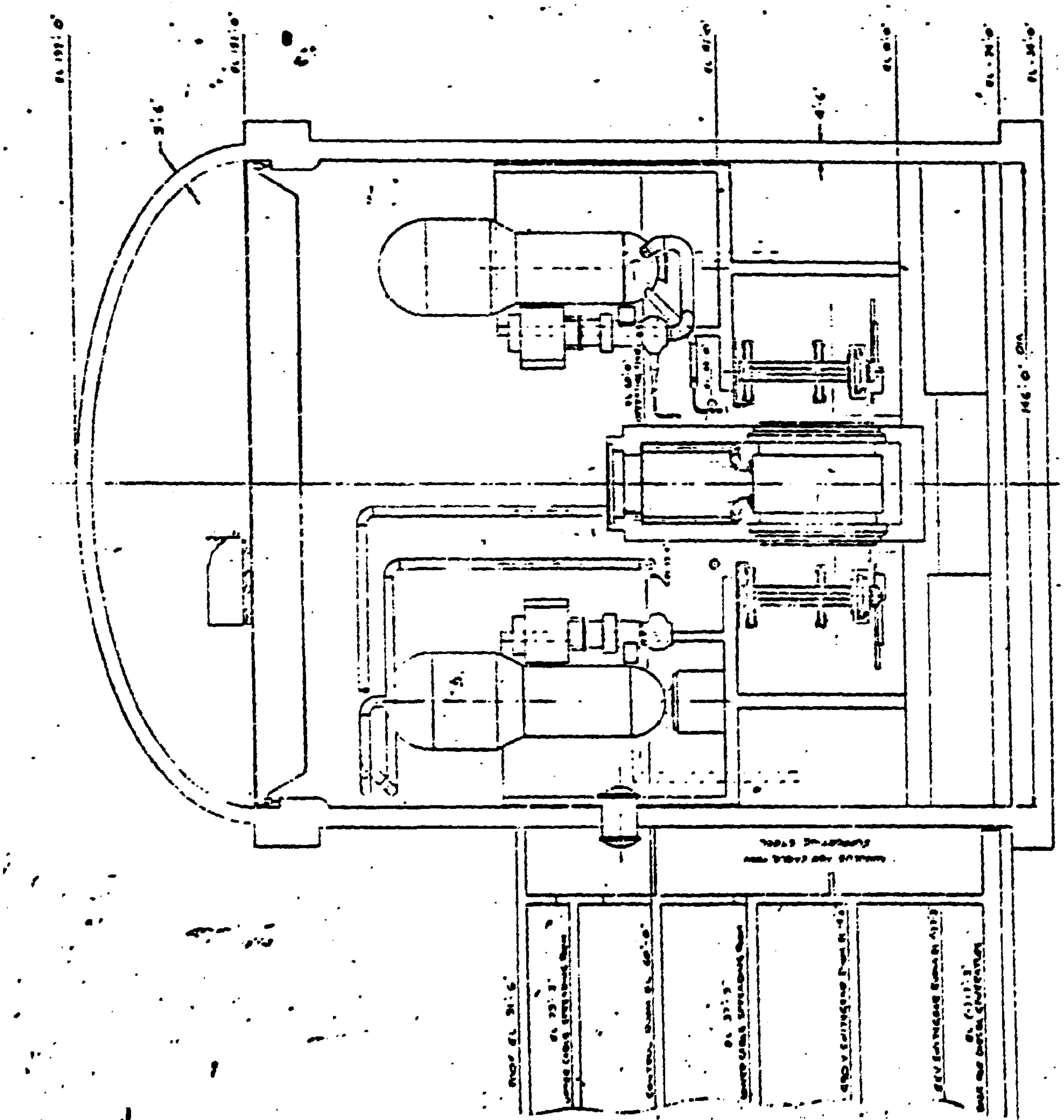
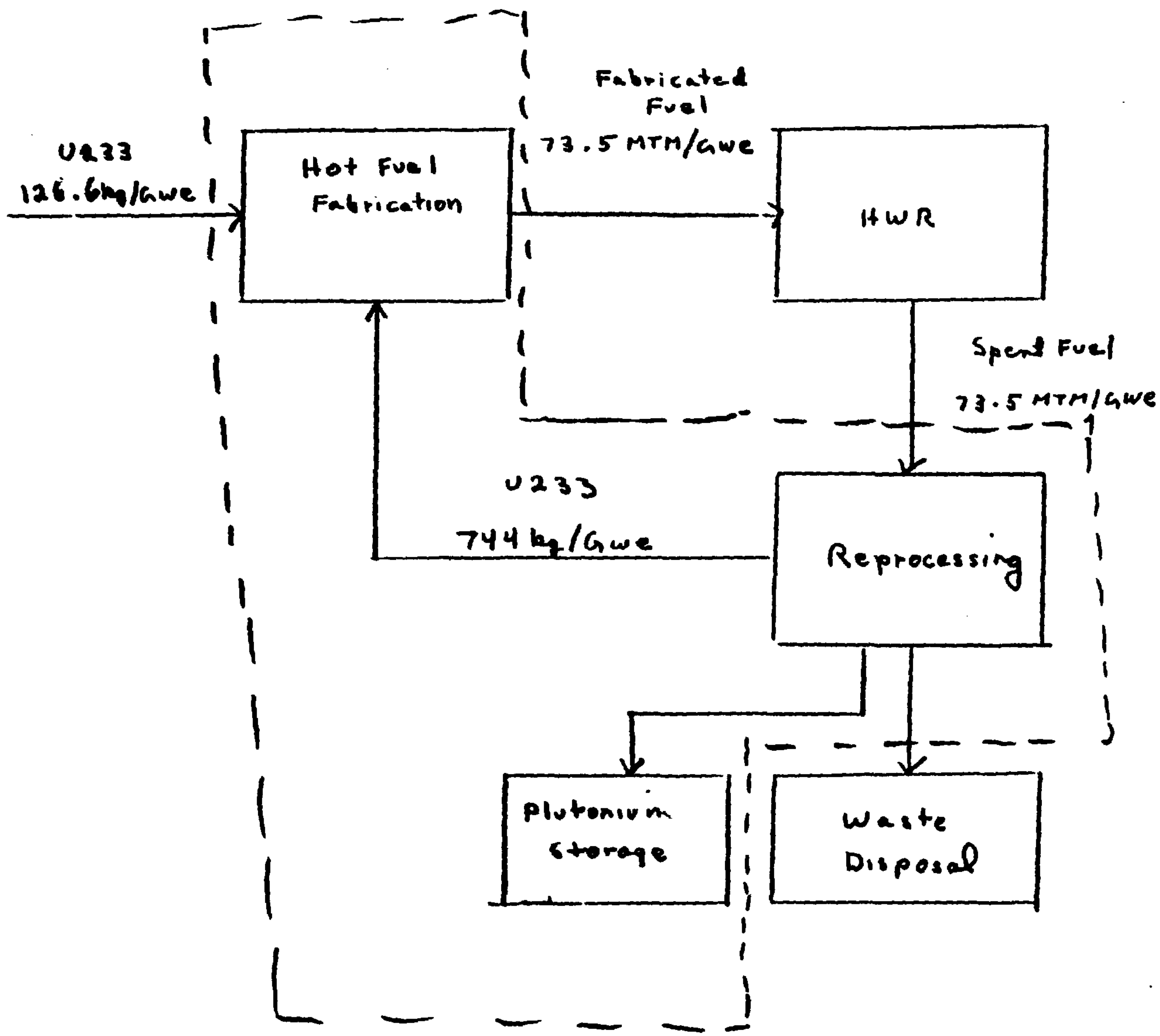


Figure II-1
Fuel Cycle Facilities



- - - indicates boundaries of secure site