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PROPOSED DRAFT PAPER:

HEAVY WATER REACTORS

ON THE ONCE-THROUGH URANIUM CYCLE

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HEAVY WATER REACTORS ON THE ONCE-THROUGH URANIUM CYCLE

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 once-through uranium fuel cycle for use in early comparisons of alternate nuclear systems. The denatured thorium fuel cycle is discussed in a companion paper. Information for this paper was developed under an ongoing program, and more complete reporting of the evaluation of the heavy water reactor and its fuel cycles is planned toward the end of the year.

Although a number of alternate heavy water reactor concepts, such as the Steam Generating Heavy Water Reactor (SGHWR) which employs light water or 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 Candu PHWR have been incorporated in the HWR reference design described in this paper in order to obtain improved uranium resource utilization on the once-through fuel cycle and to enhance the prospects for economic competition with the LWR. Perhaps foremost of these modifications is the use of slightly enriched uranium fuel, rather than the natural uranium fuel currently employed in Candu reactors. The conceptual design also utilizes an increased primary system pressure so as to increase system efficiency and reduce the effective capital cost of the plant. The 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 characterized in this section 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. As discussed in greater detail below, the conceptual reactor being characterized herein differs from the Canadian Candu 600 design primarily because of the increase in station size to 3800 Mw(th), 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 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 diagramed 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.

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 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 local power distribution within the reactor core. Vertical adjuster rods, control rods containing a graded vertical loading of cobalt poison, are also provided for xenon power override. 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.

In developing the reference conceptual NSSS design described in the preceding paragraphs, a number of design parameters were considered, including initial enrichment, primary system pressure, reflector thickness, moderator temperature, and reduced lattice pitch. The evaluation of optimal initial enrichment consisted of a parametric study in which initial enrichment was varied between natural uranium and 2 w/o U-235 to establish the effect on resource utilization, fuel cycle cost economics, and power peaking. An enrichment of 1.2 w/o was selected as being near optimal from the standpoint of both resource utilization and fuel cycle cost. Only slight improvements in these parameters are possible with higher initial enrichments, while increased core power peaking is observed as the enrichment is increased.

A primary system pressure of 2200 psi was selected based upon a parametric analysis which indicated that the decrease in achievable burn-up resulting from the increase in pressure tube thickness necesssary to accommodate this pressure is relatively small (1,000 Mwd/T) for the slightly-enriched uranium system. Increasing primary system pressure is advantageous since it allows secondary steam pressures to be increased, with a resulting increase in net electrical efficiency. As a result of the improved electrical efficiency, the effective capital cost of the HWR ($/kwe) is reduced. It is this reduction in effective capital cost which provides the motivation for high primary system pressures and improved electrical efficiency; fuel cycle costs and uranium utilization (STU_3O_8/Gwe-yr) are largely unaffected by this design change since the effects of improved electrical efficiency and the reduced burnup mentioned above largely compensate.

The thickness of the D_2O reflector which surrounds the reactor core was also reduced slightly to minimize heavy water inventory, and the temperature of the D_2O moderator was increased (from 150 to 200^oF) to minimize the size of the moderator cooling system; both these modifications have little effect on the burn-up which can be achieved with slightly-enriched uranium fuel, and result in slightly reduced plant capital costs. The parametric evaluation of pressure tube pitch indicated little incentive for changing from the standard Candu lattice design. Only small reductions in power cost resulted, with savings in D_2O inventory charges being compensated by reductions in achievable burn-up, and in view of the considerable redesign which would be required to reduce lattice pitch, the standard Candu lattice pitch was elected for the reference conceptual design.

B. <u>Balance of Plant</u>

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 coolat flow rate are shown in Table I-1.

The fuel parameters shown in Table I-1 are based upon the use of slightly enriched uranium fuel containing an initial enrichment of 1.2 w/o U-235. Optimization studies, in which the performance of fuel parametric in initial enrichment was determined, indicates that this enrichment is near optimal both from the standpoint of uranium resource utilization and fuel cycle costs.

II. FUEL MANAGEMENT AND HANDLING INFORMATION

Fuel cycle facilities required for the slightly enriched fuel cycle in the HWR are shown schematically in Figure II-1. As this figure illustrates, the fuel cycle facilities which are required are limited to an enrichment plant to provide the slightly enriched fuel utilized, a fabrication facility, and a spent fuel storage facility for long term storage (or permanent disposal of) discharged fuel; interim storage capacity of one and one-third core loadings is also provided as part of the reactor facility. Fuel cycle information for the slightly enriched cycle is provided in Tables II-1 and II-2. This fuel cycle information was generated for an average capacity factor of 75%; this assumed capacity factor does not necessarily represent the expected capacity factor for the HWR, but rather was selected as a standard value to facilitate the intercomparison of uranium requirements with other concepts.* Also note that although the HWR is refueled with slightly enriched fuel, the initial loading is comprised of natural uranium fuel elements. The use of natural uranium fuel in the initial core minimizes initial core loading requirements and the period of time during which the HWR must be operated with soluble boron shim for reactivity control.

*HWRs are expected to operate at somewhat higher capacity factors than LWRs because one-line refueling eliminates the need for outages explicitly for refueling. Assuming that both PWRs and HWRs require a four week outage every other year for maintenance and inspection (during which the PWR is refueled), that the PWR requires an additional three week outage every other year explicitly for refueling, and that both reactors have equal availability between refuelings, results in the HWR having a capacity factor 2-3 percentage points greater than the LWR.

30-Year cumulative separative work and U_3O_8 requirements for the HWR are intercompared with those of the LWR in Table II-3. As this table illustrates, separative work requirements for the HWR are only a small fraction of those of the LWR, while U_3O_8 requirements are reduced to about 60% of those of the standard LWR when slightly-enriched fueling is employed.

III. TECHNOLOGICAL STATUS AND R&D REQUIREMENTS

Current heavy water reactor technology is most universally represented by the Candu pressurized heavy water reactor (Candu-PHWR). Reactors of this type having electrical generating capacities of between 500 and 750 Mw(e) are in operation in Canada and in operation or under construction in a number of other nations, such as India, Pakistan, Argentina and Korea.

Extensive operating experience has been obtained for Candu reactors, particularly for the Pickering Station which contains four 500 Mw(e) plants.

The first of the Pickering units went into operation in mid-1971, and a fourth unit in mid-1973. Performance of the Pickering station has generally been excellent, and a lifetime average capacity factor of 75% has been achieved, a capacity factor significantly higher than that reported for the 66 LWRs operating in the U.S. Thus, it can be concluded that the technical feasibility of heavy water reactors operating with natural uranium fueling has been conclusively demonstrated. Although the operation of heavy water reactors on the natural uranium cycle has been quite satisfactory, as mentioned above, the design described herein employs a limited number of modifications. Significant among the modifications are the use of slightly-enriched uranium fuel, higher primary (2200 psia vs. 1600 psia) and secondary pressures, and modifications to system structures and components to increase the generating capacity of the unit and to meet U.S. licensing requirements. R&D requirements necessary to develop these modifications to the point of commercial application are summarized in Table III-1.

The use of slightly-enriched uranium fuel (approximately 1.2 w/o) will increase discharge burnup to approximately 20,000 Mwd/T, as compared to approximately 7500 Mwd/T for the current Canadian designs. Although major improvements in current fuel technology are not considered necessary in order to achieve these higher burnups, development testing for mechanical performance acceptability will be required. The R&D program identified in Table III-1 for slightly-enriched fuel is comprised of two major components: the determination of safety-related physics parameters and fuel irradiation testing. Safety-related physics parameter experiments would consist of critical experiments in which such parameters as core reactivity, coefficients of reactivity, power distributions, and control absorber worth are measured for slightly-enriched uranium fuel. The fuel irradiation testing program would consist of the irradiation of a number of demonstration fuel assemblies containing an initial enrichment sufficient to achieve the discharge burnups anticipated for the slightly enriched HWR. These experiments, which probably could be performed in existing test reactors, such as NPD, would provide a measure of the power peaking anticipated in the slightly-enriched uranium fuel and demonstrate the performance of such fuels up to the burnups anticipated and for the power changes which occur during refueling. Fuel-related R&D programs might take five years to complete, including time for experiment design and fabrication, irradiation, and post-irradiation examination. A total program cost on the order of $3,000,000 is anticipated.

The use of higher primary system pressures and temperatures in the conceptual HWR design will require requalification of the zirconium-niobium pressure tubes. The thicker pressure tube material requirements will also involve rolled joint development, and the higher pressures may necessitate some redesign of the refueling machines. R&D requirements necessitated by the higher system pressures and temperatures are again twofold. The first aspect of the program would consist of an experimental program to demonstrate the integrity of the pressure tubes and rolled joints at the higher operating pressures and temperatures. This program would be similar in nature to tests already performed on Candu pressure tubes and would be intended to extend the range of validity of experimental information to the higher pressures and temperatures anticipated for the conceptual design. The second aspect of the program would consist of thermal hydraulic tests intended to extend the empirical correlations which establish the margin to burnout to the higher pressures and temperatures employed in the conceptual HWR design. These experiments are estimated to cost approximately $1,000,000 and require a two-year time period for performance.

The third area which should be mentioned is the modification to system structures and components required to increase the electrical capacity to the 1000 Mw(e) size or larger, and to meet the codes and standards of the nation in which the HWR is to be deployed. The major area requiring R&D appears to be the development of the large coolant pumps presently being considered for the conceptual design (a larger number of smaller pumps could also be used, at some cost penalty). In general, other modifications are dependent on local licensing traditions, and while design development may be required, no major generic R&D proof testing is thought necessary.

Consequently, it has been assumed that the first HWR employing the modifications incorporated in the conceptual design could be a lead commercial plant of 1000 Mw(e) size or larger, and that any R&D required to achieve this plant size or meet local licensing traditions would be performed as part of the engineering for this lead commercial unit. It is estimated that an R&D time span of two years would be required for the lead commercial unit, followed by a twelve-year design, construction and licensing period. First of a kind charges, inclusive of development costs, are estimated to be in the order of 25 - 50 $M.

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 shut-down 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 back-up heat sink in the event of 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 undercooling 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.

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. The 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 bases. As noted in the discussion of capital cost ground rules discussed below, this cost of $588/kw(e) excludes 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/kw(e). 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 ten-year total construction period and schedule payments 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/kw-hr in January 1978 U.S. dollars. This value is utilized to establish the capital cost contribution to the total power costs, discussed in Sections V-C and D. 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/kw(e) in 1990 dollars assuming an 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 summarized 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 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 reactor, 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 Costs

Economic parameters which have been utilized in the calculation of fuel cycle costs are specified in Table V-2. The resulting fuel cycle costs are tabulated in Table V-3, where the fuel cycle cost for the PWR, evaluated under identical ground rules, is also given for comparison. Table V-3, in addition to specifying the total fuel cycle cost, also shows the major components, such as fabrication, fuel disposal, SWU and U_3O_8, which comprise the total fuel cycle cost. Estimates of the total fuel cycle cost for economic parameters other than those specified in Table V-2, can be obtained simply by adjusting these components and retabulating the total fuel cycle cost. Since the capital cost for the HWR excluded the cost of D_2O, D_2O inventory charges have been included in the tabulation of fuel cycle cost, based upon a fixed charge rate of 10%.

D. Power Costs

As noted in Section A, the capital cost of an HWR, designed and constructed according to U.S. practices, is evaluated to be approximately 8.7% higher than that of the PWR. Compensating for this higher capital cost is the lower fuel cycle costs of the HWR. However, the relative fuel cycle cost advantage of the HWR depends upon a number of parameters, the most uncertain and variable of which is the price of U_3O_8. Consequently, there is some price of U_3O_8 at which the HWR and PWR will achieve equal power costs; for lower U_3O_8 prices, the PWR will obtain lower power cost while for higher U_3O_8 prices the HWR will achieve superior economic performance.

This break-even U_3O_8 price, for which power costs for the HWR and PWR are equal, is plotted in Figure V-1 as a function of assumed HWR capital cost penalty. As this figure illustrates, equal power costs occur for a U_3O_8 price of about $65/lb. at the 8.7% evaluated capital cost penalty of the reference conceptual HWR design. This evaluation has assumed equal capacity factors in the PWR and HWR. The break-even U_3O_8 cost is shown in Figure V-2 as a function in capacity factor difference between the HWR and PWR, under the assumption of an 8.7% capital cost penalty. As noted in Section II, HWRs are expected to operate at a capacity factor about 2-3 percentage points higher than LWRs because of on-line refueling. Increasing the HWR capacity factor by this amount results in a break-even U_3O_8 cost of about $55/lb U_3O_8. Figure V-2 also shows the effect of two other important variables, the cost of D_2O and SWU.

It should be noted that even with the reference economic parameters (8.7% capital cost penalty and 40$/lb U_3O_8) power costs for the HWR are only 0.9 mills/kwhr or 5.3% above those of the LWR. Since this cost differential is thought to be within the uncertainty inherent in the capital cost development and projections of economic parameters, and well within the anticipated variation in capital cost from nation to nation as a result of differing licensing and construction practices, and considering the strong dependency in power costs comparisons on plant availability, uranium ore costs, heavy water costs, and uranium enrichment costs the power costs for the two reactor types can be considered as being essentially equal.

VI. ENVIRONMENTAL INFORMATION

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

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

1. The conceptual reference HWR design operates at a somewhat lower efficiency than that of the PWR, 31.5% vs. 33-34% for the PWR. This lower efficiency results because of heat loss to the moderator; the heat deposited in the moderator must be rejected to the environment since the moderator temperature is too low for electrical power generation. As a consequence, 4-5% more heat will be rejected to the atmosphere.

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

3. Because the HWR is cooled and moderated with heavy water, production of radioactive tritium is much higher than in the LWR. Careful attention is paid in the design of HWRs to minimizing D_2O/tritium losses, both because of the radiological impact and because of the cost of D_2O. Nevertheless, some tritium will reach the environment during normal operation and considerable inventories of tritium are potentially available for release during accidents.

D_2O production facilities are also required, with their attendant environmental impact. Other aspects important to the environmental assessment, such as gaseous liquid and solid effluents, land utilization, critical materials, and socio-economic factors will be similar to those of the LWR.

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 (^oC)	299
Coolant Outlet Temperature (^oC)	337
Primary System Pressure (psia)	2200
C. Fuel Parameters	
Average Fuel Temperature (^oC)	936
Maximum Fuel Temperature (^oC)	1900
Cladding Temperature (^oC)	350
Core Fuel Loading (kg)	
Total Heavy Metal	166,056
Fissile Material (initial)	1181
Discharge Exposure (MWD/Mg)	
Average	19,750
Peak (estimated)	25,000
Conversion Ratio	
Average During Equilibrium Cycle	0.72

TABLE II-1

FUEL CYCLE INFORMATION

Average Capacity Factor	75%
Fraction of Core Replaced, %/yr.	37.3
Refueling Interval	On-power
Form of Fabricated Fuel	UO_2
Fuel Residence Time, yrs	2.68
Fuel Composition	See Table II-2
Fissile Fabrication Loss, %	1.5
U_3O_8 Requirements (ST/Gwe)	
Initial Core	172.6
Annual Equilibrium	114.2
30-Year Cumulative	3546
Separative Work Requirements	
Initial Core	0
Equilibrium Annual Reload	39.0
30-Year Cumulative	1151.5
Fissile Discharge, kg/TeM	
U-235	1.0
Fissile Pu	3.4
Annual Equilibrium Fissile Discharge, kg/Gwe	
U-235	53.3
Fissile Pu	186.5

Table II-2

Reactor Fresh and Spent Fuel Characterization

TYPE (General Description) Zr clad UO2 pellets (LEU) for HWR

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

Fuel Assembly Characteristics: (where applicable)

a) type: Oxide X; Metal ____; Carbide ____;

b) weight: 21.2 kg

c) length: 0.5 m

d) core load: 166056 mass (kg HM):

e) annual reload: 55250 mass (kg HM):

Design burnup: 19750 (MWd/MT) discharge batch average

Discharge fuel radiation level: 303 r/hour @ 1 meter
(also provide a curve of radiation level versus cooling time following discharge) Not available

Discharge fuel energy generation rate as a function of cooling time. (W/hr/element) - provide curve. 479 W/element after 90 days

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

ISOTOPE	Fresh Fuel Element		Discharged Fuel Element	
	initial	equilibrium	initial	equilibrium
Th-232				
U-232 U-233 U-234				
U-235 U-236	0.13	0.22	0.049 0.013	.018 .020
U-238	18.55	18.46	18.40	17.980
Np-237			.0002	.0014
Pu-238 Pu-239 Pu-240			.00001 .0448 .016	.0003 .058 .043
Pu-241 Pu-242			.003 .0007	.010 .007
Am-241 Cm-242				

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

TABLE II-3

U_3O_8 AND SWU REQUIREMENTS FOR HWRs AND LWRs

	30-Year Requirements	
	U_3O_8 (ST/GWe)	SWU (MT/GWe)
Conceptual LEU HWR, Once-Through Cycle	3546	1152
Standard LWR, Once-Through Cycle	6128	3632
Improved LWR, Once-Through Cycle	5196	3488

TABLE III-1

R&D REQUIREMENT - ONCE-THROUGH FUEL CYCLE

		Time (yrs)	Cost M$
I.	Slightly Enriched Fuel Development	5	3
	A. Safety Related Physics Parameters		
	B. Higher Burnup Test Irradiations		
II.	Advanced Pressure Tube Development	2	1
	A. Zirconium-niobium Tube Tests		
	B. Thermal-Hydraulic Tests		
III.	System Design	2	25-100

TABLE V-1

COST ESTIMATE SUMMARY
TWO DIGIT ACCOUNT LEVEL
1260 MWe PRESSURIZED HEAVY WATER REACTOR PLANT
MIDDLETOWN, USA

Cost Basis
1/78

ACCT. NO.	ACCOUNT DESCRIPTION	TOTAL COSTS (millions)
20	LAND + LAND RIGHTS	2.3
21	STRUCTURES + IMPROVEMENTS	124.6
22	REACTOR PLANT EQUIPMENT	210.6
23	TURBINE PLANT EQUIPMENT	127.9
24	ELECTRIC PLANT EQUIPMENT	48.7
25	MISCELLANEOUS PLANT EQUIPMENT	16.3
26	MAIN CON. HEAT REJECT SYS.	30.4
2	TOTAL DIRECT COSTS	560.6
91	CONSTRUCTION SERVICES	86.1
92	HOME OFFICE ENGRG. & SERVICE	58.3
93	FIELD OFFICE ENGRG. & SERVICE	35.5
9	TOTAL INDIRECT COSTS	179.9
	TOTAL BASE COST	740.5

TABLE V-2

ECONOMIC PARAMETERS

U_3O_8 ($/kg)	88
Separative Work ($/SWU)	80
Enrichment Plant Tails (w/o)	0.2
D_2O Cost ($/kg)	213
Plant Factor (%)	75
Plant Life (Years)	30
Fabrication ($/kg)*	60
Spent Fuel Disposal ($/kg)	115

*Fabrication costs for the HWR are different than for the LWR because of differences in HWR and LWR fuel element design.

TABLE V-3

FUEL CYCLE COST (mills/kw-hr)

	Reference HWR	
Core Price, $/lb	40	100
Fabrication	0.40	0.40
Fuel Disposal	0.77	0.77
U_3O_8	1.44	3.60
SWU	0.38	0.38
Carrying Charges	0.37	0.81
D_2O	2.38	2.38
Total Fuel Cycle Cost	5.74	8.34
O&M	0.98	
Capital Cost*	10.82	
Total Power Cost	17.55	20.15

*Based upon HWR capital costs (including IDC) of 711 $/kwe.

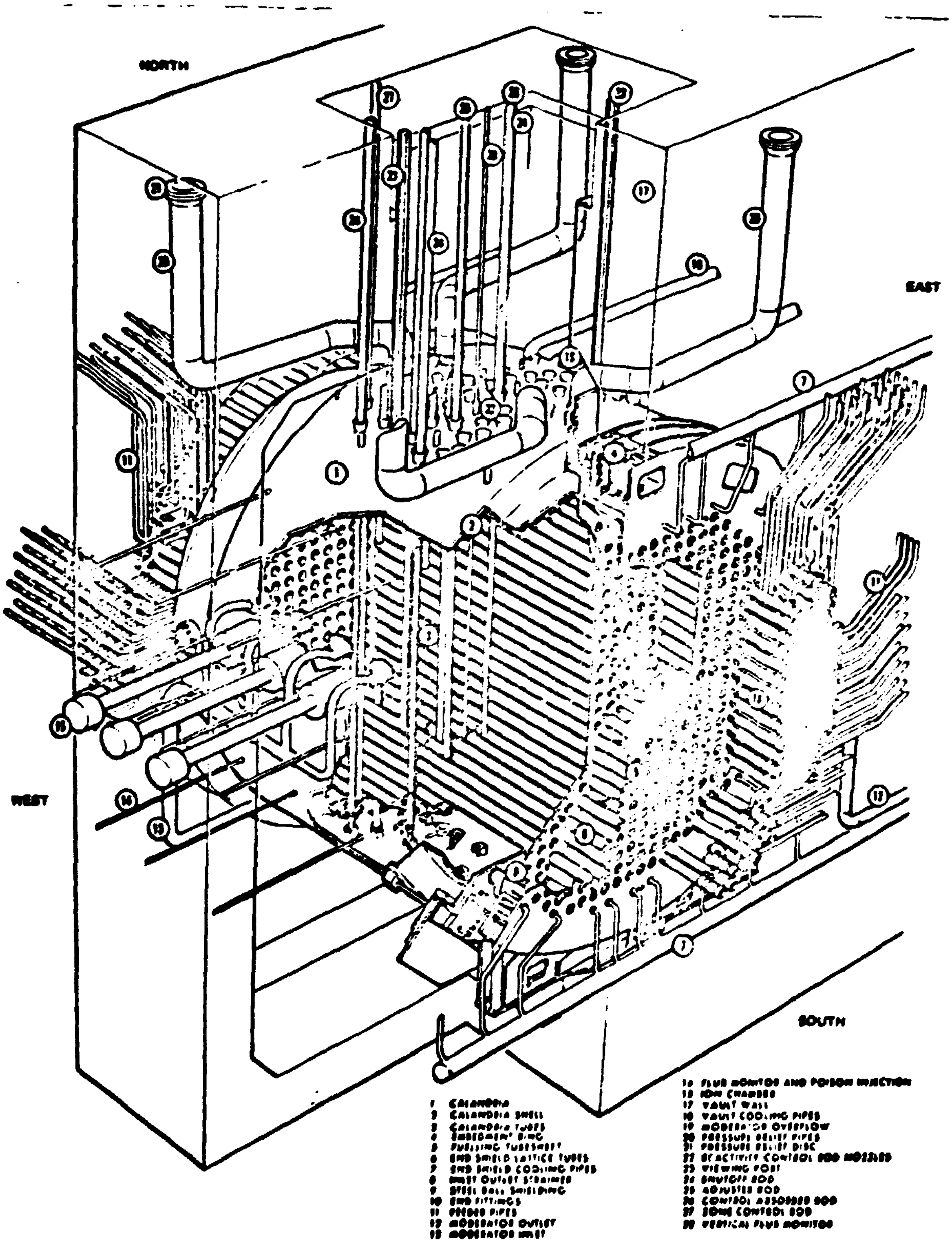


Fig.I-1 CANDU Reactor Assembly.
ANL Neg. No. 116-77-39.

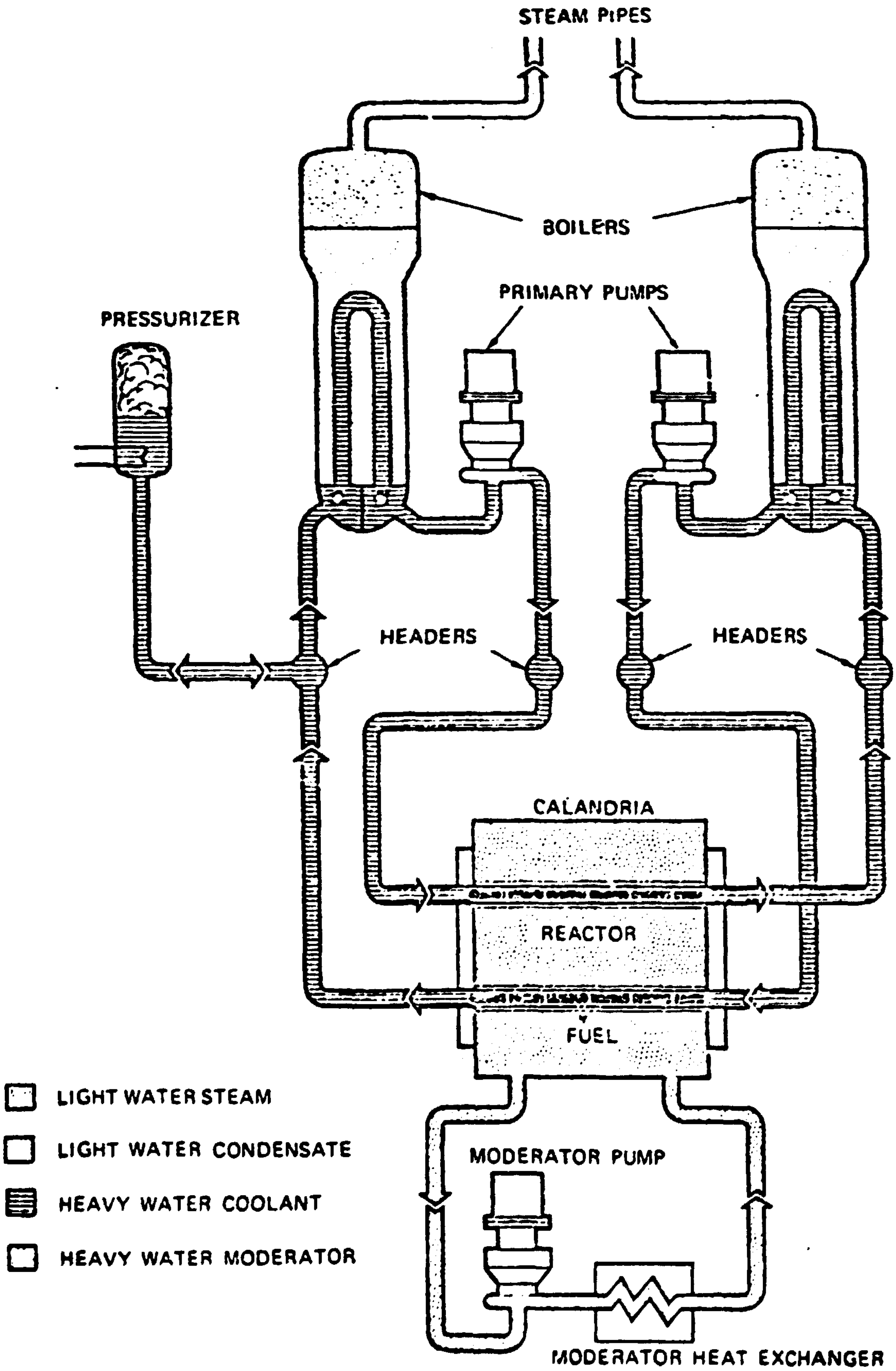
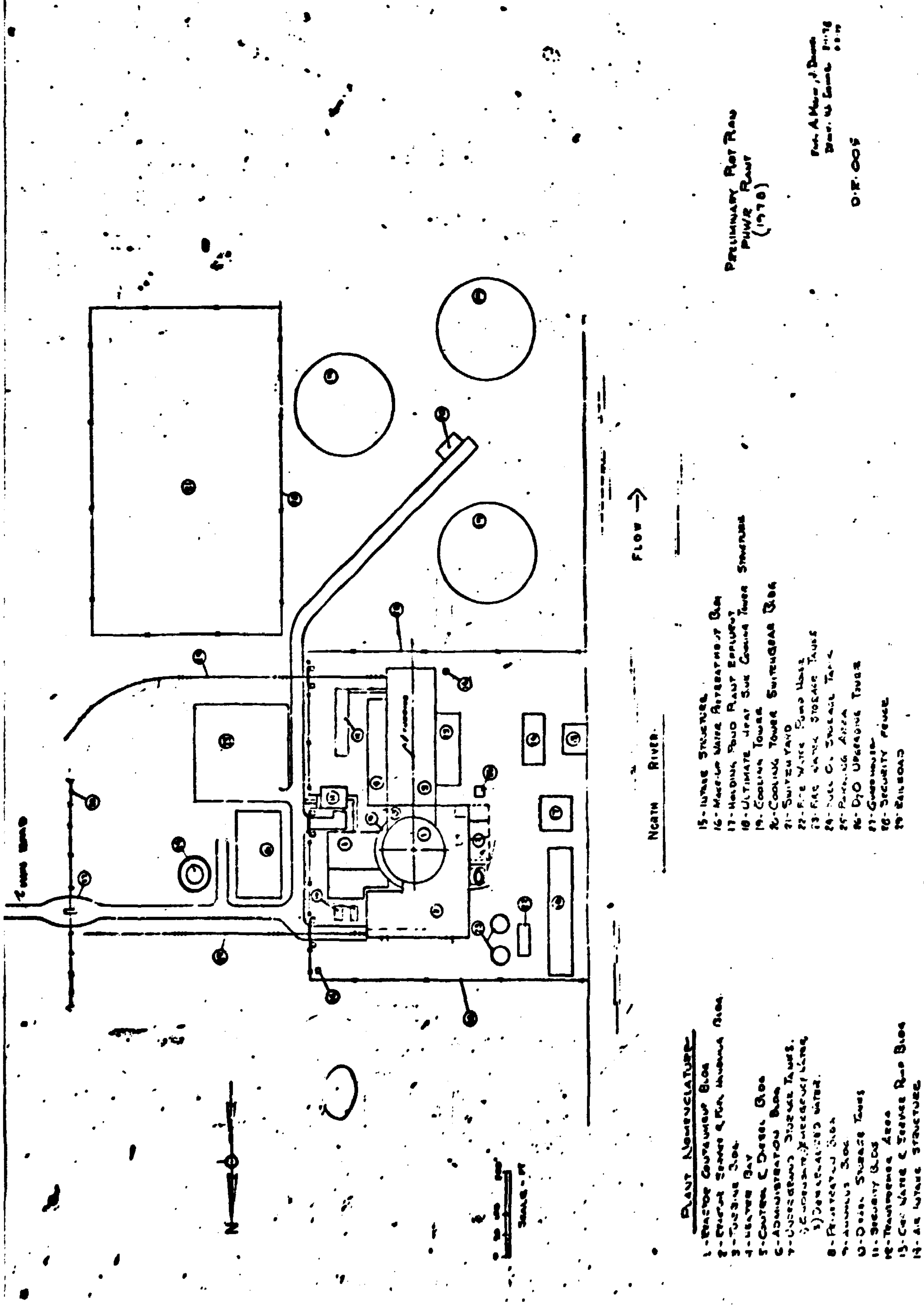


Figure I-2
Heavy Water Reactor Simplified Flow Diagram

Figure I-3



NORTH RIVER
FLOW
SCALE - FT
N
PLANT NOMENCLATURE
1-REACTOR CONTAINMENT BLDG
3-TURBINE BLDG
4-HEATER BAY
5-CONTROL & DIESEL BLDG
6-ADMINISTRATION BLDG
7-UNDERGROUND STORAGE TANKS.
9-ANNULUS BLDG
10-DIESEL STORAGE TANKS
11-SECURITY BLDG
12-TRANSFORMER AREA
14-AIR INTAKE STRUCTURE
15-INTAKE STRUCTURE
17-HOLDING POND PLANT EFFLUENT
18-ULTIMATE HEAT SINK COOLING TOWER STRUCTURE
19-COOLING TOWER
20-COOLING TOWER SWITCHGEAR BLDG
21-SWITCHYARD
23-FIRE WATER STORAGE TANKS
25-PARKING AREA
26-D2O UPGRADING TOWER
28-SECURITY FENCE
29-RAILROAD
PRELIMINARY PLOT PLAN
PHWR PLANT
(1978)
D-P-005

Figure I-4

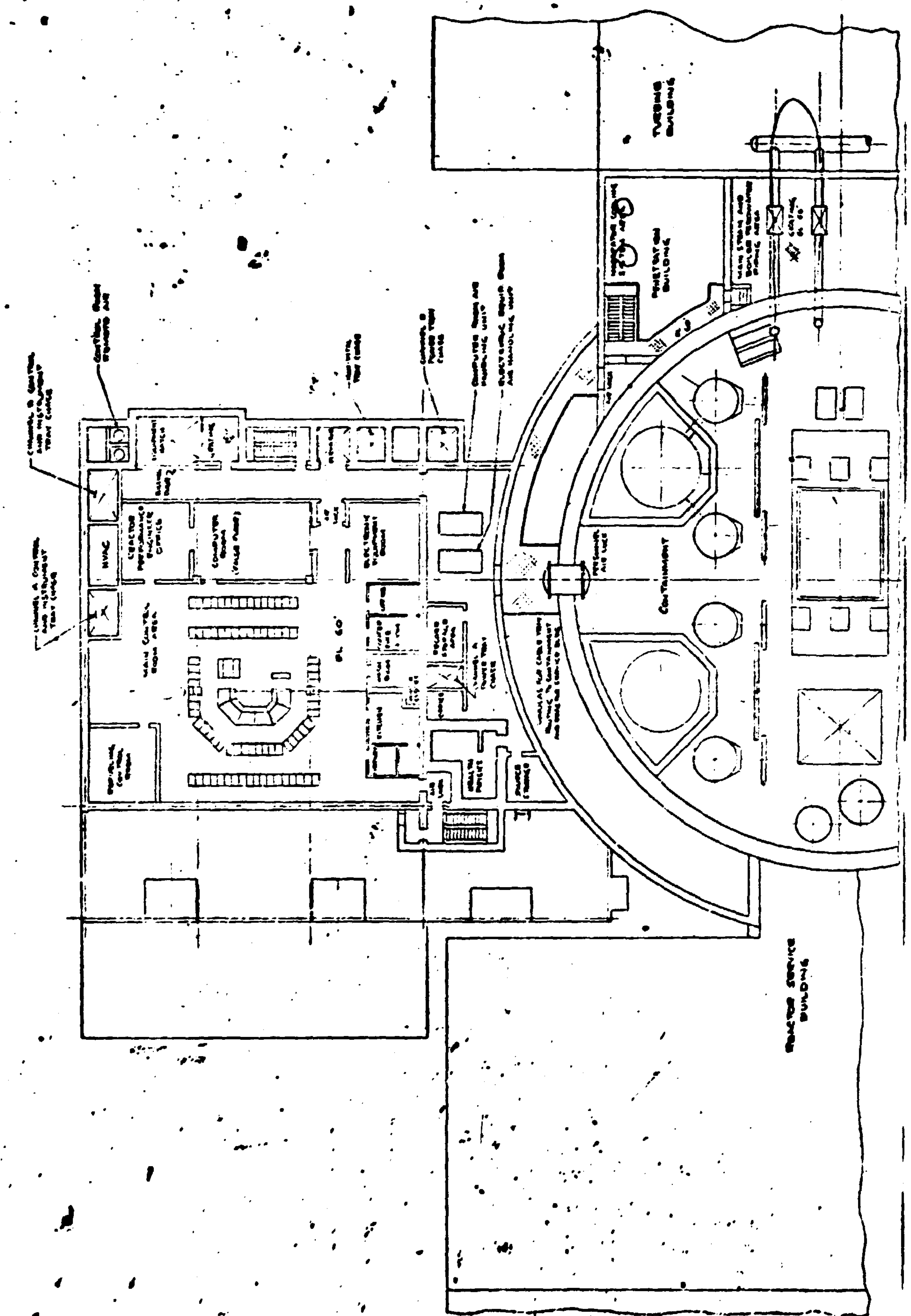

TURBINE BUILDING
CONTAINMENT
HVAC
REACTOR SERVICE BUILDING

Figure I-5

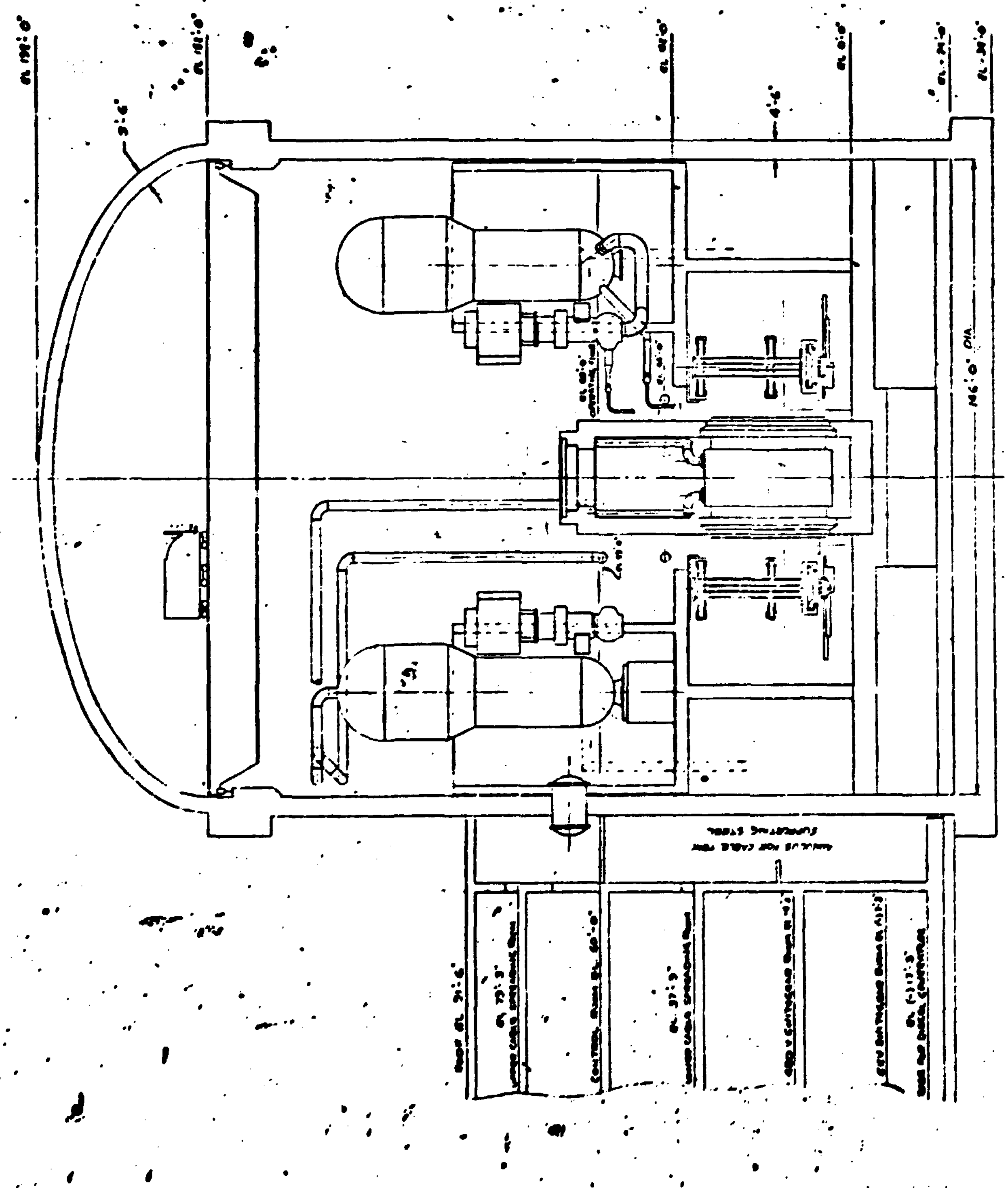


Figure II - 1
Fuel Cycle Facilities for
HWR LEU Once-Through Fuel Cycle

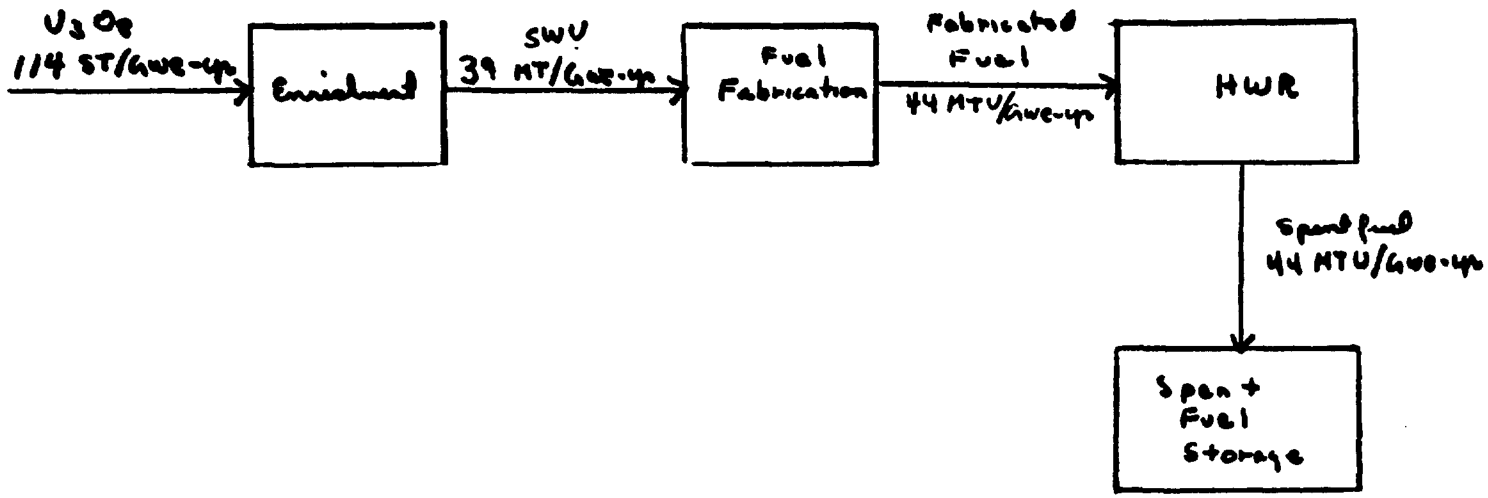


Figure V-1
HWR-LWR Breakeven U_3O_8 Price
vs HWR Capital Cost

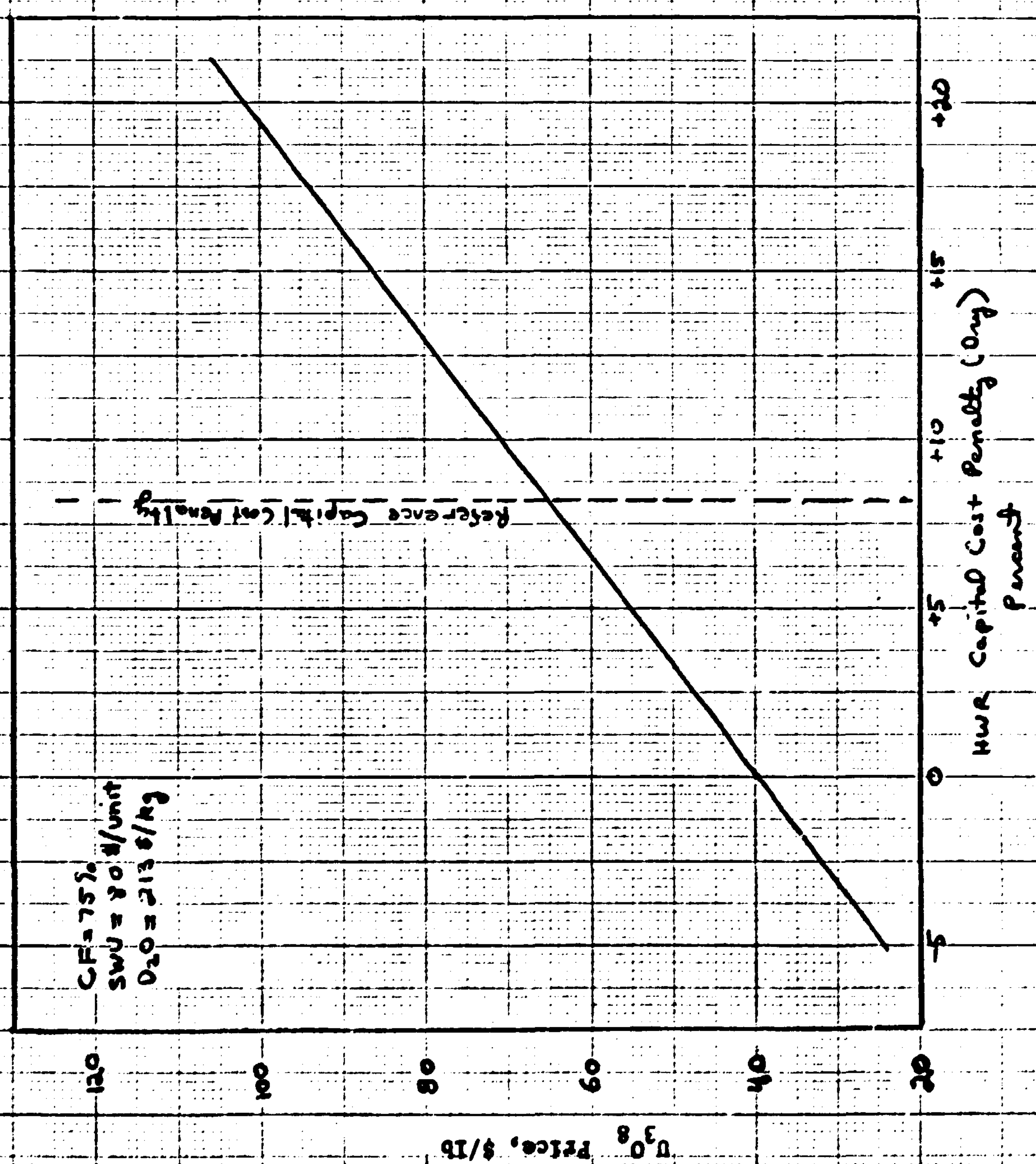


Figure V-2
HWR-LWR Breakeven U_3O_8 Price
vs Capacity Factor and SWU + D_2O Cost