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

By
N. L. Shapiro
J. F. Jesick

September 1979

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Combustion Engineering, Inc.
Windsor, Connecticut

TECHNICAL INFORMATION CENTER
UNITED STATES DEPARTMENT OF ENERGY

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Volume 2

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WATER REACTOR FOR U.S. SITING

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4. Optimization of HWR Design

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4. Optimization of HWR Design

4.0 Introduction

The overall purpose of the Heavy Water Reactor Characterization Program is to develop a reliable data base for HWRs for use by DOE in comparative assessments and evaluations as part of the Nonproliferation Alternative Systems Assessment Program (NASAP). The key criteria employed in developing this conceptual HWR design are:

- ° Technical feasibility, including environmental and safety factors.
- ° Uranium resource utilization.
- ° Economics and commercial feasibility.

Technical feasibility must be considered foremost, since it is clear that a reliable data base must be based upon a design which is technically feasible. Uranium resource utilization was considered to be the next most important criteria for the conceptual HWR design. The primary reason for considering alternatives to the light water reactor in the NASAP program is to identify reactor concepts and fuel cycles which achieve better uranium resource utilization so that the recovery and use of plutonium and the development of plutonium recycle will be either unnecessary, or can at least be delayed sufficiently far into the future that effective technical and institutional safeguards can be employed. Uranium resource utilization considerations are thus central to establishing the design parameters of the conceptual HWR, since it is the potential for improved uranium utilization which provides the motivation for considering the heavy water reactor alternative. Lastly, economic and commercial feasibility considerations are an essential ingredient in gaining acceptance and widespread deployment of the concept. Power cost economics of the conceptual HWR must

be comparable to those of other alternatives if the concept is to be commercially viable and the potential for improved uranium resource utilization realized.

In developing the conceptual HWR design, it was recognized that the factors identified above are not necessarily independent of each other. For example, alternatives achieving superior resource utilization might compromise economics, while design changes to improve resource utilization or power cost economics could raise questions of technical feasibility. Thus, the development of the conceptual HWR design represents, to some degree, compromises and trade-offs between these factors. Design changes which improve uranium resource utilization and/or power cost economics are the subject of this chapter. In evaluating these potential design changes, careful attention was paid to the question of technical feasibility, to ensure the acceptability of the resulting design and to identify R&D requirements resulting from these design changes.

In order to help ensure the technical feasibility of the conceptual HWR design, the Candu pressurized heavy water reactor (Candu-PHWR) was employed as a point of departure for the optimization study. 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 and Argentina and Korea. Extensive operating experience has been obtained for Candu reactors, particularly for the Pickering Station which contains four 500 MW(e) plants.

In contemplating design modifications to improve either uranium utilization or reduce plant capital cost, it was recognized that significant departures from present Candu designs might open questions of design feasibility, new licensing issues, and necessitate additional research and development which could delay introduction of the system. It was also recognized that a number of design modifications would have to be made to the Candu design for

siting in the U.S. in order to conform to U.S. licensing criteria and construction practices, and that further modifications would be required to increase the capacity of existing Candu designs to the size (approximately 1000 MWe) specified for this study. Care was taken to evaluate the feasibility of these modifications and to identify their potential impact on safety and licensability, RD&D requirements, and schedules for commercial deployment. Only those design modifications which resulted in significant improvements in uranium utilization and/or power cost economics were incorporated into the conceptual design.

In considering design changes which improve uranium utilization and power cost economics, it should be recognized that the Candu-PHWR has been designed to operate with on-line refueling using natural uranium fuel. The use of natural uranium fuel has had important ramifications to the design of the Candu concept. Because of the use of unenriched fuels, parasitic neutron losses to reactor structure and to the fuel element clad and hardware have been minimized, and on-line refueling is a practical necessity. In order to enhance the reactivity of the natural uranium fuel in Candu, a well thermalized spectrum is required, necessitating widely-spaced calandria tubes surrounded by large quantities of heavy water. The low discharge burnup obtained with natural uranium fueling (about 7,500 MWd/t) has resulted in fuel especially designed for low exposures.

Although the operation of heavy water reactors on the natural uranium cycle has been quite satisfactory, the resource utilization achieved with natural uranium fueling is only moderately better than that of the corresponding LWR once-through cycle. U_3O_8 requirements for the HWR operating with natural uranium fueling are about 20% lower than those of the LWR--about, 4,750 ST U_3O_8 for the HWR compared to 6,000 ST U_3O_8 for the LWR are required to operate a 1000 MW(e) plant for an assumed 30-year lifetime at a 75% capacity factor. Although this improvement in uranium utilization is certainly significant, it is questionable whether the reduction

in uranium requirements is sufficiently large to warrant development of a new reactor type for U.S. deployment. Consequently, for domestic use it is desirable to explore design modifications which might result in more significant improvements in uranium utilization; for foreign use this may not necessarily be the case since Candu's of current design have been extensively marketed overseas by the Canadians.

The capital costs (including D_2O) of the HWR generally are also evaluated to be significantly higher than the capital costs of the LWR (Reference 4-1) in the USA, resulting in higher HWR power costs. Consequently, the prospect for commercialization of the Candu-PHWR in the U.S. is not currently viewed as good. For this reason, the HWR optimization study also concentrated on evaluating potential design changes which could reduce plant capital costs.

Because of the impracticality of performing the optimization study for each of the many fuel cycle alternatives of potential interest, the optimization evaluation discussed below was limited to the evaluation of the effect of potential design changes on the performance of the once-through uranium cycle. The evaluation of the performance of other cycles of interest such as denatured thorium fueling, were then evaluated in the context of the reference design established on the basis of the optimization study performed for once-through uranium fueling. Optimizing on the basis of the once-through uranium fuel cycle has the advantage of ensuring optimal performance in the near term when this fueling mode is anticipated, and consequently of facilitating the introduction of the HWR; however, it is recognized that superior performance for the various recycle fuels could potentially be obtained by optimizing the HWR explicitly for these fuel types.

For operation on the once-through cycle, it is generally recognized that significant improvements in uranium resource utilization can be achieved by the use of slightly-enriched uranium fuel, (References 4-2 and 4-3). The use of slightly-enriched fuel is addressed in

Section 4.1 below, where the performance of the HWR from the standpoint of both U_3O_8 utilization and fuel cycle cost economics is evaluated. Sections 4.2 through 4.6 address various alternatives for HWR reactor and plant design. Since the Candu reactor is already well optimized from a neutron economy standpoint, due to the intended use of natural uranium fueling, these modifications to reactor and plant design are not anticipated to result in improvements in uranium resource utilization. Rather, these potential plant modifications, were explored primarily in an effort to reduce plant capital cost; in most instances, these modifications will actually result in some degradation of U_3O_8 utilization, and as a result the economic evaluations typically involve trade-offs between reduced capital and increased fuel-cycle costs. Sections 4.7 and 4.8 address a number of design problems specific to the use of slightly-enriched fuel, such as power peaking and the impact of higher discharge burnups on fuel rod design. Lastly, Section 4.9 discusses the various design considerations and trade-offs involved in the selection of parameters for the HWR reference design.

4.1 Fuel Enrichment Evaluation

4.1.1 Introduction

The effect of initial enrichment (burnup) on U_3O_8 consumption and power costs for the once-through, slightly-enriched uranium cycle in the standard Candu lattice are described in this section. As mentioned previously, the potential for improved U_3O_8 utilization is the prime motive for exploring slightly-enriched fueling. Slightly-enriched fueling also offers the potential for reduced fuel-cycle costs, both because of reduced uranium requirements and because of reduced total fuel fabrication and disposal costs which result from higher discharge exposures. The use of slightly-enriched fuel also makes more attractive a number of the design options discussed in subsequent sections of this chapter which serve to reduce plant capital cost; most of these design alternatives reduce cycle length somewhat, and would be unattractive for natural uranium fueling because of the already low discharge burnup.

4.1.2 Analytic Methods

The general approach used in the fuel-enrichment evaluation was to perform fuel-cycle calculations parametric in initial enrichment, and to establish the resulting discharge fuel burnup. These fuel-cycle calculations were performed for a cluster-cell geometry using the DIT-HWR code. The DIT-HWR code, described in greater detail in Appendix 7.1, is used to generate the cluster-cell spectra and few-group cross sections for heavy water reactor lattices. Space-dependent 85-group spectra are calculated in a one-dimensional representation of the cluster cell with an explicitly represented average fuel pin. These spectra are used for condensation of cross sections (ENDF/B-IV) to a four-group structure employed in the detailed spatial calculation. The cluster-cell

geometry is illustrated in Figure 4.1-1. The detailed spatial calculation is performed for the exact geometry of the HWR cluster, Figure 4.1-2, using integral transport theory. The resulting reaction rates are used for pin-by-pin depletion of the fuel. Output of the DIT-HWR code consists of reactivity, nuclide number density, and nuclide reaction rates as a function of irradiation time (burnup).

In batch-fuel reactors, such as the LWR, the achievable cycle length is established by consideration of the instantaneous core reactivity; that is, the cycle length is defined as that burnup at which neutron production just equals neutron losses ($k_{\text{eff}} = 1$) in the absence of control poisons. For reactors with on-power refueling, such as is assumed for the conceptual HWR design, burnups are limited by the time integral of reactivity rather than by the instantaneous value. Essentially, the assumption is made that although a fuel bundle may be a net producer or absorber of neutrons at any particular burnup, there is another bundle elsewhere in the reactor absorbing or producing enough neutrons to eliminate any net excess or deficit. This is equivalent to assuming that excess neutrons produced early in the cycle are saved for later in the burnup where the fuel produces less neutrons than are absorbed. The burnup limit is reached when this reserve of neutrons is depleted.

In the limit of continuous charge-discharge fueling, which closely approximates the on-power refueling of the HWR, burnup becomes limited when the total neutron losses from absorption and leakage equal the total neutron production, i.e., when

$$\int_0^B P dt = \int_0^B (A+L) dt \quad \text{Eq. 4.1-1}$$

where B is the discharge burnup, P is the neutron production, and A and L are neutron losses to absorption and leakage, respectively. This neutron balance equation can be approximated as

$$\frac{1}{B} \int_0^B \frac{1}{k^\infty} dB = \frac{1}{1+\Delta p} \quad \text{Eq. 4.1-2}$$

where k^∞ is the calculated reactivity and Δp is the reactivity worth of neutron losses from leakage and due to other considerations (for example, neutron losses in the absorber rods) excluded in the calculation of k^∞ .

The reactivity rundown calculated by the DIT-HWR code for the natural uranium fueled Candu-600 design is shown in Figure 4.1-3, while Figure 4.1-4 plots the left side of Equation 4.1-2 as a function of cycle length. After an initial drop due to the buildup of xenon, the reactivity of the natural uranium fueled system (c.f. Figure 4.1-3) initially rises slightly with burnup due to the buildup of plutonium. The difference in reactivity between fuel having a low exposure and that nearing the end of its useful lifetime (7,500 MWd/t) is relatively small, an attribute which helps to minimize power peaking when fresh fuel assemblies are added during refueling. The burnup integral (c.f. Figure 4.1-4) remains relatively constant for burnups less than 3,000 MWd/t, reflecting the small change in k^∞ over this burnup, and then increases monotonically as the burnup is increased further. The useful lifetime of the fuel is defined as the burnup at which this burnup integral equals the reactivity worth of neutron losses (right side of equation 4.1-2). These neutron losses, which total about 4% Δp , are also shown in Figure 4.1-4 and are constructed of the following items:

1. The worth of neutron losses from leakage, as calculated from the geometric buckling and neutron migration areas, i.e., $\Delta p_1 = 1 + B^2 M^2$. For a given reactor design the bucklings remain constant, while the migration areas are calculated for each fuel enrichment.
2. An allowance is made for the steel adjuster rods used to flatten the power distribution. An allowance of 18.5 mk, based upon Reference 4-4, is included to account for neutron losses to these adjuster rods. This allowance is shown as Δp_2 in Figure 4.1-4.
3. Allowances must also be made for minor parasitic absorptions in adjuster rod guide tubes, light water holdup in the zonal control system, and in-core detectors. An allowance must also be made for the fact that a discrete number of fuel assemblies are replaced each visit rather than the continuous charge-discharge mode assumed in the DIT-HWR calculations. As a result of these allowances, burnup is reduced by about 600 Mwd/TeM (Reference 4-4), equivalent to a reactivity adjustment, Δp_3 , of 0.41% Δp .
4. Including the reactivity losses identified in Items 1 through 3 above, results in a predicted HWR burnup of 7,350 Mwd/t. More detailed assessments of the burnup achievable in a standard Candu-600 design indicate a nominal cycle length of 7,500 Mwd/t (Reference 4-5). Consequently, a small reactivity adjustment, Δp_4 of 0.1% Δp or about 150 Mwd/t has been included to normalize the calculational results to the burnup quoted for the Candu-600 design. This small allowance represents the calculational bias between the best-estimate burnup and the predictions of the DIT-HWR code.

4.1.3 Fuel Enrichment Optimization Evaluation

In order to evaluate the effect of initial enrichment on uranium utilization, a series of DIT-HWR calculations parametric in initial enrichment were performed for the equilibrium cycle of operation. The reactivity rundown for various initial enrichments is shown and compared with that of natural uranium in Figure 4.1-5. Most noteworthy is the increase in the difference in reactivity between fresh and highly exposed fuel as the initial enrichment is increased. As a result of this reactivity difference, increases in core peaking should be expected as the initial enrichment is increased. The left side of equation 4.1-2 is plotted as a function of cycle length in Figure 4.1-6 for various initial enrichments. Cycle length is determined from this figure by applying the reactivity allowance discussed above for the natural uranium fueled system, with the exception that the leakage term (Δp_l) is calculated explicitly for each enrichment based upon the calculated migration area.

The resulting cycle lengths as a function of initial enrichment are plotted in Figure 4.1-7. Burnup is observed to increase approximately linearly with initial enrichment, with each additional 0.1 w/o of U-235 increasing burnup by about 2.4 MWd/kg. Also shown on this figure are AECL predictions of burnup as a function of initial enrichment (Reference 4-2). The discharge burnup predicted by DIT-HWR is about 10% higher than that quoted by AECL. Alternately, the DIT-HWR calculations predict that a given discharge exposure can be obtained with an initial enrichment somewhat lower than would be predicted by AECL calculations; for example, Figure 4.1-7 indicates that AECL calculations predict that an initial enrichment about 0.05 w/o higher than that predicted by the DIT-HWR calculations would be required to attain a discharge burnup of 20,000 MWd/t.

Relative U_3O_8 utilization is illustrated in Figure 4.1-8. In this figure, ore utilization has been normalized to unity for the natural uranium fueled system, and a diffusion plant tails assay of 0.2 w/o has been assumed. Figure 4.1-8 illustrates that U_3O_8 requirements decreased rapidly as the initial enrichment is increased from that of natural uranium to about 1 w/o, and that uranium utilization is rather insensitive to initial enrichment for initial enrichments greater than 1.2 w/o. The mathematical optimum occurs at an initial enrichment of about 1.5 w/o, where relative U_3O_8 requirements are about 62.5% of that of the natural uranium fueled system. Figure 4.1-8 also illustrates that the great majority of the improvement in uranium utilization possible with the use of slightly-enriched fuel is obtained for an enrichment of 1.2 w/o (U_3O_8 requirements are about 64.5% that of the natural uranium fueled system at this enrichment). As mentioned above, AECL calculations predict a discharge burnup about 10% lower than C-E results and consequently indicate a slightly smaller incentive for slightly enriched fueling. For example, AECL calculations indicate that the uranium requirements are reduced to about 70% of that of the natural uranium fueled system when an initial enrichment of 1.2 w/o is employed.

While U_3O_8 utilization can be increased slightly by employing an enrichment somewhat higher than 1.2 w/o, there are a number of incentives for keeping the initial enrichment as low as possible, consistent with obtaining the majority of the reduction in U_3O_8 requirements. First of all, the fuel assembly design currently employed in Candu reactors is designed for low burnups, and if the enrichment/burnup is increased to too high a value fuel assembly redesign would be required. In particular, present Candu fuel does not employ internal fission gas plenums; the fission gas release is accommodated within the free volume and porosity present in the fuel column. As discussed in Section 4.8, it appears possible to retain the existing Candu fuel assembly

design if burnups are limited to the range of 20,000 - 25,000 MWd/t; higher burnups would probably require modification of the fuel design to include internal fission gas plenums. While the development of HWR fuel containing internal fission gas plenums is certainly feasible, the use of such plenums is expected to increase power distribution peaking (because of the gaps in the fuel column caused by the plenums) and to increase fuel fabrication cost, and hence it is desirable to limit burnup to a value sufficiently low that such plenums are not required.

A second reason for choosing as low an enrichment as is practical is the increase in core peaking factor which results as initial enrichment is increased. The use of slightly enriched fuel increases both the local peaking factor (within a fuel assembly) as well as channel-to-channel and axial peaking factors. This increase in peaking factor is discussed in Section 4.1.4, while Section 4.7 discusses enrichment zoning, an option which partially offsets the increase in peaking factor. Increases in peaking factor necessitate reducing fuel channel power density in order to maintain thermal margins. As a result, the number of fuel channels in the reactor must be increased, resulting in an increase in plant capital cost. Thus, there is an incentive to limit the initial enrichment employed so as to minimize the necessity of reducing power densities.

The quantities of U-235 consumed during operation and discharged in spent fuel and the quantities of fissile plutonium produced, burned in situ, and discharged in spent fuel is shown in Figure 4.1-9. This figure illustrates that increasing initial enrichment (and hence cycle length) allows larger quantities of fissile plutonium to be produced and utilized in situ. U-235 is also depleted to lower concentrations as the initial enrichment and cycle length are increased. As a result of the more complete consumption of U-235 and of the larger contribution from plutonium

produced and consumed in situ, requirements for U-235 decrease as the initial enrichment (and burnup) are increased.

4.1.4 Economic Evaluation

Figure 4.1-10 presents the change in fuel cycle cost as a function of initial enrichment. These fuel cycle cost calculations were based upon equilibrium batch fueling using the economic parameters given in Table 4.1-1. In particular, note that the fabrication and spent fuel charges of \$60 and \$110/kg, respectively, have been employed independent of initial enrichment. Actual fuel cycle costs for natural uranium fueling are expected to be somewhat lower than indicated by this Figure since fuel fabrication and disposal costs should be somewhat lower than those assumed for slightly enriched fueling. As Figure 4.1-10 illustrates, fuel cycle costs decrease significantly (under the economic assumptions employed) as the enrichment is increased from natural uranium to 1.0 w/o. A rather broad minima in fuel cycle costs occurs for enrichments between 1.0 and 2.0 w/o; the optimal enrichment from a fuel cycle cost standpoint appears to lie between initial enrichments of 1.3 to 1.4 w/o, but an enrichment of 1.2 w/o achieves nearly equal fuel cycle costs. The optimum enrichment is rather insensitive to the price of U_3O_8 , although there appears to be a tendency for the optimal enrichment to shift to slightly lower values as the cost of uranium increases. Also shown on Figure 4.1-10 as the dashed line is the effect of uncertainties in discharge burnup on fuel cycle costs, calculated by increasing the initial enrichments to compensate for the difference in burnup predicted by the DIT-HWR and AECL calculations. As the dashed curve illustrates, the uncertainty in burnup has little impact on the selection of the optimal enrichment.

As noted above, fuel cycle costs decrease rapidly as the enrichment is increased from natural uranium to 1.0 w/o. The origin of this rapid decrease in fuel cycle costs is explained by Figure 4.1-11, which shows the contribution to the fuel cycle cost of two major

components: the cost of uranium (U_3O_8 and SWU) and the cost of fuel fabrication and disposal services. Figure 4.1-11 illustrates that the reduction in fuel cycle cost as initial enrichment is increased is due only partially to the reduction in the fuel cycle cost component attributable to the price of uranium; this component initially decreases because of the reduced U_3O_8 requirements for the slightly-enriched system, but eventually increases at higher enrichments as the rate of U_3O_8 savings decreases and the cost attributable to separative work increases. A more significant contribution to the reduction in fuel cycle costs is the reduction in the component due to fuel fabrication and fuel disposal. This component decreases as the initial enrichment is increased because higher discharge exposures are attained, and hence smaller quantities of fuel must be fabricated and disposed of each year.

The shape of the fuel cycle cost versus initial enrichment curve (Figure 4.1-10) is consequently sensitive to the assumed cost for fuel refabrication and spent fuel disposal. Low values for these fuel cycle service costs will tend to favor lower initial enrichments, while high fabrication and disposal costs will tend to result in higher optimal enrichments where, because of the increased discharge exposure, these costs are amortized over a larger energy production. The fabrication and spent fuel disposal costs employed in this analysis, 60 and \$115/kg, respectively, appear to be reasonable for slightly-enriched fuel (1.2 w/o) however, these costs appear high for natural uranium. With respect to fabrication costs, Canadian utilities have been purchasing fabrication services for about half the assumed \$60/kg fabrication cost, a cost differential which apparently results from the absence of accountability requirements associated with enriched fuel and the elimination of potential concerns of criticality during fabrication. Spent fuel disposal costs are also expected to be lower for natural uranium fuel, both because of the low reactivity of the spent fuel which will allow such fuel to be stored without regard for criticality and because of the lower decay heat burden of such low-exposure

fuel. Reduction of the cost for fuel fabrication and disposal services employed in the economic analyses would improve the perceived economics of the natural uranium enriched system and decrease the perceived incentives for the use of slightly enriched fuel. However, it appears that for any reasonable choice of fuel fabrication and disposal costs, the slightly-enriched system will achieve somewhat lower fueling costs while at the same time realizing the reductions in uranium requirements mentioned above.

4.1.5 Power Peaking Evaluation

In general, the use of slightly enriched fuel will increase core peaking. Increases in core peaking are observed in both the local peaking factor (i.e., within a fuel assembly) and in channel-to-channel and axial peaking factors.

The increase in local peaking factor is illustrated in Figure 4.1-12 where the ratio of the power in the fuel rod containing the highest power density to the assembly average power density is plotted as a function of initial enrichment. The local peaking factor increases monotonically with initial enrichment because of the increased flux depression which occurs at the center of the fuel assembly as enrichment is increased. As Figure 4.1-12 illustrates, an increase in local peaking factor of about 4% (compared to natural uranium) occurs for a 1.2 w/o initial enrichment, and still larger peaking penalties occur for higher enrichments. In general, the maximum peaking factor occurs at or near beginning of cycle, as is illustrated in Figure 4.1-13 which presents the evolution of the local peaking factor as a function of burnup for both the natural uranium fueled system and for a 1.2 w/o initial enrichment. As discussed in Section 4.7, enrichment zoning can be applied to reduce the local peaking factor of the 1.2 w/o enriched system to below that of natural uranium fueling; in fact, Section 4.7 indicates that local enrichment zoning can

provide about 4% additional margin to help mitigate the increase in channel-to-channel and axial peaking factors.

Increases in both channel-to-channel and axial peaking factors are anticipated as the initial enrichment is increased because of the greater disparity in reactivity between highly exposed and fresh fuel (Figure 4.1-5). A precise evaluation of the channel-to-channel and axial peaking factors requires detailed three-dimensional fuel management calculations in which each fuel assembly is represented explicitly; such detailed calculations were beyond the scope of this evaluation. However, some understanding of the impact of slightly enriched fueling on channel-to-channel and axial peaking factors can be obtained from more simple calculations in which only a limited number of fueling channels are represented.

Channel-to-channel peaking factors were evaluated in the simplified two-dimensional array shown in Figure 4.1-14. This calculation consisted of an eight fuel channel array which might be considered to represent a small region of the HWR core. In these calculations, fuel channel 1 contained fresh fuel while fuel channel 8 contained fuel of the highest exposures. The fuel exposures in each channel at the beginning of the calculation is specified in Table 4.1-2. After each depletion, the most highly exposed fuel was replaced with fresh fuel and the calculations repeated. Calculations were performed for both natural uranium and for 1.2 w/o initial enrichment so as to be able to compare the differences in power peaking. It should be noted that since this is a two dimensional calculation, axial power distribution effects are neglected. The results of this calculation are tabulated in Table 4.1-3. Peaking factors vary from cycle to cycle due to the fuel management (i.e., due to the various juxtapositions of fuels of differing exposure), but the peak power densities for the slightly enriched case are typically 40 to 70% higher than those for natural uranium.

As mentioned above, these calculations employed a uniform axial fuel distribution. In practice, Candu reactors employ partial channel refueling in which only a part of the fuel which resides in each channel is replaced during a refueling visit. In this refueling mode, fresh fuel is initially located in the low power axial regions of the core where the higher channel-to-channel peaking factor will have less of an impact. During subsequent refueling visits, the most recently charged fuel would be moved into the high power interior region of the core; however, channel-to-channel peaking factors are reduced by this time because of depletion effects. Thus, one would anticipate that the actual fuel management practices employed for Candu reactors would result in somewhat lower peaking factors than those mentioned above.

The effect of partial channel refueling was evaluated using the simplified three-dimensional representation of a section of the HWR core illustrated in Figure 4.1-15. As is the practice in Candu reactors, bi-directional refueling was employed in these calculations, i.e., fresh fuel is first introduced into half the fuel channels from the left (channels labeled "A" in Figure 4.1-15), while the remaining half of the fuel channels (labeled "B") are fueled from the right. In the first series of calculations, 8 fuel bundles (out of the 12 in the channel) were replaced at each refueling visit. This mode of operation is typical of that currently employed in natural uranium fueled Candu reactors. The results of this calculation are shown in Figure 4.1-16 which shows the three dimensional peak as a function of axial length for both the natural and 1.2 w/o fueled systems. As this figure illustrates, the three-dimensional peak is about 60% larger for the slightly-enriched-fueled case, as compared to natural uranium fueling. Figure 4.1-17 shows the results of a similar calculation in which two of the 12 fuel assemblies which reside in each channel are replaced during a refueling visit. This mode of

operation is impractical with natural uranium fueling, since the frequency of refueling visits becomes excessive when so few assemblies are replaced during each visit. However, replacing only two of the twelve fuel bundles at each refueling visit becomes practical when slightly enriched uranium fuel is employed because of the higher reactivity of each fuel assembly. With two-bundle refueling, the three-dimensional peak is reduced to a value of about 1.58, a value about 20% higher than that achieved using the normal refueling scheme with natural uranium fuel.

Although more detailed fuel management calculations, in which the full HWR core is explicitly represented, will be required to definitively assess the power peaking implications of slightly enriched fueling, it appears that the increase in power peaking can be held to within about 15% of that of natural uranium fueling provided local enrichment zoning is employed and careful attention is paid to the fuel management. This increase in power peaking necessitates that the number of fuel channels be increased so as to preserve thermal margins.

The effect on capital cost of increasing the number of channels to accommodate increased peaking is shown in Figure 4.1-18. The change in capital cost as the number of pressure tubes is varied results almost solely from the change in heavy water inventory, since the effect on calandria cost is very small and since increasing the number of pressure tubes does not significantly affect containment sizing because the height of containment is limited by the size of other components (steam generators). As Figure 4.1-18 illustrates, the effect of increasing the number of pressure tubes is rather modest, so that the effect of higher peaking can be accommodated by increasing the number of pressure tubes without substantially impacting capital costs. Fuel cycle costs will also increase somewhat when the number of pressure tubes is increased, because of carrying (interest) charges on the additional fuel loadings.

Thus, while it appears that increases in peaking can be accommodated by increasing the number of fuel channels without large penalties in overall power costs, nevertheless, there remains an incentive for utilizing as low an enrichment as is practical.

4.1.6 Conclusions

The conclusions reached in the evaluation of slightly enriched fuel in the heavy water reactor are as follows:

1. U_3O_8 requirements can be reduced to about 65% of that of the natural uranium fueled HWR by employing slightly enriched uranium fuel.
2. The majority of this improvement in uranium utilization occurs as the enrichment is increased from that of natural uranium to about 1.0 w/o; a broad minimum in uranium requirements is reached for enrichments between 1.2 and 2.0 w/o, over which there is very little change in uranium requirements.
3. Limiting the initial enrichment to 1.2 w/o or less is desirable since it avoids the necessity of including internal fission gas plenums to accommodate fission gas release; for higher enrichments such plenums would be required. Core peaking factors also increase with enrichment, providing a second incentive for using as low an enrichment as is practical.
4. Fuel cycle cost calculations also exhibit a broad minimum between an enrichment of 1.2 and 2.0 w/o.
5. Under the economic assumptions employed, the use of slightly-enriched fuel results in a significant reduction in fuel-cycle costs (about 2.4 mills/kw hr). This reduction in fuel cycle costs results primarily from the decrease in fuel fabrication and spent fuel disposal requirements.

6. The perceived fuel cycle cost incentive for slightly-enriched uranium fueling (item 5) depends strongly upon the economic assumptions employed for the cost of fabrication and spent fuel disposal for natural uranium fuel. However, slightly enriched fueling would appear to show some economic incentive for any reasonable cost assumptions.
7. An initial enrichment of 1.2% appears to be near optimal from a uranium resource, fuel cycle economics, and reactor and fuel performance standpoint. The great majority of the potential gains in uranium resource utilization improvement and in the reduction of fuel cycle costs is achieved at this enrichment; only very small additional gains can be realized by increasing the initial enrichment further. Limiting the initial enrichment to 1.2 w/o avoids the necessity of redesigning the fuel elements to contain the fission gas plenums which would be necessary to accommodate the higher burnup associated with enrichments above 1.2 w/o. Limiting the initial enrichment to 1.2 w/o also minimizes the increase in core power distribution peaking.
8. More detailed fuel management calculations should be performed in order to more precisely evaluate the increase in core peaking anticipated with slightly enriched uranium fueling. Explicit three-dimensional calculations are required in which each fuel channel and fuel assembly are represented.

TABLE 4.1-1

ECONOMIC PARAMETERS

Separative Work (\$/SWU)	80
Enrichment Plant Tails (wt%)	0.2
D ₂ O Cost (\$/kg)	213
Plant Factor (%)	75
Plant Lifetime (years)	30
Fabrication Cost (\$/kg)	60
Spent Fuel Disposal (\$/kg)	115

TABLE 4.1-2

BURNUP DISTRIBUTION FOR TWO-DIMENSIONAL
POWER PEAKING EVALUATION
(AT START OF FIRST DEPLETION STEP)

<u>Fuel Channel</u>	<u>Burnup (MWD/TEM)</u>	
	<u>Natural Uranium</u>	<u>1.2 wt% U-235</u>
1	50	50
2	784	4480
3	1484	8880
4	2398	10177
5	2997	14366
6	3907	15181
7	4582	17842
8	5365	19463

TABLE 4.1-3

EIGHT CYCLE (UNIFORM AXIAL)
RADIAL POWER DISTRIBUTION (EQUILIBRIUM)

Natural Uranium

	Fuel Channel							
Cycle	1	2	3	4	5	6	7	8
1	.992	.954	.980	<u>1.060</u>	1.013	1.056	1.041	.902
2	.952	1.054	<u>1.067</u>	.944	1.064	.973	.908	.988
3	1.998	1.038	<u>1.164</u>	1.892	1.133	.814	.955	1.006
4	1.030	1.028	.932	<u>1.095</u>	.856	1.038	.994	1.027
5	1.032	1.038	<u>1.124</u>	.818	1.073	.872	1.000	1.040
6	1.036	1.006	.822	1.068	.871	<u>1.139</u>	1.038	1.022
7	1.045	.900	.930	1.034	.970	1.065	<u>1.074</u>	.982
8	.916	.981	.986	1.033	1.022	<u>1.036</u>	.992	.986
9	.993	.954	.980	<u>1.060</u>	1.013	1.056	1.041	.902

TABLE 4.1-3 (Cont.)

1.2 wt% U-235

Cycle	Fuel Channel							
	1	2	3	4	5	6	7	8
1	<u>1.502</u>	1.180	1.176	.891	.940	.754	.848	.708
2	1.029	1.359	.936	.856	.750	.790	.590	<u>1.690</u>
3	1.366	.834	.776	.865	.687	.756	<u>1.632</u>	1.082
4	1.084	.757	.368	1.174	.323	<u>2.010</u>	1.262	1.022
5	.962	.752	1.116	.388	<u>1.992</u>	.646	1.287	1.715
6	.773	.744	.362	<u>2.058</u>	.602	1.686	.952	.822
7	.779	.681	<u>2.010</u>	.662	1.672	.526	.872	1.596
8	.562	<u>1.652</u>	1.281	1.281	1.052	1.071	.790	.964
9	<u>1.502</u>	1.180	1.176	.892	.940	.754	.848	.708

NOT TO SCALE

REGION

- 1 AVERAGE FUEL ROD
- 2 CLAD-GAP
- 3 COOLANT
- 4 HOMOGENIZED FUEL + CLAD + COOLANT
- 5 PRESSURE TUBE
- 6 VOID
- 7 CALANDRIA TUBE
- 8 MODERATOR

Figure 4.1-1
GEOMETRY FOR SPECTRUM CALCULATION

REGION

- 1 FUEL
- 2 CLAD-GAP
- 3 COOLANT
- 4 PRESSURE TUBE
- 5 VOID
- 6 CALANDRIA TUBE
- 7 MODERATOR

NOT TO SCALE

Figure 4.1-2

GEOMETRY FOR SPATIAL DEPLETION CALCULATION

Figure 4.1-3
REACTIVITY RUNDOWN FOR NATURAL URANIUM FUELING

Figure 4.1-4
BURNUP INTEGRATED REACTIVITY FOR NATURAL URANIUM FUELING

Figure 4.1-5
REACTIVITY vs BURNUP FOR VARIOUS INITIAL ENRICHMENTS

Figure 4.1-6

BURNUP INTEGRATED REACTIVITY FOR VARIOUS INITIAL ENRICHMENTS

Figure 4.1-7
COMPARISON OF C-E AND AECL PREDICTIONS OF DISCHARGE BURNUP

Figure 4.1-8
RELATIVE URANIUM UTILIZATION
AS A FUNCTION OF INITIAL ENRICHMENT

Figure 4.1-9
FISSILE PRODUCED, BURNED AND DISCHARGED

Figure 4.1-10
CHANGE IN FUEL CYCLE COST WITH INITIAL ENRICHMENT

Figure 4.1-11
FUEL AND FUEL CYCLE SERVICE COST COMPONENTS AS A FUNCTION
OF INITIAL ENRICHMENT

Figure 4.1-12
ASSEMBLY PEAK-TO-AVERAGE POWER DENSITY

Figure 4.1-13
EVALUATION OF LOCAL PEAKING FACTOR WITH BURNUP

Figure 4.1-14
GEOMETRY FOR TWO-DIMENSIONAL POWER
PEAKING EVALUATION

Figure 4.1-15

GEOMETRY FOR THREE DIMENSIONAL POWER PEAKING EVALUATION

Figure 4.1-16
AXIAL POWER DISTRIBUTION EQUILIBRIUM, BI-DIRECTIONAL, EIGHT FUEL BUNDLES
OF TWELVE REPLACED AT ONE TIME

Figure 4.1-17

AXIAL POWER DISTRIBUTION (EQUILIBRIUM) BI-DIRECTIONAL, TWO FUEL BUNDLES
OF TWELVE REPLACED AT ONE TIME

Figure 4.1-18
CHANGE IN CAPITAL COST vs NUMBER OF PRESSURE TUBES

4.2 Primary System Pressure Evaluation

4.2.1 Introduction

The capital cost contribution to total power costs is related directly to the capital cost expressed in terms of \$/kwe. The \$/kwe capital cost of a plant depends both upon the total plant cost (in dollars) and upon the net electrical power output which can be generated from that plant. Although the cost of a number of components which comprise the total plant cost are proportional to the electrical output (for example, the cost of turbine generator), the majority of the cost components are proportional to the thermal output of the plant, since it is the thermal output which determines such important plant parameters as reactor size, containment size, steam generator size, etc. Consequently, plant capital costs (\$/kwe) will depend upon the efficiency with which thermal power can be converted into electric power, and hence plant capital costs can be reduced by improving the net electrical efficiency. The standard Candu 600 reactor operates at a net electrical efficiency of about 29.3%, compared to the 33% efficiency achieved by the PWR.* The Candu reactor will therefore generate about 12% less electrical energy than a PWR designed to have an identical thermal power output. Consequently, one would expect the Candu reactor to have a 12% higher capital cost than the PWR due to the lower net operating efficiency.

HWR efficiencies are lower than those of the PWR for two reasons: because of the unproductive loss of energy to the moderator, and because of the lower primary and secondary operating pressures and temperatures. The loss of energy to the moderator, about 6% of the total energy produced, is an inherent feature of pressure tube reactor design and little can be done to reduce this loss of

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

efficiency. The reduced efficiency resulting from the lower operating pressures and temperatures of the HWR can, however, be eliminated by increasing coolant conditions to those comparable with the PWR, so that the thermal energy deposited in the coolant can be converted to electrical energy with an efficiency comparable to that of the PWR. The efficiency with which thermal energy deposited in the coolant can be converted to electrical energy depends primarily upon the operating pressure of the secondary system, and hence equivalent efficiencies can be obtained by increasing the secondary pressure from the 4.7 MPa (700 psia) currently employed in Candu to a value comparable to the 7.7 MPa (1100 psia) of modern PWRs. Such increases in secondary pressure, however, necessitate increasing primary coolant outlet temperatures from the 266°C currently employed to 299°C, and necessitates an increase in primary system pressure to a value comparable to that of the PWR, 15.5 MPa (2,250 psia) in order to preserve core thermal margins (critical heat-flux-ratio).

Thus, improvements in HWR efficiency can be obtained by increasing secondary temperatures and pressures, but the primary system design must be modified to accommodate the resulting increase in primary system pressures and temperatures. In particular, it will be necessary to increase the thickness of the pressure tubes in order to maintain the same degree of conservatism in pressure tube design under the modified primary operating conditions. This increase in pressure tube thickness will adversely affect the burnup which is achievable, and could potentially decrease uranium utilization and increase fuel cycle costs. The effect of primary system pressure on net electrical efficiency and plant capital cost is discussed in Section 4.2.2. The impact on fuel utilization of changes in pressure tube thickness necessary to accommodate higher operating pressures is discussed in Section 4.2.3, while Section 4.2.4 addresses pressure tube design considerations for operation at higher operating temperatures and pressures.

The effect of primary and secondary conditions on net electrical efficiency is illustrated in Table 4.2-1. The first line of this table (1600 psia) corresponds to the primary and secondary conditions of the Candu-600 reactor, while the last line of this table (2250 psia) represents primary and secondary conditions which are characteristic of modern PWRs. As the primary pressure is increased from 1600 to 2250 psia, pressure tube thickness must also be increased. The second column of Table 4.2-1 gives the required pressure tube thickness, calculated to maintain a constant value of hoop stress at the various primary system pressures. The third column shows the primary system coolant outlet temperature which can be obtained at each primary pressure without altering the thermal margin present in the reactor. Secondary system pressure, shown in the fourth column of Table 4.2-1, was then established so as to obtain the coolant outlet temperature specified in the previous column. The resulting net electrical efficiencies are tabulated in the last column of Table 4.2-1.

As Table 4.2-1 indicates, increasing primary system pressure from 1600 to 2250 psia allows the net electrical efficiency to be increased from 29.3% to 31.3%. The resulting 2.10 percentage point gain in net electrical efficiency corresponds to a 6.8% increase in the amount of electrical energy which can be generated from a given thermal reactor power output.

The effect of primary system pressure (i.e., of net electrical efficiency) on plant capital cost is shown in Figure 4.2-1. This figure illustrates that plant power cost can be reduced by as much as 0.9 mills/kWh by increasing primary system pressure to 2250 psia. The percentage reduction in power cost is roughly equal to the percentage improvement in net electrical efficiency.

4.2.3 Fuel Resource Evaluation

In order to evaluate the effect on uranium resource utilization of modifications to increase primary and secondary system pressure, a series of equilibrium cycle calculations were performed for pressures varying between the 1600 psi of the standard Candu 600 reactor and the 2,250 psi pressure characteristic of PWRs. In this series of calculations, the pressure tube thickness was increased as the primary system pressure was increased so as to maintain the same hoop stress. The design and anticipated performance of pressure tubes at elevated temperatures and pressures is discussed in greater detail in Section 4.2.4. Coolant temperatures were also increased so as to preserve the same exit quality present in the reference Candu system. Pressure tube thickness and coolant temperatures are tabulated in Table 4.2-1 for each of the primary system pressures investigated.

The results of these calculations are summarized in Table 4.2-2 where the discharge burnup achievable with various amounts of initial enrichment are given for various primary system pressures. The achievable discharge burnup and the resulting relative U_{308} utilization is also shown in Figure 4.2-2 as a function of initial enrichment for the standard Candu operating pressure of 1600 psi and for the highest operating pressure evaluated, 2250 psi; other pressures would lie intermediate to the points shown on this graph.

Table 4.2-2 illustrates that the effect of increasing primary system pressure and pressure tube thickness is to reduce the achievable burnup by an amount which is roughly independent of initial enrichment (the burnup penalty is actually somewhat higher for low enrichments than for high, decreasing from a penalty of about 1200 MWd/t for natural uranium to about 1,000 MWd/t for an enrichment of 1.6 w/o). This burnup penalty represents

a rather substantial fraction of the achievable burnup in the natural uranium fueled system, 16%, but a much smaller fraction of the burnup achievable with slightly enriched uranium fueling. For example, the 1000 MWd/t penalty observed for an initial enrichment of 1.2 w/o represents only a 4.4% reduction in achievable burnup. Consequently, the impact of increasing pressure tube thickness is much smaller when slightly enriched fueling is employed.

Figure 4.2-2 also shows the relative U_3O_8 utilization ($ST\ U_3O_8/GW(e)$) as a function of initial enrichment. This relative U_3O_8 utilization is normalized to unity for the standard Candu-PHWR operating with natural uranium fueling. For natural uranium fueling, the increase in pressure tube thickness concomitant with the increase in primary system pressure to 2250 psi results in an increase in U_3O_8 requirements slightly in excess of 10%. This penalty results from the 16% reduction in achievable burnup, which is only partially compensated for by the increase in net electrical efficiency. As the initial enrichment and discharge burnup increases, the burnup penalty becomes a smaller and smaller fraction of the achievable cycle length, and for enrichments greater than about 1.1 w/o, the improvement in electrical efficiency proves to outweigh the penalty in burnup. For enrichments higher than this value, the relative uranium utilization is actually somewhat better than is obtained using the standard Candu system pressure and net electrical efficiency.

4.2.4 Pressure Tube Design Considerations

The proposed heavy water reactor is based on the Candu concept but would be uprated to operate with enriched fuel and higher primary temperatures, pressures and neutron fluxes than in the present Candu design. These more severe operating conditions for a given pressure tube design can lead to an acceleration in axial

and diametral straining, as well as oxidation and embrittlement. In the following estimates are made the importance of these properties for three alloys: Zircaloy-4, cold worked Zr-2.5% Nb and cold worked and aged Zr-2.5% Nb. The available data suggests that the design temperature and pressure may be limited by diametral creep of the pressure tubes. Pressure-tube sag, due to the combined effects of creep and channel weight must also be taken into consideration in detailed design. Because diametral creep is so important, the strongest material considered, cold worked Zr-2.5% Nb, is recommended for the pressure tube. Nevertheless, more information on creep behavior of this alloy in this temperature range (300-350°C) is required before accurate service life predictions can be made.

1. Oxidation: Irradiation enhances oxidation of zirconium alloys if high oxygen concentrations exist in the coolant. Figure 4.2-3 illustrates this effect for the three alloys considered (Reference 4-6). Based on these data and a correlation for Zircaloy by Hillner (Reference 4-7), less than 4 mils of oxide are expected after 20 full power years of service at 651°F.

Oxidation of zirconium is accompanied by the evolution of hydrogen that may be absorbed in the tube wall. For the alloys considered, approximately 10% of the hydrogen is absorbed (Reference 4-8). This fraction is believed to be independent of irradiation flux. Thus, for a 4 mil oxide thickness on a 220 mil pressure tube, a hydrogen concentration of about 560 ppm by weight can be expected after 20-years at full power. Based on Figure 4.2-3, Zr2.5% Nb and Zircaloy oxidize at comparable rates and will contain similar amounts of hydrogen. The importance of the hydrogen concentration on mechanical properties will be discussed in a later section on embrittlement.

2. Axial Expansion: Pressure tubes will lengthen in-reactor due to the combined effects of axial creep and stress free irradiation growth. Ross-Ross and Fidleris (Reference 4-9) predict that the ratio of axial creep to diametral creep is approximately 0.22 for the present Candu design.* If the proposed reactor retains the 3% diametral creep limit (which is discussed in a following section) about 0.7% axial creep elongation due to creep alone is expected.

Zircaloy-2 pressure tubes from four vendors in the Hanford N reactor (Reference 4-10) have undergone 0.1 to 0.2% axial (stress-free) growth in 50,000 hrs in a peak fast flux of 2.2×10^{13} (n/sec.-cm²). Growth will be influenced by cold work and texture, but little effect of either temperature or alloying is expected (Reference 4-11). Assuming that Zr-2.5% Nb and Zircaloy behave similarly and that growth depends linearly on fluence, 0.4 to 0.8% stress-free axial growth of the pressure tube may occur over 20 years at a flux of 3×10^{13} (n/cm²-sec.). This must be added to axial creep to obtain the total elongation. A better estimate of growth may be obtained as Zr-2.5% Nb growth data from present Candu reactors become available.

The creep - limit guideline presently used in Canada is 5%. This limit is imposed by the calandria structural design. A Yoke-nut arrangement allows for some axial elongation.

3. Embrittlement: Pressure tubes will embrittle as a result of fast neutron irradiation and hydrogen pickup during in-reactor oxidation. However, from a critical crack length standpoint, zirconium alloys retain a significant amount of ductility. Figure 4.2-4 shows how the critical crack length for cold worked Zr-2.5% Nb pressure tubes varies with hoop stress for hydrogen concentrations ranging from less than 25 ppm to

*The most recent approximation for the ratio of axial-to-diametral creep is 0.70 ± 0.18 (Reference 4-19).

greater than 200 ppm (Reference 4-12). Data from tubes with exposures greater than 10^{20} (n/cm²) are also included in this comparison. The graph indicates that the critical crack length will remain much larger than the tube wall thickness for the lifetime of the pressure tubes. Such large cracks are easily detected and consequently should not pose a safety problem from the standpoint of sudden bursting of the pressure tube. High hydrogen level data necessary to apply this conclusion over 20-full power years could be obtained by inspection and testing of Zr-2.5% Nb tubes already in use in Candu systems.

Delayed hydrogen cracking is a mechanism that may have initiated cracks and failures in some pressure tubes in the Pickering reactors (Reference 4-13). Cracks appeared in rolled joints having residual stresses on the order of 700 MPa (Reference 4-14). Hydrogen concentrations greater than 70 ppm (Reference 4-13) are believed necessary for this mechanism to operate. This problem of delayed hydrogen cracking was essentially eliminated by modifying the rolling process such that the residual stresses in the rolled joints were reduced (References 4-13 and 4-14). In the proposed reactor, pressure tube walls will be thicker than in the present Candu reactor. In order to ensure acceptable residual stresses modification of the current rolling procedure may be required. This is not expected to be a major problem.

4. Diametral Creep of Pressure Tubes: The design limit for pressure tube diametral creep is determined by the clearance between the garter-spring spacers and the pressure tubes. The garter-spring spacers are located in the annular space between the calandria tube and the pressure tube. Lockup of these spacers could cause surface damage to both tubes and reduce stress resistance. In the conceptual design, a limit on diametral creep of 3% is employed, which is the amount of strain required for contact with the "cold" calandria tube.

One correlation of irradiation creep data obtained at temperatures less than 300°C yields the expression (Reference 4.9):

$$\dot{\epsilon} = \kappa \phi \sigma (T-160) \quad \text{Equation 4.2-1}$$

where $\dot{\epsilon}$ = Strain rate (hr^{-1})
 ϕ = Flux (n/sec-cm^2) $E > 1 \text{ MeV}$
 $10^{13} < \phi < 3.5 \times 10^{13} \text{ n/sec-cm}^2$
 T = Temperature ($^{\circ}\text{C}$)
 σ = Hoop stress (60-200 MPa)
 κ = Constant
 $= 5.8 \times 10^{-25}$ for cold worked Zircaloy-2.
 $= 2.1 \times 10^{-25}$ for cold worked Zr-2.5% Nb.
 $= 3.2 \times 10^{-25}$ for heat treated Zr-2.5% Nb.

This empirical relation reflects the fact that Zircaloy creeps faster than the Zr-2.5% Nb alloys and that Zr-2.5% Nb creeps less in the cold worked condition than in the heat treated condition. Creep predictions for all three alloys are shown in Figure 4.2-5 for the proposed operating conditions of $\sigma = 143 \text{ (MPa)}$, $\phi = 3 \times 10^{13} \text{ (n/sec-cm}^2\text{)}$ and $T = 344 \text{ }^{\circ}\text{C}$. It should be noted that the design temperature is outside the range of empirical information used to develop Equation 4.2-1. Zr-2.5% Nb creep data have been obtained at higher temperatures for fuel cladding. Due to texture differences, cladding is expected to creep 10 times faster than pressure tube material. A correlation based on fuel-clad data obtained at temperatures at and above the design temperature was scaled by a factor of 10 to obtain a formula (Reference 4.15) for cold worked Zr-2.5% Nb pressure tubes:

$$\dot{\epsilon} = 1.42 \times 10^4 \sigma^{5.3} \exp(-265,000/8.314T) + (0.05) \sigma^{1.5} (1 + 3.55 \times 10^{10} T^{-7} \phi^{.65}) \exp(-100,000/8.314 \cdot T) \quad \text{Equation 4.2-2}$$

where the symbols ϕ , σ and $\dot{\epsilon}$ have the same meaning as in Equation 4.2-1 but where T is in degrees Kelvin. As illustrated in Figure 4.2-6, Equation 4.2-2 predicts less creep than Equation 4.2-1 for the proposed operating conditions. The more conservative Equation 4.2-1 consequently was used in establishing the required pressure tube thickness.

More accurate predictions will require acquisition of pressure tube creep data in the range of temperatures and stresses of interest. Suitable data may be obtained from high temperature experiments in a facility such as the organically cooled Whiteshell reactor, which operates at higher temperatures than the Candu reactors. The desired stress levels can be obtained in pressurized tube test specimens by varying the wall thickness of tube specimens (Reference 4-16).

Inasmuch as pressure tube service life is limited by diametral creep, the most creep resistant alloy cold worked Zr-2.5% Nb is recommended for use in pressure tubes. An adequate model for Zr-2.5% Nb creep in the proposed temperature range (300-350°C) has not been found. Data in this range must be obtained before an accurate creep prediction can be made for the heavy water reactor. The need for such data is highlighted by the recent discovery that pressure tube creep had been underestimated for some Ontario Hydro Candu reactors. Since retubing is expensive, such underestimates can have serious economic consequences. On the other hand, using a too conservative creep prediction can lead to costly thermal inefficiency.

Embrittlement and axial elongation over long periods of time are probably of secondary importance but must be considered in the final design. Likewise, final design will have to consider pressure tube sag due to the combined effects of creep and channel weight. Acquisition of data from current Candu systems should allow long range prediction of these phenomena to be made.

4.2.5 Conclusions

The conclusions reached in the primary system pressure evaluation are as follows:

1. For a given thermal power output, the standard Candu reactor will generate about 12% less electrical energy than will a PWR because of the lower net efficiency of the Candu reactor; this lower net electrical efficiency contributes to a higher effective plant capital cost since less electrical energy can be generated from HWR plants designed to produce a given thermal output.
2. Part of the lower thermal efficiency of the standard Candu reactor is attributable to lower primary and secondary pressures. If primary and secondary pressures are increased to those of the PWR, HWR efficiency increases from the 29.3% of the standard Candu reactor to about 31.3%, a value about 5.2% lower than that of the PWR.
3. In order to increase secondary temperatures and pressures to those comparable with the PWR, primary temperatures must be increased and as a result primary pressures must be increased so as to preserve thermal margins. This increase in primary pressure necessitates increasing pressure tube thickness.
4. The increase in pressure tube thickness results in a decrease in achievable burnup. For natural uranium fueling, this decrease in achievable burnup is large, decreasing the discharge burnup to about 6300 MWd/t, and results in about a 10% increase in uranium requirements. However, for slightly enriched fueling, the decrease in achievable burnup becomes a small fraction of the total cycle length and is more than compensated by the increase in electrical efficiency; as a

result, uranium resource utilization is improved slightly when higher primary system pressure is employed with slightly enriched fuel.

5. Review of pressure tube design considerations associated with the use of higher coolant temperatures and pressure indicates a likelihood that the Zr-Nb pressure tubes can operate successfully under these conditions. However, it is recognized that increasing primary system pressure represents a significant departure from present Candu design, which will require pressure tube development and demonstration prior to licensing.
6. The increase in net electrical efficiency from 29.3 to 31.3%, a 6.8% improvement, is reflected in a similar reduction in effective plant capital cost. Improvements in plant efficiency thus represents a significant contribution toward reducing the capital cost of the HWR to a value more comparable to that of the PWR.
7. In view of the significant reduction in plant capital cost realized by increasing primary and secondary system pressures, the likelihood that pressure tubes can successfully operate under these conditions, the probable likelihood that additional pressure tube development and demonstration would at any rate be required for U.S. licensing, and the satisfactory uranium resource utilization performance when slightly enriched fuel is employed, it is concluded that higher operating temperatures and pressures should be incorporated into the conceptual HWR design.

Table 4.2-1
EFFECT OF PRIMARY AND SECONDARY CONDITIONS
ON NET ELECTRICAL EFFICIENCY

PRIMARY PRESSURE PSIA (MPa)	PRESSURE TUBE THICKNESS (mm)	COOLANT OUTLET TEMP. °C	SECONDARY PRESSURE PSIA (MPa)	NET ELECTRICAL EFFICIENCY %
1600 (11.0)	4.06	266	700 (4.7)	29.3
1800 (12.4)	4.57			30.1
2000 (13.8)	5.08			30.8
2250 (15.5)	5.79	299	1100 (7.7)	31.3

Table 4.2-2
EFFECT OF PRIMARY SYSTEM DESIGN (PRESSURE)
ON ACHIEVABLE DISCHARGE BURNUP

PRIMARY SYSTEM PRESSURE PSIA (MPa)	DISCHARGE BURNUP, Mwd/TeM		
	<u>NAT. URANIUM</u>	<u>1.2 W/O</u>	<u>1.6 W/O</u>
1600 (11.0)	7500	22774	32686
1800 (12.4)		22415	32130
2000 (13.8)		22056	31744
2250 (11.5)	6288	21697	31434

Figure 4.2-1
EFFECT OF PRIMARY SYSTEM PRESSURE ON POWER COST

Figure 4.2-2
 RESOURCE UTILIZATION AT 1600 AND 2250 PSIA
 PRIMARY SYSTEM PRESSURE

Figure 4.2-3
INFLUENCE OF DISSOLVED OXYGEN ON CORROSION OF ZIRCONIUM IN ALLOYS UNDER IRRADIATION, WHERE HIGH AND LOW FLUX REFERS TO IRRADIATION AT 2×10^{14} AND 6×10^{12} n/cm² - sec

Figure 4.2-4
RELATIONSHIP BETWEEN CRITICAL CRACK LENGTH AND STRESS FOR VARIOUS
HYDROGEN CONCENTRATIONS AND IRRADIATION EXPOSURES

$\phi = 3 \times 10^{13} \text{ n/cm}^2\text{-sec.}$ E 1 Mev.
 $T = 651^{\circ}\text{F}$ (344°C)
 $P = 15.16 \text{ MPa}$ (2.2 KSI)
 $ID = 104 \text{ mm}$ (4.09 IN.)
 $t = 5.64 \text{ mm}$ (0.222 IN.)

Figure 4.2-5
 COMPARISON OF DIAMETRAL CREEP AT 100% POWER
 FOR THREE ZIRCONIUM ALLOYS AS PREDICTED
 BY EQUATION (1)⁽³⁾

$$\phi = 3 \times 10^{13} \text{ n/cm}^2\text{-sec.}$$

$$T = 651^{\circ}\text{F}$$

$$P = 15.16 \text{ MPa}$$

$$\text{ID} = 104 \text{ mm}$$

$$t = 5.64 \text{ mm}$$

Figure 4.2-6
COMPARISON OF CREEP PREDICTIONS
FROM EQUATIONS (1) AND (2) FOR PROPOSED
REACTOR OPERATION CONDITIONS

4.3 Lattice Spacing Evaluation

4.3.1 Fuels Cycle Evaluation

One method for reducing the capital cost of heavy water reactors is to reduce lattice spacing, i.e., to reduce the distance between fuel channels. Reductions in lattice spacing will reduce the heavy water inventory present in the plant, and hence of the contribution to power cost attributable to heavy water inventory charges. Overall reactor dimensions will also be decreased, but this proves to have little impact on containment size since containment dimensions are set by the size of other components (the steam generator height and length of the reactor).

Decreasing the lattice spacing will result in a reduction in achievable burnup, and hence in resource utilization, since the removal of D_2O will result in a less well thermalized neutron spectrum. Improvements in power cost resulting from the reduced D_2O inventory charges will thus be compensated by higher fuel cycle costs resulting from the decrease in achievable discharge exposure.

The effect of reducing lattice spacing on resource utilization and on fuel cycle costs is shown in Figures 4.3-1 and 4.3-2, respectively. As Figure 4.3-1 illustrates, the effect of reducing lattice pitch from the reference value of 28.6 cm to 22.8 cm is to reduce the achievable burnup by approximately 2,500 MWd/t. This burnup penalty is relatively constant, i.e., independent of initial enrichment. As a result, the impact on relative U_3O_8 utilization decreases as the initial enrichment is raised, because the constant burnup penalty represents a decreasing fraction of the total achievable cycle length. The effect of decreasing lattice pitch, consequently, decreases as the enrichment is increased; for example, decreasing the pitch to 22.8 cm results

in a 7% reduction in relative U_3O_8 utilization for an initial enrichment of 2.0 w/o, but a 26% reduction for an initial enrichment of 1.0 w/o. Thus, although a rather significant penalty would occur if the pitch were decreased in a natural uranium fuel system, the penalty associated with decreasing lattice pitch becomes smaller as the enrichment is decreased, and tends to result in an optimal enrichment somewhat greater than that for the standard 28.6 cm pitch. Calculations were also run at a 17.8 cm pitch. The burnups achievable with this very tight lattice, however, were very low; a burnup of 9,160 MWd/t was achieved with an initial enrichment of 1.6 w/o, and the reactor was subcritical with an enrichment of 1.2 w/o. Consequently, such tight lattices are impractical.

The effect of decreasing lattice pitch on fuel cycle economics is shown in Figure 4.3-2. Here, the change in fuel cycle cost is plotted as a function of lattice pitch for several different values of initial enrichment and several values for the cost of U_3O_8 . The cost of the heavy water inventory has been included in these calculations. As illustrated by this figure, fuel cycle costs are quite sensitive to the initial enrichment for tight (22.8 cm) pitches, but this sensitivity is greatly reduced for the more moderated lattices. Thus, one can conclude that the more moderated lattices will allow the flexibility to employ a variety of initial enrichments (from natural uranium on up) while the use of tighter lattices would limit the choice of initial enrichment to 1.2 w/o or greater. This figure also illustrates that the optimal lattice pitch from the standpoint of fuel cycle cost economics occurs at a slightly lower pitch than the reference Candu design, and further that this optimum pitch is rather insensitive to the price of uranium, although the optimum does tend more toward the reference Candu design as the price of uranium increases. However, as Figure 4.3-2 illustrates, the differential fuel cycle cost between the optimal pressure tube

pitch and the reference Candu design is small, particularly for higher-priced uranium.

4.3.2 Mechanical Design Considerations

Reduction in lattice spacing would have a significant impact on HWR design. The two major areas of concern are refueling machine access and end shield structural integrity. On-line refueling of the HWR requires that pressure tubes be spaced sufficiently far apart to allow access of the refueling machines. Decreases in lattice spacing could potentially necessitate major redesign of the refueling machines to accommodate the smaller spacing between refueling channels.

Structural integrity for the pressure tube assembly is provided by the end shield. Decreasing lattice spacing will reduce the web-strength of the end shield, necessitating a careful reevaluation of the structural integrity of this component during seismic events. Care must also be taken to ensure that the spacing between fueling channels is sufficiently wide to allow adequate access by control rods. Thus, it appears that modifications which reduce the spacing between fuel channels has significant impact on HWR design, and consequently should be considered only if there proves to be significant economic or resource utilization incentives.

4.3.3 Conclusions

1. Reductions in lattice pitch result in a significant decrease in achievable cycle length. This reduction in achievable cycle length results in a large degradation of resource utilization, for initial enrichments less than 1 w/o, but becomes less significant when higher enrichments are employed.

2. When both the decrease in uranium utilization and the reduction in D_2O inventory are considered, only small overall reductions in fuel cycle cost are achieved by reducing lattice spacing to an optimal value.
3. Reducing lattice spacing requires careful reevaluation of Candu end-shield design to ensure that the resulting weakened end-shield structure remains adequate under seismic events. Some redesign of refueling machines, so as to allow them to fit into the smaller spacing between fuel channels, will probably also be required.
4. There appears to be little incentive for reducing lattice pitch, in view of the rather major design changes and the small fuel cycle cost incentives.

Figure 4.3-1
RELATIVE URANIUM UTILIZATION AS A FUNCTION OF INITIAL ENRICHMENT
FOR VARIOUS FUEL CHANNEL PITCHES

Figure 4.3-2
CHANGE IN FUEL CYCLE COST vs FUEL CHANNEL PITCH

4.4 Reflector Thickness Evaluation

4.4.1 Economic and Fuel Cycle Evaluation

The standard Candu reactor employs a reflector of D_2O having a thickness of approximately 66 cm. This D_2O reflector surrounds the fuel channels and is located between the fuel channels and the calandria tank wall. The calandria tank is, in turn, surrounded by a second reflector of light water, the primary purpose of which is to provide biological shielding. Reductions in D_2O inventory costs can be realized by reducing the thickness of the D_2O reflector; the thickness of the surrounding light water reflector would be increased so as to provide an equivalent degree of biological shielding.

As the thickness of the D_2O reflector is decreased, some reduction in achievable cycle length will result due to an increase in core leakage. While this decrease in achievable burnup may prove significant for natural uranium-fueled systems where the achievable burnup is already low, such reductions in achievable burnup may be acceptable for systems employing slightly-enriched uranium fuel. Also, the neutron leakage from the conceptual HWR design is smaller than that from the standard Candu because of the larger generating capacity and hence physical size of the conceptual design, and so small changes in the amount of neutron leakage will have less of an effect on the achievable burnup in the conceptual design than would occur in the standard Candu.

In order to evaluate the effect of reflector thickness, a series of two-dimensional calculations were performed with reflector thicknesses of 85.8, 57.7 and 28.6 cm. These calculations indicate that the effect of decreasing reflector thickness by a factor of two is only a small (100 MWd/t) reduction in achievable burnup. However, reducing the reflector thickness also results in a shift of power into the center of the reactor, increasing the global peaking factor. This is illustrated in Figure 4.4-1, where

channel average power densities are presented for three reflector thicknesses. In order to rebalance the power distribution, the fuel loading schedules would have to be modified to force the power distribution out toward the periphery, and the higher leakages anticipated when the power density in the peripheral assemblies is raised will likely result in some further reduction in achievable burnup. It should be noted that some rebalancing of the power distribution will be required even when the standard reflector thickness is employed, so that reflector thickness considerations don't really add to the complexity of the absorber rod management.

The effect of reducing reflector thickness on total power cost is illustrated in Figure 4.4-2 where the change in power cost is plotted as a function of reflector thickness. This change in power cost results almost exclusively from the change in heavy water inventory, since the effect on achievable burnup is small. Although the reduction in power cost achievable by reducing reflector thickness is modest, there appears to be little reason not to incorporate such reductions into the conceptual design. In practice, the reflector thickness cannot be reduced below about 31 cm because of mechanical design considerations.

REFLECTOR THICKNESS (cm)

<div>28.6 57.7 85.8</div>								2.2 3.1 3.8	2.5 3.4 4.1	2.6 3.6 4.2	2.8 3.8 4.4
				2.0 2.7 3.5	2.4 3.1 3.8	2.9 3.7 4.4	3.4 4.1 4.7	3.9 4.5 5.1	4.1 4.8 5.2	4.4 5.1 5.6	4.4 5.1 5.5
1.1 1.9 2.5	1.8 2.4 3.1	2.6 3.2 3.8	3.2 3.7 4.3	3.9 4.5 5.0	4.4 4.9 5.4	5.1 5.6 6.0	5.4 5.9 6.2	6.0 6.4 6.7	6.1 6.5 6.7	6.4 6.8 7.1	
	2.7 3.3 3.9	3.5 4.1 4.6	4.5 5.0 5.4	5.2 5.6 5.9	6.0 6.4 6.8	6.5 6.8 7.0	7.2 7.5 7.6	7.5 7.7 7.8	7.9 8.1 8.2	7.9 8.0 8.1	
		4.7 5.1 5.5	5.6 5.9 6.2	6.6 6.9 7.1	7.3 7.4 7.6	8.2 8.3 8.4	8.6 8.6 8.6	9.3 9.3 9.2	9.4 9.3 9.2	9.8 9.8 9.6	
			6.8 7.1 7.3	7.7 7.8 7.9	8.8 8.8 8.8	9.4 9.3 9.2	10.3 10.1 10.0	10.6 10.3 10.1	11.2 10.9 10.6	11.0 10.7 10.4	
				9.0 9.0 9.0	9.8 9.6 9.5	11.0 10.7 10.4	11.4 11.0 10.7	12.3 11.8 11.4	12.3 11.8 11.4	12.9 12.4 11.9	
					11.2 10.8 10.6	11.8 11.4 11.0	12.9 12.3 11.9	13.2 12.5 12.0	13.9 13.2 12.6	13.7 13.0 12.4	
						11.3 12.5 12.1	13.6 12.9 12.3	14.6 13.8 13.2	14.7 13.8 13.1	15.4 14.4 13.7	
							14.9 14.0 13.3	15.1 14.2 13.5	16.0 15.0 14.2	15.8 14.7 13.9	
								16.3 15.2 14.3	16.3 15.2 14.3	17.1 15.9 15.0	
									17.3 16.0 15.1	17.1 15.8 14.9	
										18.1 16.8 15.8	

Figure 4.4-1

LINEAR HEAT RATE (KW/FT) IN CONCEPTUAL HWR FOR VARIOUS D₂O REFLECTOR THICKNESS

Figure 4.4-2
EFFECT OF D₂O REFLECTOR THICKNESS
ON FUEL CYCLE COSTS

4.5 Primary System Arrangement Evaluation

4.5.1 Introduction

As stated previously, one of the objectives of the optimization study was to evaluate possible modifications which could be incorporated into the HWR conceptual design which would reduce the capital cost. The standardized Candu-600 employs two coolant loops, with each coolant loop containing two steam generators and two primary coolant pumps. The 1260 MW(e) electrical capacity of the conceptual design could be obtained by either increasing the size of each coolant loop or by the addition of a third coolant loop. If a third coolant loop were added, net generating capacity could be increased to the 1000-1200 MW(e) range using components of a size which have been, or which shortly will be, demonstrated. Thus, this is the simplest approach for increasing station capacity since it makes the greatest use of developed components, and may consequently be the preferred approach for the initial HWR units introduced to the U.S. This three-loop approach was employed in previous conceptual design evaluations (Reference 4.5-1), and has also been considered in Canadian large plant conceptual design studies. However, on the bases of an nth-of-a-kind plant study, it is prudent to consider the potential for a capital cost reduction due to the economy of size related to fewer but larger primary system components. This evaluation is also warranted by the evolution on the CANDU plants, where as indicated by Table 4.5-1 the direction has been towards larger components, and since the large component technology has been well developed in the U.S. where large primary system components are utilized on commercial PWRs.

4.5.2 Evaluation

In evaluating the primary system arrangement for a HWR a number of areas could be affected and thus impact on the capital cost of the plant. Areas which could be affected are as follows:

- ° Containment size
- ° Component cost
- ° Heavy water inventory
- ° Primary system piping
- ° Supports for piping and components
- ° Construction time and costs

Of the areas listed above only the first three (containment size, component cost, heavy water inventory) were evaluated as part of the optimization study. The other items listed, although important, generally favor the selection of fewer and larger components, whereas, it was not clear to what extent first three items would follow this trend for a HWR.

The containment design developed for the three loop 1100 MW(e) HWR reported in Reference 1 was used as a point of reference for determining the impact on containment of a two loop system. The containment diameter of 168 ft for this three loop plant was established by the size and arrangement of the primary system components and is larger than a PWR of comparable electrical generating capacity. After developing the primary system component sizes and developing an arrangement for a two loop system the resultant containment diameter was no longer established by the primary system but by the length of the calandria and on-line refueling equipment space requirement. As a result a reduction of 22 ft to 146 ft was realized in containment diameter, while the height of the containment shell increased to account for the increase height of the steam generators. A cost comparison

between the two containments indicate a cost reduction of approximately \$3 million dollars in direct capital cost with the two loop configuration.

In evaluating component costs, those sized for a 3-loop configuration (i.e. six steam generators and six primary coolant pumps) were compared to those for a 2-loop configuration (i.e. four steam generator and four primary coolant pumps. In each case the cost advantage favored the fewer and larger units. For both the steam generators and pumps the cost advantage was approximately 13% which would amount to approximately \$3 million.

A comparison of heavy water inventories in the primary system between 2 loops and 3 loops yielded approximately equal inventories. For this evaluation neither configuration was optimized for header or piping volumes, however, it would not be expected that the difference in inventories would change significantly to alter the conclusions reached herein.

The primary disadvantage of employing the two-loop configuration is that new and larger steam generators and pumps must be developed. The four steam generators employed in the conceptual HWR design each have evaporation rates of about 2,000 Mg/Hr. Although these steam generators are not large compared to some PWR units (some PWR units employ steam generators with evaporation rates twice as large), these steam generators are quite a bit larger than those currently in operation in Candu units. The largest Candu steam generators currently in operation are those of the Bruce Station, which have an evaporation rate of 525 Mg/Hr; the standardized Candu-600 plant will have somewhat larger steam generators, each having an evaporation rate of about 930 Mg/Hr. The development of the larger steam generators of the two-loop configuration is not thought to pose fundamental technical problems, since larger steam generators have been designed for light water reactors. The primary coolant pumps employed in the two-loop configuration

also are larger than similar pumps employed to date in Candu or light water reactors. The conceptual design employs four pumps, each having an operating load of 9275 kW; the largest pumps employed to date in Candu units are the 8200 kw Bruce units, while the pump for a large PWR is 6600 kW. Consequently, some development will be necessary to design the slightly-larger pumps required, but this development is thought to be straightforward, requiring some RD&D, but offering no fundamental technical problems.

4.5.3 Conclusions

Although the potential savings (approximately \$6 million) in capital cost indicated by this evaluation are relatively small in comparison to the total capital cost of the plant, it is more significant when considering the difference in capital cost between an HWR and the U.S. light water systems. In addition some of the items not evaluated will also provide cost reductions which are reflected in capital cost developed in Section 5.3. As mentioned before, the primary incentive for employing a three-loop configuration would be the use of standardized components. However, some change in component design would be required at any rate because of the increased temperatures and pressures employed in the primary and secondary systems of the conceptual HWR design. The lower capital cost of the two-loop configuration coupled with the need to redesign loop components to accommodate the higher operating temperatures and pressures led to the adoption of the two-loop configuration in the conceptual HWR design.

TABLE 4.5-1
EVOLUTION OF HEAT TRANSPORT SYSTEM

	<u>Pickering A</u>	<u>Bruce A</u>	<u>Candu-600</u>
Net Power, MWe	512	732	638
Main pumps per reactor - No. x kW	16 x 1400	4 x 8200	4 x 5300
Steam generators per reactor - No x Mg/Hr (evaporation rate)	12 x 245	8 x 525	4 x 930

References

- 4.5-1 C00-2477-13, "Capital Cost Pressurized Heavy Water Reactor Plant",
June, 1977.

4.6 Moderator Temperature Evaluation

4.6.1 Introduction

Approximately 6% of the total energy produced in the HWR is deposited in the moderator, primarily from the interaction of the moderator with neutron and gamma radiation. A moderator heat exchanger must be provided to remove the energy deposited in the moderator. The standard Candu 600 reactor employs once-through cooling and maintains a moderator temperature of about 71°C. The maintenance of this moderator temperature is facilitated by the use of once-through cooling, which provides ample quantities of low temperature water to the moderator heat exchanger. In the conceptual HWR design, mechanical draft cooling towers are used in place of once-through cooling and as a result, the temperature of the water available to the moderator heat exchanger is higher. Because of the low temperature differential between the operating temperature of the moderator and the temperature of the cooling water, the moderator heat exchange system would be rather large in order to accommodate closed cycle cooling.

The size and cost of the moderator cooling system could be reduced if the moderator temperature were increased so as to provide a larger differential temperature between the moderator and the cooling medium heat sink. Increasing moderator temperature, however, can potentially adversely impact the achievable discharge burnup. The objective of the evaluation discussed below was consequently to determine the effect on discharge burnup of an increase in moderator temperature so as to establish the feasibility of increasing moderator temperature.

4.6.2 Fuel Cycle and Economic Evaluation

In order to evaluate the impact of increasing moderator temperature on achievable discharge exposure, fuel cycle calculations were repeated for a moderator temperature of 93°C (200°F), a 22°C increase above that currently employed in Candu reactors. These calculations were performed both for natural uranium fueling and for fuel initially enriched to 1.2 w/o. This evaluation indicated that increasing moderator temperature has only a small effect on the achievable discharge exposure, decreasing the natural uranium fueled cycle length by about 1% and the cycle length for 1.2 w/o enriched fuel by 0.6%.

4.6.3 Conclusions

Increasing moderator temperature to 93°C results in only an insignificant decrease in achievable burnup, and hence fuel cycle cost. Thus, there appears to be no reason not to increase moderator temperature so as to allow the cost of the moderator heat exchange system to be reduced.

4.7 Enrichment Zoning Evaluation

4.7.1 Introduction

As discussed in Section 4.1, the use of slightly enriched fueling increases core peaking factors. The change in core peaking factors results from both increases in local peaking (i.e., the peaking within a fuel cluster) and from increases in the channel-to-channel factor. Although there are a number of methods for mitigating this high peaking, such as through fuel management or by decreasing the average power density in the fuel channels, one potentially attractive option is to employ local enrichment zoning within the HWR fuel assemblies.

The incentive for enrichment zoning is illustrated in Figure 4.7-1 which gives the relative power density of each fuel ring in an HWR assembly for natural and 1.2 w/o fueling. This figure illustrates that the outer fuel ring tends to run at a higher power density than do fuel rods interior to the fuel assembly, and that this effect is more pronounced when slightly-enriched fueling is employed. The power density between the outer fuel ring and the interior fuel can be balanced by local enrichment zoning, where a slightly lower enrichment is used in the outer ring and a slightly higher enrichment used for interior fuel rods. This balancing of power distribution is illustrated in Figure 4.7-2 where the power density in the outer fuel pins and in the next row in (3rd fuel ring) is plotted as a function of outer fuel ring enrichment. In these calculations, all of the fuel rods in the interior fuel rings were of a single enrichment established by requiring the average enrichment of the fuel assembly to remain constant at 1.2 w/o. As this figure illustrates, as the enrichment of the outer fuel ring is decreased from 1.2 w/o (and the enrichment of the interior fuel increased so as to maintain the 1.2 w/o average) the power density of the outer ring decreases while that of the

third fuel ring increases. The optimal enrichment occurs when the power density in these two rings is equal, i.e., at an outer ring enrichment of about 1.07 w/o.

The effect of enrichment zoning on relative power density during depletion is shown in Figure 4.7-3. Here the evolution of power peaking factor for the natural uranium and uniform 1.2 w/o case is compared with that of the zoned assemblies (2 zoned cases are shown, one with a 1.074 w/o outer ring enrichment and one for a slightly lower outer ring enrichment, 1.064 w/o). Of interest here is the increase in power peak which occurs early in the burnup cycle; the increase in power peaking late in burnup isn't significant because the average channel power density will be low for fuel having this high an exposure. Figure 4.7-3 illustrates that power peaking can be reduced from about 12% for the uniform 1.2 w/o case to about 4% with the application of enrichment zoning, i.e., an enrichment zoning can provide about an 8% improvement in power peaking. In fact, the use of local enrichment zoning reduces power peaking by about 4% with respect to natural uranium fueling. Thus, the use of local enrichment zoning will provide a margin of about 4% which can be used to compensate for the increase in channel-to-channel peaking noted in Section 4.1.

The local enrichment zoning scheme discussed above employed two different enrichments in the fuel cluster: one enrichment for the outer fuel ring and a second enrichment for all interior fuel. Power peaking can be reduced still further by using a greater number of different enrichments, one for each fuel ring. The effect of using four different enrichments in the HWR cluster is shown in Table 4.7-1. This type of enrichment zoning would reduce power peaking by about an additional 2%. However, it is expected that fabrication costs would increase because of the large number of different fuel enrichments, and the 2% gain in peaking probably does not warrant this increased complexity.

The effect of enrichment zoning on achievable cycle length was also evaluated. It was found that enrichment zoning had a negligible effect on achievable burnup, reducing the achievable burnup by less than 0.1%.

4.7.2 Conclusions

1. Local enrichment zoning can decrease the peak power density observed in uniformly enriched 1.2 w/o fuel assemblies by about 8%. The resulting peak power densities are about 4% lower than that observed for the natural uranium fueled assembly.
2. Local enrichment zoning has a negligible effect on achievable cycle length.
3. It is concluded that local enrichment zoning is an attractive method for reducing the power peaking observed for slightly enriched fuel and should be incorporated into the reference conceptual HWR design.

TABLE 4.7-1
 ENRICHMENT ZONING FOR
 1.2 w/o AVERAGE H.W. CLUSTER
 AT 50 MWD/MT

<u>Ring No.</u>	<u>No. Pins</u>	Uniform Enrichment		Zoned Enrichment	
		<u>w/o</u>	<u>Fract. Pin Power</u>	<u>w/o</u>	<u>Fract. Pin Power</u>
1	1	1.200	2.17	1.589	2.71
2	6	1.200	2.25	1.496	2.71
3	12	1.200	2.52	1.281	2.71
4	18	1.200	3.01	1.026	2.69

Figure 4.7-1
 RELATIVE POWER DENSITY FOR NATURAL URANIUM
 AND 1.2 W/O ENRICHED FUEL ASSEMBLIES (BOC)

Figure 4.7-2
PIN POWER vs ENRICHMENT OF OUTER FUEL RING
(AVERAGE U-235 W/O = 1.2, 50 MwD/TeM)

Figure 4.7-3
RELATIVE PIN POWER PEAK
AS A FUNCTION OF DEPLETION

4.8 Fuel Design Assessment

4.8.1 Introduction

A fuel rod design has been developed for a once-through pressurized heavy water reactor, based on a scale-up of the Candu design and utilizing slightly enriched uranium (~ 1.2 w/o U-235) rather than naturally enriched uranium (0.71 w/o U-235).

The fuel rod design is based on fuel performance calculations made for the lead rod considering burnup enhancement of fission gas release (a C-E developmental model was used), restrained fuel swelling rates from BOL, a uniform axial power shape, and an upper limit of $2000 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ on the gap conductance. The plenum gas temperature was conservatively assumed to be at the fuel average temperature due to the short plenum length. The fuel surface roughness specified for this application is about one-half the value typical of PWR fuel pellets, and was chosen based on preliminary calculations which showed large effects of surface roughness on the fuel temperature and EOL internal rod pressure.

4.8.2 Fuel Design Criteria

A fuel rod design has been established for the once-through HWR and satisfies the following design criteria:

- A. The fuel rod design and materials shall be such that the specified peak rod average burnup can be achieved without exceeding a tensile cladding hoop strain of 1% caused by the effects of fuel swelling and differential thermal expansion at operating temperatures.

- B. The internal gas pressure in the lead rod shall not exceed the system coolant pressure during steady-state operation.
- C. The lead rod shall be capable of operating up to a peak linear heat generation rate (LHGR) of 16.5 kw/ft to a burnup of 22,000 Mwd/Mtu without failure.
- D. The fuel rod bundle design and materials shall be such as to withstand a reactor residence time of at least 2 years without failure.
- E. Pressurized heavy water shall be used as the heat exchange medium and moderator; hence, materials shall be compatible for operation in this medium over the range of operating temperatures.
- F. The fuel rods (assembled in a 37-rod cluster) shall be run horizontally and the refueling method will be on-power (twelve bundles end-to-end in each channel).
- G. Where feasible, fuel rod dimensions shall be based on typical dimensions of a 600 Mw Candu design.

4.8.3 Fuel Rod Geometry and Materials Requirements

A nominal fuel rod design is specified in Tables 4.8-1 through 4.8-3. Where feasible, fuel rod dimensions are based on typical dimensions of a 600 Mw Candu design.

Table 4.8-1 lists the geometry and material requirements for the slightly enriched UO_2 pellet fuel. The pellets are dished, chamfered and of a stable structure which exhibits little in-reactor densification. Sufficient void volume is included in the design to accommodate fuel swelling in the lead rod.

Table 4.8-2 lists the geometry and materials requirement for the fuel rod cladding. Zircaloy-2 might be considered for its higher corrosion resistance in case of channel boiling but the zircaloy-4 cladding currently employed in CANDU plants was retained for the conceptual design. Consideration should be given to graphite lubrication of the cladding ID due to the improved performance reported (Reference 4.17) for the use of CANLUB fuel rods in test reactor experiments. Table 4.8-3 gives details of the fuel rod design. The initial diametral gap of 4 mils is small enough to be closed by differential thermal expansion and cladding creep during the first ramp to full power.

4.8.4 Projected Operating History

Tables 4.8-4 and 4.8-5 provide details of bundle average and lead rod LHGRs and exposures. The lead rod LHGR vs. burnup history given in Table 4.8-5 is based on physics calculations and has been applied in the performance calculations described in Sections 4.8.5 through 4.8.7.

4.8.5 Fuel Thermal Performance

Central fuel temperatures, gap conductances, and cladding temperatures have been calculated for the lead rod using initial helium fill pressures of 15 and 150 psia. The results are given in Tables 4.8-6 and 4.8-7. Due to initial gap closure and resulting high gap conductances, the central fuel temperatures differ by less than 100°F throughout life for the two fill pressures. There appears to be no thermal performance incentive to prepressurize to more than 1-3 atmospheres based on these results.

A peak temperature of 3580°F is calculated at about 4000 Mwd/Mtu when fuel in pile densification is completed. This peak temperature is consistent with peak temperatures reported (Reference 4.18) for the Candu Pickering Units (3630°F at 16.1 kw/ft). Peak fuel temperatures reported (Reference 4.18) for typical domestic commercial BWRs and PWRs are significantly higher (4170-5070°F). These factors suggest good fuel thermal performance would be expected in the proposed design.

4.8.6 Fuel Rod Internal Pressure

Table 4.8-8 presents the results of calculations of fuel rod internal pressure in the lead rod for initial helium fill pressures of 15 and 150 psia. Prepressurization to 150 psia results in an internal pressure which approaches the system pressure of 2200 psia during steady-state operation. Exceeding the system pressure could cause the cladding to lift off the fuel and result in decreased gap conductances and increased fuel temperatures. With 15 psia initial fill pressure, the internal pressure does not exceed 1100 psia during steady-state operation throughout life. The use of only 1-3 atmospheres initial fill pressure, therefore, is recommended since it provides sufficient margin to a system-pressure limitation on the internal pressure.

4.8.7 Cladding Mechanical Performance

A maximum cladding strain calculation was made considering the effects of differential thermal expansion and restrained fuel swelling. At BOL, the differential thermal expansion in the lead rod causes a cladding strain of 0.45% if the effects of fuel creep and hot pressing are ignored (incompressible fuel). Using a restrained swelling rate of 0.2% $\Delta V/V$ per 4000 Mwd/Mtu (after 4000 Mwd/Mtu), the cladding total strain calculation becomes 0.75% at the peak rod burnup of 22,000 Mwd/Mtu. If a volume

change of 0.5% $\Delta V/V$ due to densification is considered, the total cladding strain is reduced to about 0.6%. The results of this conservative calculation indicate that the peak cladding hoop strain limit of 1% would not be violated.

The effects of fuel creep and hot pressing are considered in the calculation results presented in Table 4.8-9 for cladding hoop stress and strain during the first ramp to full power. Results for initial helium fill pressure of 15 and 150 psia are included. The results demonstrate that significantly lower fuel stresses and strains are expected in the real case when compared to the results obtained for the incompressibility assumption. For example, peak tensile hoop strains of only 0.13 and 0.16% are calculated for 15 and 150 psia initial fill pressures, respectively. These values are significantly less than the 0.45% BOL strain presented previously using the incompressibility assumption. As shown in Table 4.8-9 the effects of external pressure and cladding mechanical restraint cause considerable stress and strain relief through the fuel flow mechanisms as full power is achieved at the top of the ramp.

4.8.8 R&D Program Recommendation

An irradiation test program is recommended to qualify the proposed fuel design for the once-through HWR concept. Two separate irradiation experiments are suggested based on the availability of test facilities having high coolant pressure capability (2000 psia or greater), such as the vertical test loops of the NRX reactor. A lead irradiation test would involve a string of up to six rods which, for example, could be irradiated in the X-4 loop of the NRX reactor. This loop is rated at 2200 psia and has an active test section length of about 118 inches. A burnup in excess of 20,000 Mwd/Mtu could be achieved in the test rods in less than two years. The second test would be of larger scale,

and its irradiations would lag the lead test by about one year. This test would feature either two 13-rod bundles or three or four 7-rod bundles of standard length. The X-5 loop of the NRX reactor, for example, would be a suitable test bed since its test section diameter (3.25 inches) is sufficiently larger to accommodate either test cluster design alternative. The loops' coolant pressure capability is about 2500 psia. A target burnup of 22,000 could be achieved in this loop also in less than two years. The two test programs cited should provide an adequate demonstration of the performance of single rods and of a cluster configuration to qualify the proposed fuel design for full-scale power reactor applicable.

A preliminary planning stage for the irradiation test programs would require about three months to complete and cost about \$50,000. The estimated cost of the irradiation program, based on the scope outlined previously, is estimated at about \$5 million and could be completed within about five years following the initiation of the planning stage, assuming availability of suitable irradiation test facilities such as the referenced test loops of the NRX reactor. The cost of the program includes test fuel and cluster design, performance analysis, materials qualification tests, licensing, irradiation, and a post-irradiation examination campaign.

4.8.9 Conclusions

The results of the evaluation performed indicate that helium prepressurization above the recommended 1-3 atmosphere is not necessary. Gap conductances were found to be good ($>1500 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$) and peak fuel central temperature ($<3580^\circ\text{F}$) not significantly different in the lead rod for either 15 psia or 150 psia initial fill pressures, since fuel-clad contract exists from BOL in either case. Prepressurization to 150 psia, however, results in

internal pressures which approach the system pressure of 2200 psia during steady-state operation. Exceeding the system pressure could cause the clad to lift off the fuel and result in decreased gap conductances and increased fuel temperatures. With a 15 psia initial fill pressure the internal pressure does not exceed 1100 psia during steady-state operation throughout life. The use of only 1-3 atmospheres initial fill pressure, therefore, is recommended since it provides sufficient margin to a system pressure limitation on the internal pressure.

A demonstration program is recommended that would span about 5 years and cost about \$5 million. The first phase of this program could begin in fiscal year 1979 and would involve a 3-month test planning stage which has been costed at about \$50 thousand.

TABLE 4.8-1

FUEL PELLET NOMINAL DESIGN

Pellet OD	(in)	0.478
Pellet Length	(in)	0.620
Stack Length/No. Pellets Per Rod	(in), (no.)	18.6/30
No. of Dishes Per Pellet		2
Chamfer		Both Ends
Dish Diameter	(in)	0.400
Dish Depth (Spherical)		0.021
Inner Chamfer Diameter	(in)	0.464
Height of Chamfer	(in)	0.007
Land Width	(in)	0.032
Fuel Column Volume (Including Dishes and Chamfer)	(in ³)	3.34
Total Dish Volume/Pellet	(in ³)	0.00265
Total Dish Volume/Stack Height	(in ³ , v/o)	0.0794, 2.38
Total Chamfer Volume/Stack Height	(in ³ , v/o)	0.0022, 0.066
Pellet Density	(% TD, g/cm ³)	96.5, 10.58
Resinter Densification Limit	(% TD)	0.5
O/U Ratio		2.00
Material		UO ₂ , Sintered
Enrichment	(U ²³⁵ - w/o)	1.2
Fuel Stack Height Density	(% TD, g/cm ³)	94.0, 10.3
Total Dish Volume Plus Chamfer Volume	(in ³ , v/o)	0.0816, 2.45

TABLE 4.8-2

CLADDING NOMINAL DESIGN

Cladding Material	Zr-2 or Zr-4 (Cold-Worked and ⁽⁺⁾ Stress-Relief Annealed)
Cladding OD (in)	.515
Cladding ID (in)	0.482
Cladding Length (in)	18.8
Cladding Thickness (in)	.017

(+) Consideration should be given to graphite lubrication of the cladding ID.

TABLE 4.8-3

FUEL ROD NOMINAL DESIGN

Helium Prepressurization (atmospheres)	1-3(*)
Fuel-Clad Diametral Clearance (gap) (in)	0.004
End Cap Material	Zr-2
End Cap Configuration	Flat - No Internal Void
Plenum Length (in)	0.2
Spring	None
Spacers	None

*High prepressurization is not recommended due to resulting high fuel rod internal pressures (See Section 4.8.6).

TABLE 4.8-4

OPERATING HISTORY DATA

Lead Bundle Average LHGR (kw/ft)	14.4
Lead Bundle Average Burnup (Mwd/Mtu)	19,500
Lead Rod LHGR (kw/ft)	16.5*
Lead Rod Axial Average Burnup (Mwd/Mtu)	22,000
Lead Rod Peak Local Burnup (Mwd/Mtu)	25,000
Exposure Time (Years)	~ 2
Total Fast Fluence in Cladding of Lead Rod (n/cm ² , E > 1 Mev)	4.1 x 10 ²¹
D ₂ O Coolant Pressure (Psia)	2200

*An outer-row fuel rod

TABLE 4.8-5

ESTIMATED LHGR HISTORY OF THE LEAD ROD

<u>Axial Average LHGR (kw/ft)</u>	<u>Axial Average Burnup Increment (Mwd/Mtu)</u>	<u>Time Increment (Hrs.)</u>	<u>Comment</u>
16.3	0	0	BOL Ramp Rate is 0.3 kw/ft-hr.
16.5	2,000	950	
16.0	"	940	
15.6	"	990	Steady-State Operation (Drop in Power Capability is Based on Physics Calculations.)
14.8	"	1040	
14.0	"	1100	
13.1	"	1180	
12.4	"	1250	
10.9	"	1420	
10.6	"	1460	
10.3	"	1500	
10.3	"	1500	
<u>13.4 kw/ft</u> Burnup Averaged	<u>22,000</u> Mwd/Mtu	<u>13,330</u> Hours	

TABLE 4.8-6

FUEL AND CLADDING TEMPERATURES IN THE LEAD ROD

Axial Average Burnup (Mwd/Mtu)	Axial Average LHGR (kw/ft)	Fuel Centerline Temperatures (°F)		Cladding Temperature (°F)	
		150 psia He Fill Pressure	15 psia He Fill Pressure	OD	ID
0	16.3	3557	3558	653	719
2000	16.5	3537	3537	"	719
4000	16.0	3578	3581	"	720
6000	15.6	3375	3427	"	716
8000	14.8	3194	3268	"	713
10,000	14.0	3013	3100	"	710
12,000	13.1	2811	2902	"	706
14,000	12.4	2656	2746	"	703
16,000	10.9	2335	2414	"	697
18,000	10.6	2273	2350	"	696
20,000	10.3	2211	2286	"	695
22,000	10.3	2211	2287	"	695

TABLE 4.8-7

GAP CONDUCTANCE IN THE LEAD ROD

<u>Axial Average Burnup (Mwd/Mtu)</u>	<u>Gap Conductance (BTU/hr-ft²-°F)</u>	
	<u>150 psia He Fill Pressure</u>	<u>150 psia He Fill Pressure</u>
0	2000(*)	2000(*)
2000	"	2000(*)
4000	"	1990
6000	"	1764
8000	"	1665
10,000	"	1600
12,000	"	1560
14,000	"	1545
16,000	"	1530
18,000	"	1525
20,000	"	1523
22,000	"	1520

*2000 Btu/hr-ft²-°F is assumed to be an upper limit on the gap conductance for steady-state operation.

TABLE 4.8-8

INTERNAL ROD PRESSURES IN THE LEAD ROD

Axial Average Burnup (Mwd/Mtu)	Axial Average LHGR (Mwd/Mtu)	Internal Fuel Rod Pressure (psia)*	
		15 psia He Fill Pressure	150 psia He Fill Pressure
0	16.3	169	1627
2000	16.5	409	1859
4000	16.0	658	2009
6000	15.6	843	2097
8000	14.8	971	2148
10,000	14.0	1041	2172
12,000	13.1	1075	2145
14,000	12.4	1087	2135
16,000	10.9	1013	1988
18,000	10.6	1041	2031
20,000	10.3	1069	2079
22,000	10.3	1076	2187

*The helium and fission gas mixture in the plenum is assumed to be at the fuel average temperature during steady-state operation.

TABLE 4.8-9

CLADDING STRESSES AND STRAINS FOR THE BOL RAMP TO 100% POWER

Time (Hrs.)	Axial Average LHGR (kw/ft)	Cladding Hoop Strain In The Lead Rod (in/in)		Cladding Hoop Stress In The Lead Rod (psi)	
		15 psia He Fill Pressure	150 psia He Fill Pressure	15 psia He Fill Pressure	150 psia He Fill Pressure
0	0	-0.00100	-0.00063	-30,770	-26,530
7	2	-0.00116	-0.00068	-30,610	-25,700
13	4	-0.00150	-0.00083	-30,300	-24,640
18	6	-0.00184	-0.00095	-29,960	-23,220
24	8	-0.00142	-0.00112	-20,820	-21,120
30	10	0.000002	0.00015	- 7,400	- 7,716
36	12	0.00125	0.00155	- 375	2,624
42	14	0.00010	0.00128	-15,360	- 4,342
48	16	-0.00140	0.00078	-24,620	- 8,882
54	16.3	-0.00231	0.00059	-25,110	- 9,462

As a result of the design optimization study, the following design changes relative to the standard Candu design were selected for incorporation into the conceptual HWR:

- a. Slightly-enriched (1.2 w/o) uranium fuel
- b. Primary system coolant pressure of 2250 psia
- c. A D₂O reflector thickness of 33 cm
- d. A moderator temperature of 93°C
- e. A two-loop primary system configuration incorporating four steam generators and four primary coolant pumps.

Of the various design modifications evaluated in the optimization of HWR design, only the option of reducing lattice spacing proved unattractive. As discussed in greater detail in Section 4.3, this option was not incorporated into the reference design since the complexity of design changes and the decrease in uranium utilization efficiency were felt to outweigh the small improvements in power costs which could be obtained by optimizing the spacing between fuel channels.

The reasons for incorporating the design changes identified above into the conceptual HWR design are reviewed briefly below. A more detailed discussion of each item is provided in Sections 4.1 through 4.6.

4.9.2 Primary System Coolant Pressure of 2250 psi

The primary system coolant pressure of the conceptual HWR design has been increased to 2250 psi so as to allow primary system temperatures and secondary system temperatures and pressures to be increased without adversely affecting thermal margin. This results in an increase in electrical generating efficiency, and a concomitant reduction in effective plant capital cost. In order to obtain a high efficiency, a primary system coolant pressure of 2250 psi, identical to that of the PWR, was selected. The evaluation of pressure tube performance at the elevated temperatures and pressures of the conceptual HWR design indicated that the zirconium-niobium pressure tube material is suitable for operation under these conditions. As a result of the higher primary system pressure, pressure tube thickness must be increased. The effect of this increase in pressure tube thickness on uranium utilization proved insignificant for slightly enriched fueling, since the increase in net electrical generating efficiency proved to outweigh the reduction in achievable burnup; as a result, both uranium utilization and fuel cycle costs are improved slightly (for slightly enriched fueling) when higher primary and secondary pressures and temperatures are employed.

4.9.3 D₂O Reflector Thickness

The D₂O reflector thickness was decreased to 33 cm, about half that currently employed in Candu. This resulted in only small decreases in achievable burnup for slightly enriched fueling. The reflector thickness employed in a conceptual HWR design was set based upon mechanical design considerations (clearance for moderator cooling system inlet and outlet nozzles and thermal expansion of the calandria).

The moderator temperature of the conceptual HWR design was increased by 28°C to a value of 93°C, in order to reduce the size and cost of the moderator heat exchange system. This increase in moderator temperature proved to have an insignificant effect on uranium utilization or fuel cycle cost for the slightly enriched system.

A two-loop primary system configuration incorporating four steam generators and four primary coolant pumps was selected for use in the conceptual HWR design. A three-loop six steam generator configuration, which is currently contemplated for future large plant Canadian designs, was also considered. The main incentive for the three-loop configuration is one of standardization of component size with existing Candu plants. This incentive for standardization is reduced in the context of the conceptual HWR design since components would have to be modified at any rate to accommodate the higher system pressure. The two-loop configuration was selected since design evaluations indicated that plant capital costs are lower than when a three-loop configuration is employed. The reduction in plant capital cost results from economy of scale in steam generator design and because the two-loop configuration allows containment size to be reduced.

Section 4 References

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5. Pressurized Heavy Water Reactor Conceptual Design

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5. Pressurized Heavy Water Reactor Conceptual Design

5.0 Introduction

This section contains the evaluations and descriptions of the conceptual pressurized Heavy Water Reactor Plant which has been developed for U. S. siting. The design incorporates the various features determined desirable from the optimization studies reported in Section 4 and other modifications arising from consideration of current U. S. licensing requirements and U. S. construction practices for a U. S. sited PHWR.

Due to the volume of material presented, this section has been divided into subsections pertaining to various aspects of the description and evaluation. Subsection 5.1 presents the description of the plant layout, structures, nuclear steam supply, and other systems which make up the total plant. A discussion of the modifications incorporated into the plant relative to the CANDU-600 design is presented in Subsection 5.2. The capital cost evaluation of the total plant, organized by accounts in accordance with expanded AEC Code of Accounts (USAEC Report NUS-531), is given in Subsection 5.3 followed by the fuel management characterization, covering the slightly enriched uranium once-through fuel cycle, plutonium recycle, and thorium fuel cycle, in Subsection 5.4. The last Subsection 5.5, presents the safety and licensing, environmental impact, proliferation risks, and safeguards evaluations.

It should be noted that in this report particular attention has been given to those systems, components, and structures which were either affected by modification or could significantly impact power costs. Less attention has been given to areas which are generally considered standard in the industry and a reasonable bases exists for establishing costs.

The conceptual Heavy Water Reactor (HWR) Plant described herein is based upon the pressurized heavy water Canadian deuterium uranium reactor (Candu-PHW) and, in particular, upon the standard CANDU 600 reactor currently under construction at the Gentilly II and Point Lepreau stations in Quebec and New Brunswick, respectively. As discussed in greater detail in Section 4.0 this conceptual design differs from the Canadian Candu 600 design primarily because of the increase in station size to 1260 Mw(e), the use of slightly enriched uranium fuel, modifications to increase station efficiency, and because of design changes made to facilitate conformance to U.S. design or licensing practices; where possible, the design and features of the Candu-PHW reactor have been retained in order to take advantage of the extensive development and proven performance of the Canadian heavy water reactor design.

The layout for this single unit conceptual HWR plant is shown on Figure 5.1-1 and the station data is given in Table 5.1-1. The arrangement of buildings provides for optimum use of space for the various functions performed in the plant and provides controlled access to vital areas. Mechanical draft evaporative cooling towers are provided to reject heat from the condenser. An ammonia refrigeration system is provided to remove heat from the moderator heat exchanger. The arrangement of the safety-related structures are shown in Figures 5.1-2 to 5.1-9.

The Reactor Containment Building (RCB) houses the Nuclear Steam Supply System (NSSS). It is a seismic Category I reinforced concrete cylindrical structure with a hemispherical dome and a flat reinforced concrete base. The reinforced concrete base is 165 feet diameter, 10-feet thick, and founded 34 feet below finished grade. The upright cylindrical portion of the containment has an inside diameter of 146 feet, measures 183 feet - 6 inches from the top of foundation mat to the springline of the dome, and

has a 4 foot 6 inch thick wall. The dome portion is a hemispherical - shaped head having an inside height of 73 feet, 0 inches, and a 3 foot 6 inch thickness of reinforced concrete. The inside height from the top of the mat to the inside of the dome is 256 feet, 6 inches. A 3/8 inch thick carbon steel liner is applied to the inner surface of the cylinder, 1/2 inch thick liner on the inside of the dome, and a 1/4 inch thick liner on the top of the foundation mat. A 100-ton polar crane, for construction and maintenance operation, is located near the spring line of the containment and is supported from the wall of the containment. The Reactor Service Building (RSB) houses all the auxiliary systems, emergency systems, fuel handling equipment and related systems as well as all the systems associated with heavy water inventory control. It is a seismic Category I building located north-northwest of the Reactor Containment Building with exterior walls, interior walls, floor slabs and roof slab of reinforced concrete. The exterior walls are a minimum of two feet thick. The spent fuel pool is 25x25 ft. and will accommodate one year accumulation of spent fuel plus one core of fuel. The pool is lined with 1/4 inch thick Type 304 austenitic stainless steel plate, which provides a watertight membrane.

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

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

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

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

housed in this building (see Figures 5.1-10 and 5.1-11). Other non-seismic Category I buildings that are part of the plant complex, include:

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

The nuclear steam supply system (NSSS) is housed in the containment building and consists of the reactor assembly including fuel, the heat transport system, steam generators, primary system pumps, pressurizer, shutdown and control systems, and instrumentation. The reactor design is based upon a heavy water moderated, pressurized heavy water cooled, slightly enriched uranium dioxide fueled, horizontal pressure tube reactor. A simplified diagram illustrating the features of this type of reactor is given in Figure 5.1-13. The reactor is installed in a steel lined concrete vault which is filled with light water for shielding. The main reactor structure is an austenitic stainless steel calandria tank with integral end shields and peripheral internal thermal shield. The calandria tank contains the low temperature (200°F) heavy water moderator, and operates at near atmospheric pressures. The horizontal coolant tubes, in which the fuel resides, are located within calandria tubes which pass through the calandria tank. These calandria tubes are made of thin-walled (1.4 mm minimum wall thickness) Zircaloy and are separated from the coolant (or pressure) tubes by a sealed annulus containing dry nitrogen. The purpose of this annular gas gap is to provide thermal insulation between the high temperature coolant tubes and the low temperature heavy water moderator. The coolant tubes are located within the calandria

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

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

The heat transport system is diagramed schematically in Figure 5.1-13. 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 5.1-13 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 operators 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, which contain a graded vertical loading of cobalt poison, are 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 mechanical 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.

Detailed descriptions of the various systems and components developed as part of this study are given in the following Sections 5.1.1 through 5.1.9. The major modifications relative to the CANDU are described in Section 5.2.

TABLE 5.1-1
STATION DATA

GENERAL PROJECT

GENERAL

Number of reactor units	1
Net power output	1260 MW(e)
Total fission (thermal) power	4029 MW(th)
Overall efficiency	31.3%

REACTOR

Type	Pressurized heavy water (PHW), horizontal pressure tube
Moderator and reflector	D ₂ O
Coolant	Pressurized D ₂ O
Fuel	LEU 1.2%
Refueling method	On-power, bi-directional in adjacent channels
Normal direction of fueling	Direction of flow
Fuel cycle	Once through

HEAT GENERATION BALANCE

Fission Power Distribution

Heat generation in:

Fuel	3753.9 MW(th)
Sheaths and bundle structure	19.0 MW(th)
Coolant	20.3 MW(th)
Pressure tubes	16.8 MW(th)
Total (fuel channels)	3810 MW(th)
Moderator	182.1 MW(th)

TABLE 5.1-1 (Cont.)

Reflector	16.9 MW(th)
Calandria tubes	6.3 MW(th)
Calandria shell	3.8 MW(th)
Tubesheets	3.7 MW(th)
End shields	1.5 MW(th)
Other calandria components including reactivity mechs. and guide tubes	3.7 MW(th)
Shields and structures outside calandria and end shields	1.3 MW(th)
Total (other reactor components)	219.3 MW(th)
Total fission heat	4029.3 MW(th)

HEAT TRANSFER BALANCE (PRIMARY)

Heat transport system

Heat from fuel channels	3810 MW(th)
Heat loss to moderator	5.6 MW(th)
Heat loss to end shields	4.4 MW(th)
Total heat loss (mod. and shields)	10 MW(th)
Net fission heat to coolant	3800 MW(th)
Heat loss to ht. trans. piping	5.5 MW(th)
Heat loss to ht. trans. aux.	11.0 MW(th)
Total heat loss (pipe and aux.)	16.5 MW(th)
Net fission heat to steam generators	3783.5 MW(th)
Pump energy appearing in coolant	37.1 MW(th)
Total heat transferred to steam generators	3820.6 MW(th)

Moderator system

Heat generated in moderator	182.1 MW(th)
Heat generated in reflector	16.9 MW(th)
Heat gen. in calandria tubes	6.3 MW(th)

TABLE 5.1-1 (Cont.)

Heat gen. in guide tubes and reactivity mechanisms	3.7 MW(th)
Heat from fuel channels	5.6 MW(th)
Heat from cal. shell and tubesheets	3.8 MW(th)
Total fission heat to moderator	218.4 MW(th)
Heat loss to moderator piping	0.6 MW(th)
Net fission heat to moderator	217.8 MW(th)
Pump energy appearing in moderator	1.3 MW(th)
Net heat to moderator	219.1 MW(th)
Shield Cooling System	
Heat from calandria shell	
Heat from end shields	
Heat from tubesheets	6.5 MW(th)
Heat from thermal shield structures outside calandria/end shields	
Heat from fuel channels	4.4 MW(th)
Net fission heat to shield system	10.9 MW(th)
Auxiliary systems	
Heat from ht. trans. aux.	11.0 MW(th)
Heat from ht. trans. piping	5.5 MW(th)
Heat from moderator piping	0.6 MW(th)
Net fission heat to aux. systems	17.1 MW(th)
SUMMATION	
Net fission heat to steam generators	3783.5 MW(th)
Net fission heat to moderator	217.8 MW(th)
Net fission heat to shield sys.	10.9 MW(th)
Net fission heat to aux. systems	17.1 MW(th)
Total fission heat	4029.3 MW(th)

TABLE 5.1-1 (Cont.)

HEAT TRANSFER BALANCE (SECONDARY)

Heat to steam generators	3820.6 MW(th)
Heat loss from steam generators	0.6 MW(th)
Heat loss from blowdown	1.3 MW(th)
Total heat loss	1.9 MW(th)
Net heat input to turbine cycle	3818.7 MW(th)
Generator output	1343.6 MW(e)
Heat removed by condenser cooling water	2475.1 MW(th)

UNIT ENERGY BALANCE

Generator output (gross)	1343.6 MW(e)
Station service power	83 MW(e)
Generator output (net)	1260.6 MW(e)
Power not returned to energy system (83 MW less 38.4 MW pump energy)	44.6
Heat to moderator	219.1 MW(th)
Heat to shield sys.	10.9 MW(th)
Heat to aux. sys.	17.1 MW(th)
Heat to blowdown	1.3 MW(th)
Heat to condenser	2475.1 MW(th)
Total heat loss	2768.1 MW(th)
Total energy (fission power)	4029.3 MW(th)

EFFICIENCY

Efficiency of fission heat conversion and transmission to steam generators ($3783/4029 \times 100$)	93.9
Overall station thermal efficiency ($1260/4029 \times 100$)	31.3

TABLE 5.1-1 (Cont.)

PHYSICS

Core data

Number of cells	740
Cell array	Square
Lattice pitch	28.575 cm
Core radius	438.66 cm
Core length	594.36 cm
Extrapolated length	606.0 cm
Average reflector thickness at midpoint	31.24 cm
Fuel load in core	166.06 MTU

Fuel channel data

Average geometric and physical properties
per cm length of fuel bundle

Volume

Fuel	41.086 cm ³ /cm
Void	1.686 cm ³ /cm
Sheath	6.979 cm ³ /cm
Coolant	34.185 cm ³ /cm
Pressure tube	19.589 cm ³ /cm
Air gap	27.084 cm ³ /cm
Calandria tube	5.721 cm ³ /cm
Moderator	680.201 cm ³ /cm
Total	816.531 cm ³ /cm

Mass

Fuel (UO ₂)	435.512 g/cm
Sheath	45.712 g/cm
Coolant	28.543 g/cm
Pressure tube	128.308 g/cm
Calandria tube	37.473 g/cm
Moderator	735.352 g/cm
Total	1410.90 g/cm

TABLE 5.1-1 (Cont.)

Dimensions

Pressure tube inner radius	5.1689 cm
Pressure tube outer radius	5.7479 cm
Calandria tube inner radius	6.4478 cm
Calandria tube outer radius	6.5875 cm
Homogenized fuel zone radius	4.944 cm
UO ₂ density	10.6 g/cm ³
Zr ₂ density	6.55 g/cm ³

BUILDINGS AND STRUCTURES**REACTOR BUILDING**

Form	Upright cylinder with flat base and hemispherical dome
Material	Reinforced concrete
Diameter	
Inside	44.5 m
Outside	47.24 m
Height (basement elev. to inside top of dome)	55.93 m
Design pressure, gauge	448 kPa
Wall thickness	1.37 m
Reactor (containment) building volume (approx.)	76,455 m ³

REACTOR, BOILER AND AUXILIARIES**REACTOR**

Type	Horizontal pressure tube
Coolant	Pressurized D ₂ O
Moderator	D ₂ O
Fuel	LEU 1.2%
Number of channels	740

TABLE 5.1-1 (Cont.)

REACTIVITY CONTROL

Main method	On-power refueling and moderator poison control
Trim	H ₂ O zone control assemblies and vertical absorber rods
Poison override	Vertical adjuster units
Shutdown:	
System No. 1	Gravity shut-off rods
System No. 2	Moderator poison injection

FLUX CONTROL

Axial	Zone control system
Radial	Differential fueling rate in zones
Distortion	H ₂ O zone

FUEL CHANNEL ASSEMBLIES

Quantity	740
Length	
Overall, including end fittings	10.8 m
Weight, dry and empty	415.4 kg
Cross-sectional areas (cold and unpressurized):	
Pressure tube	11,197
Fuel bundle	4972 mm ²
Coolant flow area	3421 mm ²
Flow:	
Total flow (all channels)	14.1 Mg/s
Maximum channel flow	22.9 kg/s
Coolant volume (each assy.)	0.07 m ³
Coolant exit velocity (central channel)	9.93 m/s

TABLE 5.1-1 (Cont.)

Temperature:

Reactor inlet header	298.9°C
Fuel inlet	298.9°C
Fuel outlet	339.4°C
Reactor outlet header	337.4°C

Pressure:

Reactor inlet header	15.52 MPa(a)
Fuel inlet	15.34 MPa(a)
Fuel outlet	14.34 MPa(a)
Reactor outlet header	14.14 MPa(a)

Maximum channel power	6.16 MW
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Steam quality:

Fuel outlet	2%
Reactor outlet header	2.9%

PRESSURE TUBES

Quantity	740
Material	Zirconium-2.5% niobium alloy
Dimensions (cold):	
Length trimmed for installation (approx.)	6.30 m
Inside diameter, minimum	103 mm
Cross-sectional area (internal)	8393 mm ²
Wall thickness, minimum	5.64
average	5.79
Corrosion and wear allowance	.20 mm
Weight, approx.	81.6 kg
Temperature	
Design temperature	344°C
Pressure:	
Design pressure	15.16 MPa(a)
Design stress	156.2 MN/m ²
Location of critical point design pressure/ allowable stress ratio	72% from inlet to core

TALBE 5.1-1 (Cont.)

CALANDRIA-END SHIELD ASSEMBLY

Overall length	7.77 m
Inside (core) length	5.94 m
Dry weight,	
Less shielding balls	361.5 Mg
With shielding balls	685.2 Mg
Calandria-moderator net volume	345.6 m ³
Calandria tube lattice	285.8 mm
Overpressure relief openings at top of calandria:	
Quantity	4
Size	457 mm

CALANDRIA SHELL

Form	Horizontal stepped cylinder
Material	All welded fabrication of 304L stainless steel to ASTM A240
Dimensions	
Main shell	
Inside diameter	9.4 m
Thickness	31.8 mm
Inside length	4 m
Subshell	
Inside diameter	9.2 m
Thickness	31.8 mm
Inside length	965 mm
Dry weight, approx.	46.0 Mg

TABLE 5.1-1 (Cont.)

REACTIVITY MECHANISM DECK

Material	Carbon steel top and bottom plates and webs, filled with ordinary concrete
Penetrations	
Flux detectors	52
Liquid zone controllers	12
Adjuster units	42
Shut-off units	56
Viewing ports	4
Solid control absorbers	8
Reference points	6

SHUT-OFF UNITS

Absorber elements

Quantity	56 (four uniformly distributed banks each containing 14 units)
Type	Stainless steel-clad cadmium tube
Reactivity worth, minimum with two out of service	-80 mk
Drive Mechanisms	
Type	Motor-driven winch with cable and rotary hydraulic damper

VERTICAL FLUX DETECTOR UNITS

No. of units	52
--------------	----

HORIZONTAL FLUX DETECTOR UNITS

No. of units	10
--------------	----

TABLE 5.1-1 (Cont.)

LIQUID ZONE CONTROL UNITS

Quantity	12 (8 two zone, 4 three zone)
Type	Vertical through-tubes, variable quantity water filling by helium pressure regulation

SOLID CONTROL ABSORBER UNITS

Absorber elements

Quantity	8
Type	Stainless steel-clad cadmium tube
Reactivity worth, approx.	10 mk
Drive mechanisms	
Type	Motor-driven winch with cable and rotary hydraulic damper

ADJUSTER UNITS

Quantity	42 (6 banks of 2, 4 banks of 3, 2 banks of 4, and 2 banks of 5)
Material	Stainless steel - clad cobalt
Total reactivity worth, approx.	15.2 mk
Drive mechanisms	
Type	Motor-driven winch with cable, direct coupling

MODERATOR

Type	Heavy water
Quantity, at 21°C	413.5 Mg
Isotopic purity	99.75% D ₂ O (by weight)

TABLE 5.1-1 (Cont.)

Volume (inc. D₂O management sys.)

Calandria	339.6 m ³
Reactivity mechanism and liquid injection shutdown	1.56 m ³
Main moderator and piping	24.64 m ³
Purification system	3.20 m ³
Deuteration and de-deuteration	0.6 m ³
D ₂ O collection	0.08 m ³
D ₂ O sampling	0.03 m ³
Liquid poison system	1.70 m ³
D ₂ O supply	.53 m ³
D ₂ O cleanup	1.53 m ³
Total	373.47 m ³

Volumetric expansion of moderator as temp. rises from
21 to 71°C

6.67 m³

Heat load

222 MW(th)

MODERATOR HEAT EXCHANGERS

Quantity	2
Type	Horizontal shell and tube
Heat load	110 MW
Coolant	Ammonia

MAIN COVER GAS CIRCUIT

Cover gas

Type	He and D ₂ O vapor
Volume	7.6 m ³
Make-up gas	
Type	He, 99.99% purity
Source	Gas bottles

TABLE 5.1-1 (Cont.)

MODERATOR LIQUID POISON SYSTEM

Long term reactivity control

Poison	Boric anhydride
Solubility limit (in D ₂ O at 20°C)	27 g/l

Startup reactivity control

Poison	Gadolinium nitrate
Solubility limit (in D ₂ O at 25°C)	2.35 Kg/l

PRIMARY HEAT TRANSPORT SYSTEM

Coolant

Type	Heavy water purity > 95 mol%
Quantity at 21°C	401.5 Mg

Flow

Arrangement	Bi-directional in adjacent channels
Total core	14.1 Mg/s
Maximum channel	22.9 kg/s

D₂O inventory

Steam generator tubes	78.7 m ³
Steam gen. inlet head	16.2 m ³
Steam gen. outlet head	15.6 m ³
Pump suction piping	11.7 m ³
Pump	33.4 m ³
Pump discharge piping	4.7 m ³
Inlet headers	9.3 m ³
Inlet feeders	16.9 m ³

TABLE 5.1-1 (Cont.)

Inlet end fittings	18.1 m ³
Channels	15.1 m ³
Outlet end fittings	18.1 m ³
Outlet feeders	37.0 m ³
Outlet headers	14.5 m ³
Steam gen. inlet piping	8.0 m ³
Total	293.3 m ³
Temperatures	
Reactor inlet header	298.9°C
Fuel inlet	298.9°C
Fuel outlet	339.4°C
Reactor outlet header	337.4°C
Mean (shutdown)	298.9°C
Mean (full power), approx.	318.1°C
Pressures	
Reactor inlet header	15.52 MPa(a)
Fuel inlet	15.34 MPa(a)
Fuel outlet	14.34 MPa(a)
Reactor outlet header	14.14 MPa(a)
Maximum channel power (nominal)	6.16 MW(th)
Steam conditions:	
Fuel outlet quality	2.0%
Outlet header quality	2.9%
STEAM GENERATORS	
Quantity	4
Type	Vertical U-tube with integral steam drum and preheater
Flow arrangement	Two independent banks each with two steam generators in parallel

TABLE 5.1-1 (Cont.)

Material	
Shell and drum	Carbon steel
Tubesheet	Inconel clad
Tubing	Inconel
Total heat transfer area per steam generator	4456 m ²
Design pressure	
D ₂ O side	17.24 MPa(g)
H ₂ O side	9.1 MPa(g)
Design temperature	
D ₂ O side	372°C
H ₂ O side	305°C
Coolant	
Volume per steam generator	28.24 m ³
Pressure at stm. gen. inlet	14.04 MPa(a)
Pressure at stm. gen. outlet	13.58 MPa(a)
Temperature:	
Inlet	336.3°C
Outlet	298.9°C
Inlet Quality	3.3%
Nominal tube outlet velocity	4.75 m/s
Flow (total for 4 stm. gens.)	14.11 Mg/s
Heat transferred (to H ₂ O sys.)	3820.5 MW(th)
Steam	
Total steam output (evap. rate)	2196 kg/s
Pressure at drum nozzle (full power)	7.72 MPa(a)
Temperature at nozzle	292.5°C
Quality at drum exit	99.75%
Feedwater	
Flow	2179.7 kg/s
Inlet temperature (full power)	232.2°C
Inlet temperature (minimum)	93.3°C
Inlet enthalpy	1001.7 J/g

TABLE 5.1-1 (Cont.)

Blowdown

Continuous blowdown rate, min. (.2% of steam flow rate)	4.4 kg/s
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Continuous blowdown rate, max. (1.0% of steam flow rate)	22 kg/s
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PUMPS

Quantity

4

Type

Vertical, single stage, double volute, single suction, double discharge, centrifugal

PRESSURE CONTROL SYSTEM

Heat transport system swell above zero power hot

38.38 m³

Temperature swell, above 260°C

20.24 m³

Void volume in HTS (full power)

Channels

0.10 m³

End fittings

2.40 m³

Outlet feeders

5.85 m³

Outlet header

2.51 m³

Steam generator inlet lines

1.5 m³

Steam generator head

3.24 m³

Steam generator tubing

0.22 m³

Total

15.82 m³

Mass of void

1549.7

Equivalent volume of liquid

2.32 m³D₂O inventory

Piping

2.66 m³

Pressurizer

12.74 m³

Degasser condenser tank

1.75 m³

Total

17.15 m³

TABLE 5.1-1 (Cont.)

SHIELD COOLING CIRCUIT

Coolant	Demineralized light water
Coolant flows	
To shield tank	584.8 m ³ /h
To end shield rings	126.1 m ³ /h
To ion exchanger	30.5 m ³ /h
To end shields	1209.9 m ³ /h
Total	1951.3 m ³ /h
Coolant temperatures	
End shields, inlet	60°C
outlet	66°C
Shield tank, inlet	49°C
outlet	54°C
Heat removed	10.9 MW(th)

SHIELD COOLING SYSTEM

Heat exchangers

Quantity	2 x 50%
Heat removal capacity	13.4 MW

Pumps

Quantity	2
Capacity, each pump	542 l/s

LIQUID ZONE CONTROL SYSTEM

Through tubes

Quantity	12 (vertically oriented)
Inside dia. approx.	114 mm

TABLE 5.1-1 (Cont.)

Zone control compartments

Quantity	28 (four assemblies have 3 compartments each, eight assemblies have 2 compartments each)
Capacity, each compartment	18 to 38 l
Outflow, each compartment	0.45 l/s approx.
Inflow, each compartment	0 to 0.9 l/s
System flow	16.7 l/s
Reactivity worth, approx.	7.5 mk
Maximum rate of change	± 14 mk
Cover gas	Commercial grade helium
Cover gas flow rates	
Bubbler header to each compartment	140 l/h
Balance header to each compartment	Variable
System	47 l/s

FUELING MACHINE HEAD

Quantity	2
Length	7 m
Weight (loaded and filled)	9.1 Mg
Volume of D ₂ O	636 l
Operating pressure, gauge	15.7 MPa
Operating temperature	93°C

STEAM SYSTEM

Steam mains

Quantity	4
Location	Between boilers and main steam header

TABLE 5.1-1 (Cont.)

Material	Carbon steel
Inside diameter	711.2 mm
Design pressure, gauge	9.1 MPa
Design temperature	305°C
Relief (Safety) valves	
Quantity	20
Type	Spring loaded
Function	To protect steam generators from overpressure
Material	Carbon steel
Operating pressure, gauge	7.62 MPa
Pressure set points (staggered)	9.1 to 9.29 MPa(g)
Relieving capacity (17 valves)	8,776,000 kg/h
Exhaust	To atmosphere
Fluid	Saturated steam
Inlet size	6 ins.
Outlet size	10 ins.
FUEL	
General	
Fuel	LEU 1.2%
Form	Fuel bundle assembly of 37 elements
Weight	
Total weight of UO ₂ in reactor	185 Mg
Total weight of U in reactor	163 Mg

TABLE 5.1-1 (Cont.)

Pellets

Quantity	Approx. 29 per element
Form	Cylindrical pellets with concave dished ends
Diameter	12.154 mm
Stack length	
Outer elements	470.314 mm
Intermediate elements	480.314 mm
Inner ring	480.314 mm
Center element	480.314 mm
Dish void (Volume %)	1.3%
Density	
Nominal	10.6 Mg/m ³
Range	10.45-10.75 Mg/m ³

FUEL DESIGN

Element assembly

Sheath

Quantity (per fuel bundle)	37
Material	Zircaloy-4
Weight	2.319 kg
Outside diameter	13.08 mm
Thickness (average)	0.419 mm
Sheath-fuel diametral clearance	0.089 mm

Sheath attachment joints

End caps to sheath	Resistance welded
Bearing pads to sheath	Brazed
Spacer pads to sheath	Brazed

TABLE 5.1-1 (Cont.)

Bundle assembly

Quantity of fuel bundles

In reactor 8880

Per fuel channel 12

Weight

 UO_2 per bundle 21.2 kg (approx.)

U per bundle 18.7 kg (approx.)

Nominal cold fuel dimensions

Length of bundle 495 mm

Outside diameter 102.4 mm

Weight of bundle 23.5 kg

X-sectional area, including sheath 4972 mm²X-sectional area, UO_2 (mid-plane) 4292.6 mm²Area ratio UO_2 /coolant 1.255

Outer element ring

No. of elements 18

P.C. diameter 86.61 mm

Intermediate element ring

No. of elements 12

P.C. diameter 57.51 mm

Inner element ring

No. of elements 6

P.C. diameter 29.77 mm

Center element

1 only

Element-to-element spacing

At mechanical spacers 1.55 mm

Element-to-pressure tube spacing 1.02 mm

TABLE 5.1-1 (Cont.)

Operating conditions

Estimated pressure drop per bundle	1000 kPa
Maximum UO_2 temperature	1900°C
Max. surface temp. of sheath	351°C
Max. axial force on fuel during refueling	22240 N
No. of bundles loaded per day, 100% load factor	12.1

HEAVY WATER MANAGEMENT

Inventory

Moderator and associated systems	413.5 Mg
Heat transport and associated systems	401.5 Mg
Fuel handling systems	4.5 Mg
Total	819.5 Mg

TURBINE GENERATOR AND AUXILIARIES

TURBINE

Quantity	1
Type	Tandem compound, one double flow hp cylinder with four external combined double moisture separator reheaters and three double flow lp cylinders

Steam conditions at hp valves

Quality	99.84%
Flow, to hp cylinder to reheater	2048.6 kg/s 121 kg/s
Pressure	7480 kPa(a)
Temperature	290.3°C
Enthalpy	2760 J/g

TABLE 5.1-1 (Cont.)

Steam conditions at hp exhaust

Quality	90.2%
Flow	780 kg/s
Pressure	3329.5 kPa(a)
Enthalpy	2513 J/g

Steam conditions at lp valves

Flow	1421 kg/s
Pressure	1197.6 kPa(a)
Temperature	274.2°C
Enthalpy	2989 J/g

Feedwater return flow

2172 kg/s

Feedwater enthalpy

1001.6 J/g

No. of stages of feedwater heating

6

Feedwater return temperature

232.2°C

GENERATOR

Quantity

1

Type

Direct coupled, hydrogen cooled rotor, water cooled stator

Power factor

0.9

Terminal voltage

25 kV

Frequency

60 Hz

Shaft speed

1800 rpm

Power output, gross

1343.6 MW

CONDENSER

Quantity

2

Type

Two shell, single pass

Surface area

30,796 m²

TABLE 5.1-1 (Cont.)

Coolant flow	16,478 l/s
Coolant inlet temperature	33.3°C
Coolant temperature rise	14.4°C
Back pressure	12.7 kPa(a)
Tube data	
Outer diameter	1.125 in
Thickness	20 BWG
Effective length	16.75 m
Water velocity in tubes	2.2 m/s
Material	90-10 Cu-Ni

PLANT NOMENCLATURE

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

5.1-33

0 50 100 200
SCALE IN FEET.

FIGURE 5.1-1
PLOT PLAN

5.1-34

5.1-35

FIGURE 5.1-3
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
PLAN EL. (-)17'-3"

5.1-36

FIGURE 5.1-4
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
PLAN EL. 0'-0"

5.1-37

FIGURE 5.1-5
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
PLAN EL. 14'-3"

5.1-38

FIGURE 5.1-6
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
PLAN EL. 37'-9"

FIGURE 5.1-7
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
PLAN EL. 60'-0"

5.1-40

FIGURE 5.1-8
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
PLAN EL. 79'-3"

FIGURE 5.1-9
GENERAL ARRANGEMENT
SAFETY-RELATED STRUCTURES
SECTION A-A

5.1-42

FIGURE 5.1-10
TURBINE BUILDING
GENERAL ARRANGEMENT
PLAN EL. 55'-0"

0 6 12 24
SCALE IN FEET

5.1-43

FIGURE 5.1-11
TURBINE BUILDING
SECTION B-B

FIGURE 5.1-12
SECURITY BUILDING
GENERAL ARRANGEMENT

FIGURE 5.1-13

HEAVY WATER REACTOR SIMPLIFIED FLOW DIAGRAM

5.1.1 Plant Structures

Design descriptions for structures are given in the following sections:

Reactor Containment Building (Section 5.1.1.1)
Control, Auxiliaries and Diesel Generator Building (Section 5.1.1.2)
Reactor Service and Fuel Handling Building (Section 5.1.1.3)
Penetrations Building (Section 5.1.1.4)
Control Room Emergency Air Intake Structure (Section 5.1.1.5)
Ultimate Heat Sink Basin (Section 5.1.1.6)
Plant Security Building (Section 5.1.1.7)
Administration and Warehouse Building (Section 5.1.1.8)
Fire Pump House (Section 5.1.1.9)
Pipe Tunnels (Section 5.1.1.10)
Holding Pond (Section 5.1.1.11)

5.1.1.1 Reactor Containment Building

◦ Functions

The Reactor Containment Building (RCB) houses the reactor, steam generators, reactor coolant system, and selected auxiliary systems and components. The RCB protects those systems and components against normal, abnormal and extreme environmental conditions, and externally generated missiles.

◦ Design Requirements

The RCB is designed to limit fission product release during normal and design basis events (DBE), including the loss-of-coolant-accident (LOCA) and main steam line break, and the postulated maximum hypothetical fission product release

(MHFPR). The RCB is designed to sustain the maximum pressure and temperature resulting from:

Double-ended rupture of the largest pipe in the reactor coolant system resulting in an immediate blowdown of one primary loop and subsequent release of the coolant from the unfailed loop.

Rupture of a nozzle assembly resulting in loss of coolant as well as fuel discharge from the pressure tube.

Main steam pipe rupture.

Rupture of moderator piping resulting in draining of D_2O from the calandria.

The containment design data are given in Table 5.1.1-1.

The RCB is provided with a new fuel loading facility, spent fuel transfer pool, and a refueling machine service area.

◦ Design Description

The containment houses the Nuclear Steam Supply System (NSSS). It is a seismic Category I reinforced concrete cylindrical structure with a hemispherical dome and a flat reinforced concrete base. The reinforced concrete base is 165 feet diameter, 10-feet thick, and founded 34 feet below finished grade. The upright cylindrical portion of the containment has an inside diameter of 146 feet, measures 183 feet - 6 inches from the top of foundation mat to the springline of the dome, and has a 4 foot 6 inch thick wall. The dome portion is a hemispherical-shaped head having an inside

height of 73 feet, 0 inches, and a 3 foot 6 inch thickness of reinforced concrete. The inside height from the top of the mat to the inside of the dome is 256 feet, 6 inches. The gross volume of the containment is 3,174,000 cubic feet with a free volume of 2,700,000 cubic feet.

A 3/8 inch thick carbon steel liner is applied to the inner surface of the cylinder, 1/2 inch thick liner on the inside of the dome, and a 1/4 inch thick liner on the top of the foundation mat. The liner is anchored to the inside face of the containment concrete. All seams of the liner are continuously welded and examined to assure that the liner provided an impervious membrane to prevent the release of containment atmosphere to the environs. The bottom liner is supported on top of the foundation mat and is protected by a 4 feet thick reinforced concrete slab which forms the floor of the containment.

A combination equipment hatch and personnel air lock is provided in the containment at El. 18 feet, 3 inches and is used as an entrance through which equipment is moved into the containment during construction.

A personnel air lock is provided in the containment at the control room floor (EL 60'-0") of the Control, Auxiliary, and Diesel Building (CAD). Piping, electrical penetrations, and HVAC ducts penetrate the wall at various locations. These penetrations are designed to provide a leak-tight connection during normal and abnormal operating plant conditions. Containment penetrations are housed in either the Penetrations building, the CAD building, or the Service Building. These buildings provide protection against environmental conditions and externally generated missiles for the penetrations, containment isolation system valves, and power and instrumentation wiring.

A 100-ton polar crane is provided for construction and maintenance operations. The crane rail is located near the spring line of the containment and is supported from the wall of the containment. The crane is designed to seismic Category I criteria with the requirements that the suspended load will not drop and that the crane will remain intact during an SSE event.

The containment is designed for 65 psig internal pressure and a temperature of 315°F. In addition, the containment is designed to withstand earthquake forces corresponding to an SSE of 0.30 g and an OBE of 0.15 g.

The containment will be constructed in accordance with the ASME Boiler & Pressure Vessel Code, Section III, Division 2 - Concrete Reactor Vessels and Containments. Interior concrete support structure is in accordance with the requirements of the ACI Standard 349-76, Code Requirements for Nuclear Safety Related Structures, supplemented with the ACI Standard 318-71, Building Code Requirements for Reinforced Concrete.

TABLE 5.1.1-1
CONTAINMENT DESIGN DATA

<u>Item</u>	<u>Data</u>
Type	Upright cylinder on a flat base with a hemispherical top head, reinforced-concrete, steel-lined.
Inside Diameter	146 Feet
Inside Height (base to springline of dome)	183 ft-6 in
Design Pressure	65 psig
Liner	1/4" (min.) thick carbon steel
Wall Thickness	4 ft-6 in
Dome Thickness	3 ft-6 in
Design Leak Rate	0.1 Volume-percent/24 hours

5.1.1.2 Control, Auxiliaries, and Diesel Generator Building

- Functions

The Control, Auxiliaries and Diesel Generator (CAD) building houses the control and electrical equipment, including support systems required for plant operation. The CAD building also houses essential features including the main control room, remote shutdown area, and the emergency diesel-generators.

- Design Requirements

The controls for nuclear steam production and its conversion into electricity are operated from the control room.

The remote shutdown room has the capability to initiate a reactor trip and other functions to safely shut down the reactor. This remote shutdown area is isolated from the main control room and containment with the requirement that it can be continuously occupied in the event an accident renders the main control room inaccessible.

- Design Description

The design of the CAD building evolved from previous studies conducted over a period of time to combine the functions of the control building, emergency power building, cable tray transition building, and the reactor containment building annulus. The CAD building houses the main control room, the safe shutdown control room, computer rooms, standby power supply (diesel-generator units), Class 1E medium and low voltage a-c distribution system, Class 1E d-c power distribution system and batteries, ECCS motor controllers, control rod

drive cabinets and the major portion of the Class 1E cable and raceway system.

Two separate and independent cable spreading rooms, one above (EL 79'-3") and one below (EL 37'-9") the control room (EL 60'-0"), are provided for the three Class 1E and one non-Class 1E power, control, and instrumentation channels. In addition, a network of four separate and independent vertical cable chases and containment penetration rooms are provided for the four channels.

A sector of a circular annulus building is provided between the CAD building and the containment. This structure houses a steel structure provided to support the various cable trays for their routing to the outside of the containment. The structure is designed to seismic Category I requirements and is provided with a minimum of 2 foot thick walls and roof.

Redundant heating, ventilating and air conditioning systems are provided for the control room and computer room areas. Redundant heating and ventilating systems are provided for the diesel-generator areas, switchgear area, containment penetration areas, and the cable spreading rooms. A separate and independent heating and ventilating system is provided for non-Class 1E equipment areas. Redundant control room air supply from remote air intakes is also provided.

The CAD building HVAC systems are housed in a seismic Category I structure on the roof (EL 91'-6").

Equipment access to any location within the CAD building is through the equipment hatch area which accommodates the largest piece of equipment to be removed. The diesel-engine

crankshaft and generator can also be removed. Personnel access is provided by one elevator and two stairwells. The stairwells and elevator allow egress from any point in the building through two separate paths. Building access is controlled so that all ingress and egress is thorough the guard house.

The CAD building is a reinforced concrete seismic Category I structure. A 4 foot thick reinforced concrete base slab is founded on fill concrete. The overall dimensions of the 7 story reinforced concrete building are 124 feet wide, 113 feet long, and 91 feet 6 inches high.

The annular sector building subtends an arc of 102 degrees, is 16 feet wide, and extends the full height of the CAD building (EL 91'-6"). The building is supported on a common mat with the CAD building.

The exterior walls, interior walls, and top slab of the building are reinforced concrete. The exterior walls are a minimum of 2 feet thick. Reinforced concrete walls separate the diesel-generators and support the upper floors. All walls and slabs are cast-in-place concrete with slabs supported on concrete beams, reinforced concrete columns and/or bearing walls. The top slab is covered with a roofing membrane.

Design of the structure is in accordance with the ACI Standard 349-76 entitled, "Code Requirements for Nuclear Safety Related Structures". Supplementary design is in accordance with the ACI Standard 318-71 entitled, "Building Code Requirements for Reinforced Concrete".

5.1.1.3 Reactor Service and Fuel Handling Building

◦ Functions

The Reactor Service and Fuel Handling Building houses all the auxiliary systems and emergency systems associated with the reactor plant. The building also houses the fuel handling equipment and related systems as well as all the systems associated with heavy water inventory control. The heavy water upgrading tower is located adjacent to the Reactor Service and Fuel Handling Building.

◦ Design Requirements

The Reactor Service and Fuel Handling Building is a Seismic Category I structure designed to house all radioactive or potentially radioactive systems and materials. The building houses all the emergency process systems required to mitigate the consequences of the following postulated accidents or abnormal occurrences:

1. Loss of coolant accident
2. Steam line break
3. Emergency shutdown due to loss of off-site power
4. Emergency shutdown due to safe shutdown earthquake

◦ Design Description

The Reactor Service and Fuel Handling Building is a seismic Category I reinforced concrete building located north-northwest of the Reactor Containment Building. Access into

the Service Building is thorough the Control/Diesel Generator Building. Provisions are made in the building layout to restrict access into any area that contains engineered safeguards equipment or high level radiation.

The building ventilation system is designed such that air flow is directed from areas of low level contamination into areas of potentially higher level contamination and is filtered before discharge.

The lower level, or basement of the Service building houses the following systems:

1. End shield cooling system
2. Emergency core cooling system
3. Containment spray system
4. Shutdown cooling system
5. Moderator purification system
6. Spent fuel pool cooling and purified system
7. Liquid radwaste system
8. Gaseous radwaste system
9. Solid radwaste and spent resin storage

The second level of the Service Building houses the following systems and facilities:

1. Spent fuel pool
2. New fuel storage
3. Fuel casking area
4. Shipping and receiving area
5. Heavy water storage
6. Deuteration-De-deuteration system
7. D₂O and helium addition system
8. Annulus gas system
9. Hydrogen recombiners
10. Drum storage and cleaning
11. Laundry and cleaning rooms

The exterior walls, interior walls, floor slabs and roof slab are reinforced concrete. The exterior walls are a minimum of two ft. thick, with the exception of the spent fuel pool wall which is nine and one-half ft. thick for all portions which are above grade. The spent fuel pool is 25 x 25 ft. and will accommodate one year accumulation of spent fuel plus one core of fuel. The Spent Fuel Reception Bay is 25 x 9 ft. The pools are lined in 1/4 inch thick Type 304 austenitic stainless steel plate. This plate forms a watertight membrane for the pools.

The floor slabs are cast-in-place concrete over metal deck and supported on steel framing. The roof slab is cast-in-place concrete over metal deck, covered with a roofing membrane and supported on steel framing.

Design of the structure is in accordance with the ACI Standard 349-76 entitled "Code Requirements for Nuclear Safety Related Structures". Supplementary design is in accordance with the ACI Standard 318-71 entitled, "Building Code Requirements for Reinforced Concrete".

5.1.1.4 Penetration Building

- Functions

The Penetration Building provides a structure to house and protect the main steam isolation and relief valves, and the safety related valves of the boiler feedwater lines. It also houses the heat exchangers and pumps of the moderator cooling system. The protection afforded by this building is from normal and abnormal environmental conditions and from externally generated missiles.

- Design Requirements

The Penetration Building is designed to afford protection and separation of the main steam and boiler feedwater pipe and valves. It is designed to withstand the pressure transient in the event of a double-ended steam line break.

The Penetration Building provides support, protection, and separation from the main steam and feedwater lines, for the Moderator heat exchangers and related pumps. The area containing the Moderator system components is provided with a D₂O vapor recovery system.

° Design Description

The Penetration Building houses the main steam and feedwater piping and valves, and portions of the Moderator system.

The building is a seismic Category I structure constructed from reinforced concrete. It is on a common foundation mat with the containment. The building shares a common wall with the containment on one side and has two-foot thick walls on the other three sides and on the roof.

The penetration building measures 120 ft. long, 32 ft. wide from outside of containment wall to edge of turbine building, and is 98 ft. high. Access to the Penetration building is through the CAD building at the control room floor elevation, 60 ft elevation. Entry to the main steam and boiler feedwater lines and valves is provided by doors through the separation walls in the building. Stairwells are provided in the building for access to any level. Doors into the Penetration building and doors through separation walls are of the bulkhead type designed to withstand the differential pressure loads without plastic deformation.

The main steam and boiler feedwater lines are separated from the Moderator system components by reinforced concrete walls. The area containing the main steam piping includes the power operated relief valves and safety relief valves. The relief valves are provided with vent stacks that permit the steam to exit to the environment. Boiler feedwater piping valves are contained within the penetration building.

Two components are furnished in the building to house the moderator coolers and pumps. The volume of air in these compartments will be conditioned and the compartments will

be furnished with D₂O vapor recovery systems. These two compartments will also house additional equipment for the operation of the plant.

5.1.1.5 Control Room Emergency Air Intake Structure

- ° Function

Each air intake structure is capable of providing adequate fresh air to ensure habitability of the control room in the event of an abnormal event during plant operation.

- ° Design Requirements

The emergency air intake structures are designed to:

1. Provide fresh air intake for the control room in event of radioactive contamination in the normal air intake stream.
2. Provide fresh air intake for the control room in the event of chemical contamination in the normal air intake stream.
3. Provide the above functions during SSE or other adverse environmental conditions.

- ° Description

The control room emergency air intake structures are two Seismic Category I reinforced concrete structures remotely located approximately 500 ft northeast and southwest of the control building. The air intake piping pit is located below grade and founded on rock ten ft. below grade. Each

structure has a base slab and walls of reinforced concrete ten ft. square and nine ft. deep, and has a volume of 500 cu. ft. Each structure has a top slab of reinforced concrete, which covers the piping pit and forms the floor of the security area around the air intake pipe. Each slab is at grade level and is 18 ft. square. The air intake structures are located inside the plant security fence.

5.1.1.6 Ultimate Heat Sink Basin

- Functions

The Ultimate Heat Sink Basin is provided to contain sufficient water to cool the nuclear plant equipment which is essential for a safe and orderly shutdown of the reactor and to maintain it in a safe condition.

- Design Requirements

The Ultimate Heat Sink Basin must be capable of accomplishing its safety function for a normal cooldown or an emergency cooldown following a loss-of-coolant accident (LOCA) without the availability of off-site power. The heat sink shall contain sufficient water that allows for drift and evaporation losses, to permit continuous use for 30 days without supplemental water from outside sources.

The Ultimate Heat Sink Basin shall be designed to seismic Category I requirements and to remain watertight under any abnormal environmental event.

- ° Design Description

The Ultimate Heat Sink Basin is constructed of reinforced concrete and is designed to remain watertight under operational (OBE) or safe-shutdown earthquake (SSE) loadings. The basin is coated with a waterproofing finish that acts as an impervious membrane to prevent leakage.

The basin is sized to contain 18,000,000 gallons of water and is provided with a 2 foot free-board above the nominal water level. The inside of the basin measures 150 feet by 400 feet by 42 feet deep. A pump sump is provided in the basin measuring 20 feet by 50 feet by 10 feet below the bottom of the basin.

Seismic Category I reinforced concrete supports for the cooling tower are provided in the basin. A seismic Category I pumphouse structure is provided to house the circulating water pumps.

5.1.1.7 Plant Security Building

- ° Functions

The purpose of the Plant Security Building is to provide a controlled means of preventing illicit passage of such objects as firearms, explosives, and incendiary devices which could aid in industrial sabotage to the facility or in the theft of special nuclear material.

- ° Design Requirements

The Plant Security Building provides a physical barrier to all sections of the plant as well as a central location for

monitoring the security of restricted areas and the plant grounds in general.

- ° Descriptions

The Plant Security Building is a 63 ft. by 53 ft. 6 inch building located at the perimeter fence of the main plant structures (see Figure 5.1.1-1). All ingress and egress of personnel in the main plant must pass through the Security Building and its associated security systems before entering sensitive plant structures or areas where radioactive materials are stored. The plant security system is monitored and operated from the security building.

5.1.1.8 Administration and Warehouse Building

- ° Functions

The Administration and Warehouse Building houses the general offices, conference rooms, auxiliary steam boiler plant, storage and warehousing, laboratories, machine shop, small shops, lockers and change rooms, and toilets.

- ° Design Requirements

The Administration and Warehouse Building has no safety or protection design requirements.

- ° Design Description

The design of the Administration and Warehouse Building provides a central location to conduct administrative routines for the successful operation of the power plant. In addition, this building houses the records that must be retained, by

law, for the operating life of the plant. The offices and record storage rooms are on the second floor of this two-story structure.

On the ground floor is the auxiliary steam boiler plant, a major-repair machine shop, small specialty shops, and warehousing space. Replacement parts and components, necessary for the continued operation of the plant, are stored in the warehouse section.

This is a two story structural steel framed structure 176 feet wide, 270 feet long, and 38 feet high. The structure is supported on reinforced concrete spread footings on rock. The reinforced concrete ground floor is located at grade. The intermediate floor is reinforced concrete supported on a metal deck on steel framing. The roof is concrete channel plank covered with a roofing membrane. The exterior walls are insulated metal siding, and the interior walls are either concrete block or metal partitions. Most areas are provided with suspended acoustical ceilings. A 20-ton crane is provided to service the machine shop, and suitable hoists are provided in the small shops. The building volume is approximately 1,500,000 cubic feet.

A HVAC system is provided to maintain environmental conditions of normally occupied spaces within the prescribed comfort zone and ensure that temperatures and humidity in other areas are suitable for material storage and equipment operation. Heating and air conditioning is provided by fan cooler heating units located throughout the building. Supply air is heated by steam preheaters and hot water coils in the supply air units. Hot water and chilled water for the fan cooler/heating units are supplied from central hot water heating systems and refrigerated chilled water systems.

Steam for preheating supply air to the building is supplied by the auxiliary boiler system. The HVAC machinery room is located on the roof of the structure.

5.1.1.9 Fire Pump House

- Functions

The Fire Pump House houses the fire protection pumps that furnished water through the entire plant area.

- Design Requirements

The structure provides support and weather protection for the fire protection pumps, piping, valves, and controls.

- Design Description

The Fire Pump House is a non-seismic Category I single story steel frame structure 28 feet wide, 76 feet long, and 17 feet high. The structure is supported on reinforced concrete spread footings. The ground floor is located at grade and is a reinforced concrete slab. The structure has insulated built-up roofing on metal deck supported on structural steel and the exterior walls are insulated metal siding. A ten foot square by 20 foot deep reinforced concrete pit is located at one end of the building. The approximate volume of the structure is 34,000 cubic feet.

Two foundations for 500,000 gallon fire-protection water storage tanks are located near the Fire Pump House. These are reinforced concrete ring walls 45 feet inside diameter, 1 foot 6 inches thick, and 5 feet 6 inches high. The area within the ring wall has a compacted sand fill bed to properly support the bottom plates of the tanks.

5.1.1.10 Pipe Tunnels

- Functions

Pipe tunnels are designed to provide protection for all plant piping between structures.

- Design Requirements

The pipe tunnels are reinforced concrete box type structures. Where the pipe in the tunnel is Seismic Category I the tunnel is also designed to be Seismic Category I.

- Design Descriptions

Pipe tunnels are reinforced concrete structures eight ft. wide ten ft. high and founded at various elevations. Where pipe tunnels extend beneath a building, the bottom floor of the building serves as the roof of the tunnel. In other areas the roof is cast-in-place concrete over metal deck and supported on steel framing.

5.1.1.11 Holding Pond

- Functions

The Holding Pond collects effluents from non-radioactive plant drains.

- Design Requirements

The holding pond is sized to collect the effluent and hold for a sufficient length of time, and treat if necessary, prior to disposal from the plant area.

° Design Descriptions

The holding pond is a non-seismic Category I reinforced concrete basin. It is located near the point where effluents will be discharged from the plant area.

The foundation of the basin is two feet thick and founded approximately 8 feet below grade. The basin is 67 feet square and eight feet deep. Walls of the basin are one-foot thick. Volume of the basin is approximately 36,000 cubic feet.

FIGURE 5.1.1-1
SECURITY BUILDING
GENERAL ARRANGEMENT

5.1.2 Reactor Plant Systems

The Reactor Plant Systems are composed of the following systems and components as presented in the following sections:

Calandria and End Shields (Section 5.1.2.1)

Main Moderator System (Section 5.1.2.2)

Moderator Liquid Poison System (Section 5.1.2.3)

Shutdown System No. 2 (Liquid Poison Injection System)
(Section 5.1.2.4)

Annulus Gas System (Section 5.1.2.5)

Moderator Purification System (Section 5.1.2.6)

Reactivity Control Systems (5.1.2.7)

5.1.2.1 Calandria and End Shields

◦ Function

The calandria vessel and its two end shields form an integral multi-compartment structure whose function is to contain the fuel channels, calandria tubes and the incore components of the reactivity control mechanisms. The calandria also maintains separation between the heavy water moderator and the light water used for shielding.

◦ Codes and Classification

The calandria will be designed to Safety Class 3 and Seismic Category I criteria. It will be designed and constructed to the criteria of the ASME B&PV Code, Section III, Class 3. The load path provided to support the pressurized tubes must have all components to Class 1 requirements. This includes some parts of the calandria and end shields.

° Description

Figure 5.1.3-1 of Section 5.1.3.1 shows the orientation of the reactor and thus the calandria with respect to the Reactor Coolant System. Figures 5.1.2-1 and 5.1.2-2 show sectional views of the calandria and typical penetration locations. Table 5.1.2-1 contains the calandria and end shield design parameters.

Calandria

The calandria shell is a horizontal austenitic stainless steel cylindrical vessel which contains the heavy water moderator/reflector and supports the various incore components of the reactivity mechanisms. It will consist of a main shell and two sub-shells. The ends of the calandria will have reduced diameters which will be welded to circular tubesheets. The tubesheets form the inner end of the end shields. There will be two annular plates connecting the two shells which will stiffen the calandria shell and will provide a flexible member at each end to allow for mechanical and thermal expansion. The calandria will have nozzles for reactivity mechanisms and four nozzles for moderator cooling. Cool moderator enters the calandria through two flow connections at each side of the calandria and leaves through two nozzles at the bottom of the calandria.

Four pressure relief pipes will be provided at the top of the calandria to prevent overpressurization of the calandria shell in the event of severance of a pressure tube and calandria tube. Each relief pipe will extend through the light water shield in the calandria vault to a rupture disc located in the shielding deck.

A total of 740 calandria tubes will span the calandria axially on a 11.25 inch square pitch to form a circular

lattice array. The calandria tubes will pass through the calandria and have expanded ends that are roll expanded with stainless steel inserts into the calandria tubesheets of the end shield forming high integrity leak tight "sandwich" joints.

The tubes will be fabricated from zirconium alloy. They will provide an annular gap around the fuel channel assemblies which will be filled with dry inert gas. This gas provides thermal insulation between the hot coolant and the relatively cool moderator. Refer to Figure 5.1.2-3 for details. A description of the annulus gas system is found in Section 5.1.2.5.

The calandria tubes will act as axial stays between the two end shields. The calandria tubes will be separated from the fuel channels by toroidal (garter) springs which also permit some load transfer between the pressure tube and the calandria tube.

The diameter of the calandria tubes will be limited by the lattice pitch and by the size of the pressure tube and the gap needed for insulation between the coolant and moderator fluids.

The thickness of the calandria tubes will be established to preclude collapse under external pressure and stressing from other applied loads during normal operations and liquid poison injection. The calandria tubes will be seam welded and annealed.

End Shields

Each end shield will be composed of a calandria tubesheet, a refueling machine tubesheet, 740 lattice tubes, and a connecting shell as shown in Figure 5.1.2-4. The lattice tubes will be

welded at each end to the tubesheets. All material will be austenitic stainless steel.

The end shield will be filled with carbon steel balls varying from 0.5 - 0.375 in. in diameter and demineralized light water (60/40 ratio) to permit access during reactor shutdown. The water will be circulated through the end shields to provide for removal of heat generated within the end shield by conduction through the calandria tubesheet, and convection from the end fittings containing the primary coolant.

Shield Cooling System

The shield cooling consists of two, 100% capacity pumps and two 50% capacity heat exchangers arranged in a series parallel arrangement. One pump is normally operating with the other on standby. The second pump starts automatically if the operating pump stops. The operating pump discharges to both heat exchangers. Component cooling water provides the secondary side fluid for cooling the shield cooling water. The shield cooling system is constructed to Safety Class 3, ASME B&PV Code Section III, Class 3 and Seismic Category I criteria. However, the system is not required to operate past an accident and is not provided with emergency standby power.

Support Structure

As shown in Figure 5.1.2-5, an integral support structure will be provided which consists of a shell and a plate. Both will be stainless steel and will be welded to the circumference of each end shield and to the carbon steel embedment. The carbon steel embedment will be grouted into the vault end walls. The support structure will be designed

to allow for differential movements between the reactor assembly and the vault which result from thermal and loading effects.

The support rings which are embedded in the calandria vault walls will consist of 1.5 inch thick carbon steel plates, 1 inch thick carbon steel shell and cooling coils. They support the calandria assembly via the support plate and shell and transmit the loads into the concrete vault. To ensure that the concrete temperature will be below 150°F from heat gained by nuclear heating, both water cooling coils and thermal shield slabs are incorporated in the design. Eight inch thick shielding slabs provide protection against radiation streaming through the air gap.

Details of the embedment ring are shown in Figure 5.1.2-6.

°
Design Parameters

Refer to Table 5.1.2-1 for design parameters.

TABLE 5.1.2-1
CALANDRIA AND END SHIELD DESIGN PARAMETERS

<u>Item</u>	<u>Data</u>
General:	
Overall length	25.5 ft
Inside (Core) Length	19.5 ft
Dry Weight, less shielding balls	403 tons
with shielding balls	775 tons
Calandria - moderator net volume	12,210 ft ³
Calandria Tube lattice	11.25 in
Overpressure Relief Openings At top of Calandria:	
Quantity	4
Size	18 in
<u>Calandria Shell</u>	
Form	Horizontal Stepped Cylinder
Material	All Welded Fabrication of Stainless Steel to Carbon Steel
Dimensions:	
Main Shell:	
Inside Diameter	30 ft. 10 in
Thickness	1.125 in
Inside Length	13 ft. 2 in
Subshell:	
Inside Diameter	30 ft. 2 in
Thickness	1.125 in
Inside Length	38 in
Dry Weight, Approx.	45 tons
<u>End Shields</u>	
General:	
Quantity	2
Form	Cylindrical vessel with inner and outer tubesheets filled with steel ball shielding and penetrated by 740 lattice tubes.

TABLE 5.1.2-1 (Cont.)

<u>Item</u>	<u>Data</u>
<u>Material:</u>	
Vessel	Stainless Steel to Carbon Steel
Tube Sheets	Stainless Steel to Carbon Steel
Filler	Carbon Steel Balls
Lattice Tubes	Stainless Steel
<u>Weight, Each (Approx.):</u>	
Unfilled	125 tons
Filled	301 tons
Overall length (including both tube sheets)	36 inches
Inside Diameter	30 ft 2 inches
Thickness:	
Shell	1 inch
<u>Tube Sheet:</u>	
Inner (Calandria Side)	2 inches
Outer (Refueling Machine Side)	3 inches
Lattice Arrangement	740 tubes on 11.25 in square pitch
Radius of Outermost Lattice Position	14 ft. 5.7 inches
Minimum Tubesheet Rim Ligament	5.51 inches
Lattice Tube:	
Inside Diameter	7.5 inches
Outside Diameter	8 inches
<u>Shield Coolant:</u>	
Type	Demineralized Light Water
Inlet Temperature	140°F
Outlet Temperature	150°F
Design Pressure	90 psig 70 psig due to balls: remainder water
Operating Pressure	30 psig
Design Temperature	250°F
Total Flow through both shields	5350 gpm

TABLE 5.1.2-1 (Cont.)

<u>Item</u>	<u>Data</u>
<u>Pumps:</u>	
Quantity	2
Fluid	Demineralized Water
Type	Centrifugal
Duty	Continuous
Flow Rate	5350 gpm
Operating Temperature	150°F
Motor Type	3 Phase, Induction
Materials	Carbon Steel
<u>Heat Exchanger:</u>	
Quantity	2
<u>Flow Rate:</u>	
Tube Side	2675 gpm
Shell Side	3350 gpm
Heat Rate	26.33×10^6 BTU/hr
<u>Fluid:</u>	
Tube Side	Demineralized Water
Shell Side	Component Cooling Water
<u>Temperatures:</u>	
Tube Side: Inlet/Outlet	150°F/140°F
Shell Side: Inlet/Outlet	105°F/117°F
Materials	Carbon Steel
<u>Calandria Tubes:</u>	
<u>General:</u>	
Quantity	740
Material	Zircaloy-2 Annealed

TABLE 5.1.2-1 (Cont.)

<u>Item</u>	<u>Data</u>
Dimensions:	
Length (overall installed)	19 ft. 8 inches
Inside Diameter, minimum	5.077 inches
Wall Thickness, minimum	0.054 inches
Outside Diameter, maximum	5.209 inches
Average Weight, Each	49.8 lb.
<u>End Shield Support</u>	
General:	
Quantity	2
Form	Cylindrical ribbed shell grouted in concrete vault end wall with circumferential shell and support plate connected to end shield outer tube sheet.
<u>Material:</u>	
Embedded parts	Carbon Steel
Circumferential Shell	Weldment of carbon steel and stainless steel to carbon steel
Diaphragm Plate	Stainlees Steel to carbon steel
Dimensions:	
Diameter Overall	35 ft. 9 inches
I. D. Circumferential Shell	30 ft. 6 inches
Thickness of Support Plate	0.75 inches
Thickness of Support Shell	1 inch
Weight, Each	45 tons

5.1.2.2 Main Moderator System

° Function

The function of the heavy water moderator in the calandria is to slow down the neutrons liberated in the fission process at high energy to the thermal energy range. Other functions of the moderator are to provide:

- a) Cooling
- b) Purification and control of soluble neutron poisons.

° Description

A flow diagram of the main moderator system is shown in Figure 5.1.2-2. The system is comprised of two 100% capacity pumps and two 50% capacity heat exchangers. The pumps, arranged in parallel, are in series with the two heat exchangers, which are in parallel with respect to each other. Connections are provided for purification, liquid poison addition, heavy water collection, D₂O supply and sampling systems. The cooling system for the moderator is described in Section 5.1.7.10.

1) Electrical Power

The moderator pump motors will not be connected to the standby onsite power supply. However, each pump will have a pony motor capable of driving the pump at 25% speed and this pony motor will be connected to the standby onsite power supply. In the event of a failure of the normal power supply a pony motor will start and operate a moderator pump at lower speed so that the

circulation of heavy water will be continued. Cooling water flow to the heat exchangers will also continue at a lower rate following a failure of power supply. The rate of heat removal will be sufficient to limit the increase of moderator temperature in the calandria to an acceptable value during a failure of normal power supply and subsequent reactor shutdown.

2) Instrumentation and Controls

The temperature of the D_2O in the calandria will be held constant at approximately 200 F by circulating the D_2O from the bottom of the calandria through the main moderator heat exchanger and returning it through nozzle jets inside both sides of the calandria at the horizontal diameter.

The feedback control loop for the main moderator system will be used to control the D_2O temperature at the calandria outlet. A control program in the station digital computer positions the heat exchanger cooling water valves to maintain a constant temperature at the calandria outlet. The feedback signal is the median signal from three temperature sensors located at the calandria outlet piping. Temperature sensors located at the outlet of the heat exchangers provide control room indication and alarm on high temperature. The alarm setpoint will be a function of the reactor power.

Other instrumentation includes:

- a) calandria level
- b) calandria head tank level

- c) pump suction and discharge pressure
- d) pump motor temperature
- e) pump bearing and shaft vibration.

3) Materials and Codes

All components will be austenitic stainless steel with the exception of the heat exchanger tubes, which will be made from titanium.

The main moderator system will be a Safety Class 3 and Seismic Category I system. It will be in accordance with the Section III of the ASME B & PV Code, Class 3. Piping penetrating containment and valves provided for containment isolation will be constructed in accordance with ASME B&PV code, Section III, Class 2 and Safety Class 2.

° Components

Main Moderator Pumps

The pumps in the main moderator system will be electric motor driven, centrifugal pumps. Each of the two pumps will be rated at 100% of the normal required by heavy water flowrate. The pumps will be designed to function during and following a safe shutdown earthquake, and will be controlled from the main control room. Normal operation is one pump running and the other on standby. Failure of the running pump causes the standby pump motor to start automatically.

Each pump main motor will be directly coupled to the pump shaft so that the pump and the motor operate at the same speed. The main motors are three phase induction motors and will be totally enclosed and cooled by a water cooled heat exchanger. Heaters will be provided to prevent condensation of moisture inside the enclosure when a main motor is not operating. Each pump pony motor will be directly coupled to the main motor shaft and drives the pump at 25% of the main motor speed. The pony motors will be three phase induction motors connected to the standby onsite power supply, and will be totally enclosed and fan cooled. Heaters will be provided to prevent condensation of moisture inside the enclosure when a pony motor is not operating.

Double mechanical seals operating in series-tandem will be provided on each of the pump shafts. One of the two mechanical seals will be capable of sealing the pump shaft in the event of failure of the other seal. The mechanical seals will be cooled by a heavy water flow from the pump discharge. An atmospheric seal will be designed to function as a back-up shaft seal in the event of failure of both of the mechanical seals.

Heat Exchangers

Each of the two heat exchangers will be rated at 50% of the maximum moderator system heat load and will be capable of up to 100% of the normal heavy water flow rate. The tube side will be constructed in accordance with the requirements of Section III of the ASME Code, Class 3. The heat exchanger will meet Safety Class 3 and Seismic Category I requirements.

Design Bases

Refer to Table 5.1.2-2.

TABLE 5.1.2-2
MAIN MODERATOR SYSTEM DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
Main Circuit	
General:	
Moderator	Heavy Water
Calandria Inlet Temperature	150°F
Calandria Outlet Temperature	200°F
Moderator Flow Rate	27,400 gpm
Cooling Capacity	222 Mw
Applicable Code	ASME Section III, Class 3
Safety Class	3
Seismic Category	I
Heat Exchangers:	
Quantity	2
Type	
Duty	Continuous
Capacity	50%
Heat Transferred	748.2×10^6 BTU/hr
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	14,100 gpm
Inlet Temperature	200°F
Outlet Temperature	150°F
Design Pressure	300 psig
Design Temperature	250°F
Shell Side Data:	See Section 5.1.7.10
Pumps:	
Quantity	2
Type	Centrifugal
Duty	Continuous
Capacity	100%

TABLE 5.1.2-2 (Cont.)

<u>Item</u>	<u>Value</u>
Pump Data:	
Fluid	D ₂ O
Flow Rate	28,200 gpm
Operating Temperature:	200°F
Design Pressure	240 psig
Design Temperature	250°F
Main Motor Data:	
Type	3 Phase, Induction Motor
Pony Motor Data:	
Type	3 Phase, indication motor
Speed	1/4 Main Motor Speed
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME-Section III, Class 3
Tanks:	
Name	Calandria Head Tank
Quantity	1
Type	
Capacity	100 ft ³
Design Conditions:	
Operating Pressure	0 psig
Operating Temperature	110°F
Design Pressure	15 psig
Design Temperature	250°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

5.1.2.3 Moderator Liquid Poison System

- Function

The liquid poison system will be used:

- a) To add negative reactivity to the moderator to allow for excess reactivity in new fuel.
- b) To compensate for the absence of fission product xenon during a startup after a long shutdown.
- c) To provide a means of adjusting reactivity in conjunction with other reactivity control devices.

- Codes

The Moderator Liquid Poison System will be constructed to Safety Class 3 and Seismic Category I criteria. The system will be constructed to the ASME B&PV Code Section III Class 3.

- Description

A flow diagram of the Moderator Liquid Poison System is shown in Figure 5.1.2-3. The system consists of two chemical storage tanks, one delay tank and the necessary interconnecting piping and equipment to manually add the required reactivity control chemicals to the Main Moderator System.

The liquid poison system is used to control the reactivity of the heavy water moderator by adding soluble chemicals with large neutron capture cross-sections. Boron, as HBO_3 , and gadolinium, as $\text{Gd}(\text{NO}_3) \cdot 6\text{H}_2\text{O}$, are the chemicals used.

Boron is used to compensate for the lack of neutron absorbing fission products in the fuel at initial plant start-up, as a poison shim during refueling, for long term shutdowns and for any other situation which requires poison for a long term basis.

Gadolinium is used on reactor start-up following a long shutdown since it is burned to a lower neutron capture cross section at approximately the same rate as the fission product xenon is produced.

Heavy water will be supplied to the Poison Storage tanks from the outlet of the moderator heat exchangers through a delay tank which allows nitrogen-16 and oxygen-19 to decay. Both poison storage tanks will have agitators to aid in the preparation of the poison solutions. Separate lines from the mixing tanks will carry the poison to the suction of the moderator pumps. Needle valves will be manually adjusted to obtain the required rate of poison flow from the tanks and air operated valves will start or stop the poison flows as required by operation. Poison addition valves will be controlled from the control room.

All components in contact with heavy water will be austenitic stainless steel.

° Design Parameters

Refer to Table 5.1.2-3.

TABLE 5.1.2-3
MODERATOR LIQUID POISON SYSTEM
DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
General:	
Long Term Reactivity Control:	
Poison	Boric Anhydride
Solubility Limit in D ₂ O at 68°F	24,000 ppm
Concentration of Boron in the Moderator to Simulate - 28.5 mk*	3 ppm
Concentration of Poison in the Poison Tank	7,600 ppm
Rate of Poison Solution Addition to Achieve - 0.5 mk/min	2.5 gpm
Startup Reactivity Control:	
Poison	Gadolinium Nitrate Hexahydrate
Solubility Limit in D ₂ O at 68°F	2,100,000 ppm
Concentration of Gadolinium in the Moderator to Simulate - 28.5 mk*	0.75 ppm
Concentration of Poison in the Poison Tank	3,000 ppm
Rate of Poison Solution Addition to Achieve 0.5 mk/min	1.5 gpm
Pumps:	
Name	Poison Solution Sampling Pumps
Quantity	2
Type	Canned Centrifugal
Duty	Intermittent
Capacity	100%
Pump Data:	
Fluid	Poison Solution in D ₂ O
Flow Rate	1.25 gpm
Operating Temperature	Ambient
Head	15 ft

*28.5 mk is the reactivity required to compensate for complete absence of equilibrium xenon in the reactor fuel.

TABLE 5.1.2-3 (Cont.)

<u>Item</u>	<u>Value</u>
Design Pressure	100 psia
Design Temperature	250°F
Motor Data:	
Type	3-Phase, Squirrel Cage Induction Motor
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Safety Class	3
Seismic Category	I
Tanks:	
Name	Poison Mixing Tanks
Quantity	2
Type	Vertical Cylinder with Dished Heads
Capacity	26 ft ³
Design Conditions:	
Operating Pressure	5 psig
Operating Temperature	100 to 200°F
Design Pressure	100 psig
Design Temperature	250°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Name	Delay Tank
Quantity	1
Type	Horizontal Cylinder with Dished Heads
Capacity	3.6 ft ³

TABLE 5.1.2-3 (Cont.)

<u>Item</u>	<u>Value</u>
Design Conditions:	
Flow Rate	18 gpm
Operating Pressure	80 psig
Operating Temperature	200°F
Design Pressure	100 psig
Design Temperature	250°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Safety Class	3
Seismic Category	I
Mixer:	
Name	Poison Mixer
Quantity	2
Type	Electric motor-driven agitator, mounted in mixing tank.
Motor Data:	
Type	3 Phase, Squirrel Cage, Induction Motor
Construction:	
Material in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	None

5.1.2.4 Shutdown System No. 2 (Liquid Poison Injection System)

- Function

The Liquid Poison Injection System will be provided as a backup to the Shutdown System No. 1. It will rapidly terminate reactor power operation upon detection of specified trip parameters by injecting a strong neutron absorber, gadolinium nitrate, into the moderator.

- Codes and Classification

The Liquid Poison Injection System will be designed to Safety Class 1 and Seismic Category I criteria. It will be constructed in accordance with the ASME B&PV Code, Section III, Class 1.

- Description

The Shutdown System No. 2, as shown in Figure 5.1.2-9, consists of six helium storage tanks each containing helium at high pressure and each connected to one of the six poison injection tanks filled with 8000 ppm gadolinium nitrate dissolved in heavy water. Interconnections will not be provided between either the poison nor the helium tanks except for the makeup of the helium supply. Reverse flow from the helium tank to the makeup source will be prevented by redundant check valves in the helium lines. During normal reactor operation the helium tanks will be isolated from the poison tanks by redundant quick-opening air operated valves. The valves will be fail closed upon loss of air. Each air operator is provided a type A source of air per ANSI N-59.3-1977 to ensure operability of the valves. The helium storage tanks, quick opening valves and poison tanks

will be located outside the reactor vault where they are accessible for testing during power operation. The manual valves on the poison tank entrance and discharge will be locked open during normal operation. Administrative control will ensure that only one tank train be tested at one time.

The solution of gadolinium nitrate will contain approximately 1000 Milli-K of negative reactivity in the moderator when completely dispersed. Operation of 3 out of the 6 tank trains will fully terminate reactor operation.

Each poison tank contains a floating solid polyethylene ball. When an injection is initiated, the helium driving gas forces the gadolinium nitrate solution to the calandria. When the tank is empty, the ball is driven to the tank bottom where it blocks off the tank outlet and thereby prevents helium from entering the calandria and overpressurizing the moderator*. The ball will be designed to back seat at the top of the tank to restrict the movement of poison due to the variations in calandria moderator level.

Each tank will be connected to a horizontal nozzle in the moderator by a 4 inch, schedule 40 stainless steel pipe. Each injection nozzle assembly consists of a Zircaloy-2 extruded section containing several hundred injection holes. A stainless steel bellows is used to seal the moderator and to reduce sag and vibration on the nozzle and a second bellows to seal against light water intrusion from the shield tank.

There will be a liquid to liquid interface between the poison solution and the moderator. This interface will be located below each poison tank in the downstream vertical

*Calandria overpressure protection is adequate to prevent the helium in this system from overpressurizing the calandria greater than 110% of calandria design pressure assuming failure of all six polyethylene balls.

leg of a U-shaped section of a pipe. The interface will move by diffusion from the higher density gadolinium nitrate solution towards the calandria. Conductivity probes will be provided to detect when the interface is approaching the top of the U section.

Upon alarm from the conductivity probes, the injection lines will be back-flushed and the associated poison tanks refilled with gadolinium nitrate solution.

TABLE 5.1.2-4
SHUTDOWN SYSTEM 2 (LIQUID INJECTION) DESIGN DATA

<u>Item</u>	<u>Data</u>
Injection Nozzle Assembly:	
Quantity	6
Material	Zircaloy 2
Inside Diameter	3.826 inches
Bellows Assembly:	
Quantity	6
Material	Stainless Steel
Poison Tank:	
Quantity, per unit	6
Description	Cylindrical Shell Capped at Each End by Heads, with Ball Valve
Location	Outside Wall of Reactor Vault
Material:	
Shell and Heads	Austenitic Stainless Steel
Ball	Solid Polyethylene
Codes and Classification:	
ASME Code Class	1
Safety Class	1
Seismic Category	I
Volume	7.4 ft ³
Connections:	
Inlet	3 inch Sch. 80
Outlet	4 inch Sch. 80
Design Pressure	1800 psig
Operating Pressure:	
During Normal Reactor Operation	8 psig
During Poison Injection	1500 psig
After Poison Injection	480 psig
Design Temperature	200°F
Ambient Temperature	40-150°F

TABLE 5.1.2-4 (Cont.)

<u>Item</u>	<u>Data</u>
Contents:	
During Normal Reactor Operation	8000 ppm of Gadolinium in D ₂ O
Following Poison Injection	Helium
Helium Supply Tanks:	
Quantity	6
Description	Cylindrical Shell Capped at Each End by Hemispherical Heads
Location	Wall of Reactor Vault
Material:	
Shell and Heads	Austenitic Stainless Steel
Codes and Classification:	
Applicable Code	ASME Section III
Safety Class	1
Seismic Category	I
Volume	18 ft ³
Connections:	
Inlet	3/4 inch Sch. 80
Outlet	3 inch Sch. 80
Design Pressure	1800 psig
Design Temperature	200°F
Normal Operating Pressure	1100-1200 psig
Ambient Operating Temperature	40-150°F
Contents:	
During Normal Reactor Operation	Pressurized Helium Gas
Gadolinium Mixing Tank:	
Quantity	1
Description	Cylindrical Shell Capped at Each End by Dished Heads
Location	Reactor Building
Material	Austenitic Stainless Steel
Volume	30 ft ³

TABLE 5.1.2-4 (Cont.)

<u>Item</u>	<u>Data</u>
Connections:	
Inlet	3/4 Inch Sch. 40
Outlet	3/4 Inch Sch. 40
Design Pressure	100 psig
Design Temperature	200°F
Operating Pressure	40 psig
Operating Temperature	40-150°F
Contents	8000 ppm of Gadolinium in D ₂ O and Helium
Applicable Code	ASME Section III, Class 3
Safety Class	3
Seismic Category	I

5.1.2.5. Annulus Gas System Design Parameters

- Function

The purpose of the annulus gas is to restrict heat transfer from the fuel channel pressure tubes to the moderator and provide a means for pressure tube leakage detection.

- System Description

A schematic flow diagram of the Annulus Gas System is shown in Figure 5.1.2-10, Sheet 1. The method of distribution to the individual nozzles is shown in Figure 5.1.2-10, Sheet 2.

The fluid employed will be dry carbon dioxide gas which will be supplied from high pressure storage cylinders. Provisions will be made to sample the gas in the system to detect moisture, which would serve as an indication of leakage from the moderator, or more likely, from the heat transport system into the annulus.

A regulated supply of gas will be directed to headers and branches at one face of the reactor and will be distributed by tubes so the annuli in a series-parallel arrangement. A ring header at the other face of the reactor will collect the discharge from the annuli and a branch line will direct a flow to the sample station and to the Reactor Building ventilation exhaust.

Normally, flow within the system will be restricted to make-up flow. An air to gas heat exchanger will cool the discharge before sampling or release to the ventilation exhaust. The system piping will be stainless steel and will be constructed to ASME Section III Class 3 Code requirements. The system

will be designed to Safety Class 3 and Seismic Category I criteria.

- Instrumentation and Control

System pressure will be monitored and displayed in the control room and high and low pressures will be annunciated. All other measurements and all of the system valves will be in normally accessible areas of the plant. The system will be protected against overpressure by rupture discs and safety valves.

- System Design Parameters

See Table 5.1.2-5.

TABLE 5.1.2-5
ANNULUS GAS SYSTEM
DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
Fluid	CO ₂
Estimated Makeup	3.5 ft ³ min.
Purge Flow Rate	40 ft ³ min.
Operating Pressure	2 psig
Design Pressure	60 psig
Design Temperature	450°F
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Safety Class	3
Seismic Category	I

5.1.2.6 Moderator Purification System

- Functions

The Moderator Purification System:

Maintains the purity of the moderator water,
Controls the pH of the water,
Adjusts the concentration of soluble poisons in the moderator water.

- Design Requirements

The Moderator Purification System provides:

Variable system flow of moderator water from 800 to 4000 lbs/min depending on the number of ion exchange columns used. The moderator purification heat exchanger controls the moderator water at approximately 110°F.

This is not an engineered safeguards system but is constructed to seismic Category I and ASME B&PV Code, Section III, Class 3 requirements due to the cost of D₂O product.

- System Description

The Moderator Purification System removes impurities, controls the pH, and adjusts the concentration of soluble poisons in the moderator water. The system is not safety-related but all pressure-boundary components, and their supports, are constructed to seismic-Category I and ASME B&PV Code Section III, Class 3 requirements. The purification system is located in the basement of the Reactor Service and Fuel Handling Building (RSFHB). A schematic of the system is

shown in Figure 5.1.2-11 and a flow diagram in Figure 5.1.2-12.

The system has a flow capacity varying from 800 to 4000 lbs/min depending on the number of ion exchange columns placed into service. The temperature of the system is maintained at approximately 110°F through control of cooling water supplied to the moderator purification heat exchanger.

The ion exchange columns each have a capacity for approximately 9 cubic feet of ion-exchange resin. The columns are provided with screens designed to retain the resin, prevent plugging, and to withstand relatively large pressure drops. In the unlikely event of screen blockage, the column design permits removal of either the inlet or outlet screen. The column is provided with a conical bottom to facilitate removal of the resin from the column as a slurry.

The cartridge-type purification filter is provided to remove any insoluble debris of particles larger than 5 microns from the moderator water.

A purification system cooler is provided to cool the moderator water purification flow that originates upstream of the moderator heat exchanger from 200°F to 110°F before the D₂O enters the ion exchanger columns.

The wetted surfaces of the moderator purification system components are constructed with austenitic stainless steel.

A backup strainer is provided at the outlet of the system to prevent resin from entering the moderator system. The resin bead size ranges from 0.38 to 0.60 mm and the screen is sized to retain any particle larger than 0.2 mm.

A conductivity cell and a hypodermic sample pot is provided on a sample line downstream from the ion exchanger columns to permit the plant operator to determine the useful life of the resin in each of the columns.

There are five exchange columns used as follows:

- ° One ion exchange column, which contains a mixed resin bed of equal capability for removal of cations and anions, is sized to provide adequate cleanup of corrosion products at a flow of 87 gpm at 110°F. The resin used will not alter the boron content of the system but will remove gadolinium if it is present. The cleanup half-life for the maximum flow is approximately 10.5 hours.
- ° Three ion exchange columns containing a mixture of cation and anion in the ratio of 1:6 are used to remove boron from the moderator water.

A flow control valve located in the purification system return line enables the operator to remove boron from the moderator system at a controlled rate. Poison may be added by valving in a column already saturated with boron to release boron to the moderator at a smaller rate than is possible with the liquid poison system.

- ° The remaining two ion exchange columns are used for gadolinium removal. The columns are filled with a mixture of equal parts of cation and anion. Gadolinium removal is required after normal addition of gadolinium to meet reactivity needs such as xenon transients, and after a poison injection shutdown.

In the first case only one column is required for gadolinium removal with a moderator flow rate of 87 gpm at 110°F, and with a half-life of 10.5 hours.

In the second case a cleanup within the poison out period is desirable. A 24-hour cleanup is normally adequate. Columns are used in parallel with a resulting flow of 174 gpm at 110°F for a cleanup half-life of 5.75 hours.

The ion exchange columns are individually shielded with 2 feet thick concrete walls with a 4 feet 6 inches thick concrete floor above the vessels. A removable slab (4 feet, 6 inches thick) of concrete is provided in the floor to allow access to one vessel without exposing the room to the source of all of the vessels. A shielding wall, 2 feet, 6 inches thick, is provided to protect the accessible area of the room from the equipment. Valves and strainers are located behind the shielding wall as well as the purification system cooler. Connections on the ion exchange columns for resin addition and removal are in an accessible area.

A monorail is provided in the room above the center of the ion exchange columns and filters. The monorail and hoist are sized to remove either a concrete shielding slab or an ion exchange column. When removing a vessel, the vessel is raised through a shielded flask to protect the plant operators from excessive amounts of radiation during the period of time it requires for complete removal.

Design Parameters

For system design parameters refer to Table 5.1.2-6.

TABLE 5.1.2-6
MODERATOR PURIFICATION SYSTEM
DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
General	
Purification Flow Rates:	
Normal	87 gpm
Maximum	436 gpm
Ion Exchange Inlet Temperature	110°F
Cleanup Half Lives:	
At Normal Flow	6 hours
At Maximum Flow	3 hours
Applicable Code	ASME Section III, Class 3
Heat Exchangers	
Name	Moderator Purification Heat Exchanger
Quantity	1
Type	Tube in Shell
Duty	Continuous
Capacity	100%
Heat Transferred	24×10^6 Btu/hr
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	436 gpm
Inlet Temperature	200°F
Outlet Temperature	110°F
Design Pressure	240 psig
Design Temperature	250°F

TABLE 5.1.2-6 (Cont.)

<u>Item</u>	<u>Value</u>
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	1275 gpm
Inlet Temperature	95°F
Outlet Temperature	133°F
Design Pressure	150 psig
Design Temperature	200°F
Construction:	
Tube Material	Incoloy
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 3
Shell Side	ASME Section III, Class 3
Filters and Strainers	
Name	Purification Filter
Quantity	1
Type	Removable Cartridges housed in Pressure Vessel
Filter Porosity	5-10 micron
Clean ΔP at Normal Flow	1 psig
Dirty ΔP at Normal Flow	30 psig
Design Conditions:	
Normal Flow Rate	125 gpm
Maximum Rated Flow	250 gpm
Cartridge Material	Pleated Paper
Operating Pressure (Gauge)	90 psig
Operating Temperature	110°F
Design Pressure (Gauge)	240 psig
Design Temperature	200°F

TABLE 5.1.2-6 (Cont.)

<u>Item</u>	<u>Value</u>
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Ion Exchangers	
Name	Moderator Ion Exchange Columns
Quantity	5
Type	Vertical Cylinder with Dished Top and Conical Bottom
Volume	12 ft ³
Resin Capacity	9 ft ³
Type of Screen	Wedge-Wire
Design Conditions:	
Maximum Flow Rate	125 gpm
Direction of Flow	Downward
Operating Pressure	90 psig
Operating Temperature	110°F
Design Pressure	240 psig
Design Temperature	200°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

5.1.2.7 Reactivity Control Systems

Shutdown System No. 1 (Shutoff Rods)

Function

Shutdown System No. 1 serves as the primary means of terminating reactor operation when certain parameters enter an unacceptable range. Fifty-six spring-assisted shutoff rods fall into the reactor by gravity inside perforated zirconium alloy guide tubes.

Description

The cadmium absorber element is sheathed in stainless steel and supported by a stainless steel cable which is wound on a drum inside the shutoff rod drive mechanism. The drive mechanism is mounted directly above the shutoff rod channel on the concrete shield deck. Refer to Figure 5.1.2-13 for details.

The cable sheave is driven by an electric motor through a gear train engaged by an electromagnetic clutch. When the clutch is de-energized, the sheave is released and the element falls.

The total negative reactivity introduced by the absorber rods is approximately 80 milli-k. The trip system employs two out of four coincidence logic on the parameters given in Table 5.1.2-7.

Each trip parameter is measured at four separate locations and each channel is provided with its own power supply. The logic is arranged so that loss of power to a channel results in a channel trip.

A block diagram of Shutdown System 1 is shown in Figure 5.1.2-14 and design data is given in Table 5.1.2-8.

Shutdown System No. 2 (Liquid Poison Injection)

Function

As a backup to Shutdown System No. 1, the Liquid Poison Injection System is provided to rapidly terminate reactor operation by injecting concentrated gadolinium nitrate solution into the bulk moderator through six horizontally distributed nozzles.

Description

This system is described in Section 5.1.2.4.

Liquid Zone Control System

Function

The Liquid Zone Control System varies the quantity of demineralized ordinary water in 28 separate compartments of the 12 vertical liquid zone control absorber assemblies for the purpose of spacial flux control. The ordinary water level in the 28 compartments can be varied either in unison for bulk reactivity control or independently for zonal reactivity control.

Description

Liquid Zone Control Assemblies

The 12 vertically oriented zone control assemblies, are divided into 28 zone control compartments. Four assemblies are divided into three compartments, while the other eight assemblies have two compartments. Each compartment has four feed tubes of which two are used for the inflow and outflow of water, and two are used for the inflow and outflow of the helium cover gas. Figure 5.1.2-15 shows a typical in-core assembly.

Process Flow

The liquid zone control system is designed so that a safe shutdown earthquake cannot result in a decrease in the quantity of water in the compartments. A flow diagram of this system is shown in Figure 5.1.2-16. In addition to the 12 zone control assemblies, the system consists of a subsystem for water and gas circulation. Except for the delay tank, all equipment of this subsystem is located in an accessible area of the reactor building.

A constant outflow of water from each compartment permits the quantity of water in the compartment to be controlled by varying the rate of inflow. The outflow from all compartments is delayed for decay of short lived neutron activation products before entering the accessible area of the reactor building.

Continuous water circulation is normally maintained with one of the three pumps provided. The water is pumped from the delay tank to the 28 compartments through individual control valves and also to the helium gas compressors. There are two gas compressors, one of which operates intermittently with the other on standby. The helium for the liquid zone control system is supplied from the moderator cover gas system bottle storage. Helium feed and bleed valves control the gas pressure in the 28 compartments so that the feed valves open as the pressure decreases, following a decrease in the water inventory and the bleed valve opens as the pressure increases when the water inventory increases. The bleed valves return the helium to the delay tank.

The catalytic recombination unit connected in series with the bleed valve reduces the concentration of radiolytic hydrogen in helium to acceptable limits. Flame arrestors are provided at the inlet and outlet of the recombination unit.

Control and Instrumentation

The water level measurement in each compartment is accomplished by measuring the differential between the pressure in the bubble tube extending to the bottom of each compartment, and the pressure in the gas balance header. The instrument output is connected to the station computers. A cathode ray tube display is used to show compartment levels. A computer output controls the compartment inlet water control valve. Manual control stations are also provided. The gas balance header pressure is controlled by feeding helium from the storage tank into the header or by bleeding excess helium from the gas balance header into the delay tank. The pressure difference between the delay tank and the gas balance header is used to control the feed and bleed valves.

Annunciation of system malfunctions or process excursions beyond normal operating limits is provided. The control system is designed to prevent a water level decrease in the compartments following a loss of pressure in the H₂O supply header (e.g. loss of power). The control computer monitors the H₂O supply header and the helium storage tank pressure. If either of these pressures falls below preset limits, an automatic reactor power setback is initiated. Design data for the system is give in Table 5.1.2-9.

Adjuster Units

Function

The adjuster rods are provided to shape the neutron flux for optimum reactor power and fuel burnup, and to provide excess reactivity for overriding xenon poison following power reduction. The reactor is designed to develop full power with all adjuster rods inserted.

Description

There are forty-two adjuster rods in the reactor. A typical adjuster unit is shown in Figure 5.1.2-17. Each adjuster element is a stainless steel tube varying in thickness lengthwise to provide optimum neutron absorption characteristics. The elements are guided within the calandria by perforated zirconium guide tubes. They are individually raised or lowered slowly by winching drives mounted on top of the deck, powered by electric motors on command from the control computers. Indication of vertical location is provided and vertical overtravel is limited. The casing for the entire drive unit is sealed, since an effective seal in the supporting cable is not practical. Design data for the adjuster unit is given in Table 5.1.2-10.

Solid Control Absorber Units

Function

The solid control absorber units adjust the flux level at times when greater reactivity rate or depth is required than that provided by the Liquid Zone Control System.

Description

There are eight solid control absorbers in the reactor. The design is essentially the same as that of the shutoff absorber rods (Figure 5.1.2-13). The absorber element is of the stainless steel clad-cadmium tube type. Design data is given in Table 5.1.2-11.

TABLE 5.1.2-7
SHUTDOWN SYSTEM NO. 1 TRIP PARAMETERS

<u>TRIP PARAMETER</u>	<u>DETECTOR TYPE</u>	<u>SETPOINT</u>	<u>INSTRUMENT* DELAY TIME SECONDS</u>	<u>CONDITIONING PARAMETERS</u>
High Neutron Power	Vertical In-core Detectors	115% Full Power	0.15	-
High H.T.** Pressure	Pressure Transmitters	14.76 MPa(g)	0.21	
Low Gross Coolant Flow	P Transmitters	(55% Nominal)	0.40	Low Power 1% Full Power
High Rate Neutron Power	East Ion Chambers	10%/Sec	0.25	
Access Control Key Interlock	-	-	-	-
Manual	-	-	-	-
Start-up+ Count Rate				High (Initial Operation Only)

* Time from trip value to trip signal

**H.T. - Heat Transport System

+ For Initial Start-up Only

TABLE 5.1.2-8
SHUTDOWN SYSTEM 1 (SHUT-OFF RODS) DESIGN DATA

<u>Item</u>	<u>Data</u>
Shut-Off Units	
Absorber Elements:	
General:	
Quantity	56 (four uniformly distributed banks each containing 14 units)
Type	Stainless Steel-Clad Cadmium Tube
Dimensions:	
Outside Diameter	4.45 in
Length, Cadmium Tubes:	
24 units	23 ft 8-3/4 in
4 units	20 ft 3-3/4 in
Weight:	
Longer units	100 lb
Shorter units	78 lb
Insertion Sequence	Simultaneous drop, spring augmented acceleration
Withdrawal Sequence	One bank (14 units) at a time
Withdrawal Time (Each Bank), Min.	130 s (Variable)
Travel	25 ft
Reactivity Worth, Approx.	-55 milli-k
Drive Mechanisms:	
Type	Motor-driven winch with cable and rotary hydraulic damper
Coupling, Winch to Motor	Gear train engaged by a dc Electro-Magnetic friction clutch
Power Requirement	230 Vac, 3 Phase, 60 Hz
Electrical Load	Class 3, 1/10 hp (75 W)

TABLE 5.1.2-8 (Cont.)

<u>Item</u>	<u>Data</u>
Guide Tubes (In-Vault Section):	
Material	Stainless Steel
Outside Diameter	5-1/4 in
Wall Thickness, Approx.	0.148 in
Guide Tubes (In-Core Section):	
Material	Zircaloy
Outside Diameter	5.022 in
Wall Thickness	0.061 in
Axial Tensioning Load Applied	1200 lb
Thimble:	
Material	Stainless Steel
Outside Diameter	6-1/8 in
Wall Thickness, Approx.	3/16 in

TABLE 5.1.2-9
LIQUID ZONE CONTROL SYSTEM DESIGN DATA

<u>Item</u>	<u>Data</u>
1. <u>Liquid Zone Control Absorber Assemblies</u>	
Quantity	12 (8 two zone, 4 three zone)
Reactivity Worth (Approx)	6 milli-k
Maximum Rate of Change	± 11 milli-k/s
Type	Vertical through-tubes, variable quantity water filling by helium pressure regulation
Material:	
Through-tubes	Zircaloy
Thimble Tubes	Stainless Steel
Length:	
Two Zone Units	48 ft
Three Zone Units	51 ft
Outside Diameter:	
Through-tubes	4.60 in
Thimble tubes	3/16 in
Compartments:	
Quantity	14
Inside Diameter	4.50 in
Capacity, Light Water	14 cu ft
H ₂ O Flow Rates	
Compartment Inflow	0 to 15 gpm
Compartment Outflow	7.5 gpm

TABLE 5.1.2-10
ADJUSTER UNITS DESIGN DATA

<u>Item</u>	<u>Data</u>
1. <u>Absorber Elements</u>	
Material	Stainless Steel
Length:	
Quantity 12	14 ft 4-1/2 in
Quantity 30	16 ft 3 in
Outside Diameter	3 in
Wall Thickness	1/32 in to 1/8 in
Weight	35 lb
Withdrawal + Insertion Sequence	One bank at a time
Withdrawal + Insertion Time, Each Bank	65 Seconds
Poison Override Time	30 Minutes
Total Reactivity Worth, Approx.	14 milli-k
Max Rate of Change (per bank)	.08 milli-k/s
2. <u>Drive Mechanisms</u>	
Type	Motor-driven winch with cable, direct coupling
Power Requirement	230 Vac, 3 Phase, 60 Hz
Electrical Load	Class 3, 1/10 HP (75 W)
3. <u>Guide Tubes (In-Core Section)</u>	
Material	Zircaloy 2
Outside Diameter	3.600 in
Wall Thickness	0.048 in

TABLE 5.1.2-10 (Cont.)

<u>Item</u>	<u>Data</u>
4. <u>Guide Tubes (In Vault Section)</u>	
Material	Stainless Steel
Outside Diameter	3.875 in
Wall Thickness	0.148 in
5. <u>Thimble</u>	
Material	Stainless Steel
Outside Diameter	4.750
Wall Thickness	.203 in

TABLE 5.1.2-11
SOLID CONTROL ABSORBER UNITS DESIGN DATA

<u>Item</u>	<u>Data</u>
1. <u>Absorber Elements</u>	
Quantity	8
Type	Stainless Steel-clad Cadmium Tube
Dimensions:	
Outside Diameter	4.45 in
Length, Cadmium Tube	19 ft 3-3/4 in
Weight	78 lb
Insertion Sequence + Method:	
Reactor Stepback	Simultaneous, free drop, gravity acceleration
Reactor Setback	Simultaneous, motor driven
Withdrawal Sequence + Method	Simultaneous, motor driven
Insertion and Withdrawal Time	Variable (130 s minimum)
Full Travel, Max	26 ft
Reactivity Worth, Approx.	10 milli-k
Max Rate of Change (Driving)	<u>±</u> .11 milli-k/s
2. <u>Drive Mechanisms</u>	
Type	Motor driven winch with cable and rotary hydraulic damper
Coupling, Winch to Motor	Gear Train engaged by a d-c electro-magnetic friction clutch
Power Requirement	230 Vac, 3 Phase, 60 Hz
Electrical Load	Class 3, 1/10 HP (75 W)

TABLE 5.1.2-11 (Cont.)

<u>Item</u>	<u>Data</u>
3. <u>Guide Tubes (In-Vault Section)</u>	
Material	Stainless Steel
Outside Diameter	5-1/4 in
Wall Thickness, Approx	0.148 in
4. <u>Guide Tubes (In-Core Section)</u>	
Material	Zircaloy
Outside Diameter	5.022 in
Wall Thickness, Approx	0.061 in
Axial Tensioning Load Applied	1200 lb
5. <u>Thimble</u>	
Material	Stainless Steel
Outside Diameter	6-1/8 in
Wall Thickness, Approx	3/16 in

FIGURE 5.1.2-1
REACTOR GENERAL ASSEMBLY (SECTION)

5.1-117

FIGURE 5.1.2-2
CONCRETE CALANDRIA VAULT

5.1-119

FIGURE 5.1.2-3

SCHEMATIC OF INSTALLED CALANDRIA TUBE

FIGURE 5.1.2-4

END SHIELD DIMENSIONS AND DETAILS

5.1-121

FIGURE 5.1.2-5

CALANDRIA VESSEL BOUNDARY

FIGURE 5.1.2-6
EMBEDMENT RING
5.1-122

MAIN MODERATOR SYSTEM

FIGURE 5.1.2-7

5.1-123

MODERATOR LIQUID POISON SYSTEM

FIGURE 5.1.2-8

SCHEMATIC OF SECOND SHUTDOWN SYSTEM

FIGURE 5.1.2-9

5.1-126

FIGURE 5.1.2-10
ANNULUS GAS SYSTEM

ANNULUS GAS SYSTEM

FIGURE 5.1.2-10

Sheet 2 of 2

5.1-128

FIGURE 5.1.2-11
MODERATOR PURIFICATION
SYSTEM

5.1-129

FIGURE 5.1.2-12
FLOW DIAGRAM
MODERATOR
PURIFICATION SYSTEM

FIGURE 5.1.2-13

SHUT-OFF AND SOLID CONTROL ABSORBER UNITS

FIGURE 5.1.2-14

SHUTDOWN SYSTEM NO. 1 - BLOCK DIAGRAM

FIGURE 5.1.2-15
LIQUID ZONE CONTROL ASSEMBLY

FIGURE 5.1.2-16
LIQUID ZONE CONTROL SYSTEM

FIGURE 5.1.2-17

ADJUSTER UNIT

5.1.3 Reactor Coolant and Connected Systems

The following sections comprise the Reactor Coolant and Connected Systems:

Reactor Coolant System (Section 5.1.3.1)

Integrity of Reactor Coolant Pressure Boundary (Section 5.1.3.2)

Reactor Coolant Pressure and Inventory Control System (Section 5.1.3.3)

Gland Seal System (Section 5.1.3.4)

Reactor Coolant Purification System (Section 5.1.3.5)

5.1.3.1 Reactor Coolant System

The Reactor Coolant System will circulate pressurized heavy water through the reactor fuel channels to remove heat produced by fission of the uranium fuel. The heat will be transported by the reactor coolant to the steam generators where it will be transferred to light water in the steam generator secondary to generate steam, which subsequently drives the turbine generator.

Design Bases

System Sizing

The Reactor Coolant System (RCS) will be sized for the normal operation of transferring 3820.5 Mwt (13.036×10^9 Btu/hr), 3783.4 Mwt from the reactor core and 37.1 Mwt from the reactor coolant pumps, to the steam generators. This heat is transferred to the secondary side of the steam generator forming 17.24×10^6 lb/hr (total four steam generators) of 1120 psia saturated steam with a 0.25 percent maximum moisture content. This steam is used in a turbine-generator set to produce approximately 1300 Mwe of electrical power.

System Classification

The Reactor Coolant System is a Safety Class 1 (except for the steam generator secondary side which is Safety Class 2) and Seismic Category I system. The System is ASME Code Class 1 (steam generator secondary side is ASME Code Class 2).

Physical Arrangement

All components of the Reactor Coolant System will be arranged inside the Containment building as shown in Figure 5.1.3-1. The four steam generators and the four reactor coolant pumps will be located on opposite sides of the calandria. Their location with respect to the reactor will be based upon:

- a. The optimum pipe lengths and bends to reduce pipe friction and shock losses to a minimum, thereby reducing pump work;
- b. The shortest pipe lengths to reduce coolant inventory;
- c. Arrangement of the feeders;
- d. Location and arrangement of the fueling machine;
- e. Foundation arrangements for components supports;
- f. The space necessary for initial construction and erection, for required shielding, and for maintenance.

The steam generators will be located at a higher elevation than the reactor. The elevation difference creates natural circulation which will remove core decay heat following coastdown of all reactor coolant pumps.

Description

The Reactor Coolant System (RCS) will consist of 740 reactor pressure tubes, four vertical steam generators, four shaft sealed motor driven reactor coolant pumps, four reactor inlet headers, four reactor outlet headers, and all necessary interconnecting piping and valving.

A schematic diagram of the RCS is shown in Figure 5.1.3-2. The system, which will be located entirely within Containment, will be arranged in two heat transport loops, each with two RCS pumps, two steam generators, two inlet headers, two outlet headers, one set of 370 inlet feeders, pressure channels and an equivalent number of outlet feeders.

The two loops will be connected to a common pressurizer and purification circuit, which will be isolated if an emergency condition should occur and thereby isolated from one another. Two loops will be provided in order to reduce the rate of positive coolant void reactivity insertion in the event of a loss of coolant accident. Refer to Figure 5.1.3-3 for a composite flow diagram.

Temperature and Flow

Feeder flows will be matched to individual fuel channel powers to give an equal D_2O steam quality of approximately 2.9 percent by weight for each channel at the reactor outlet headers when the reactor is at full power. Pressure drop causes the D_2O steam quality to increase from about 2.0 percent at the fuel outlet to 2.9 percent at the reactor outlet headers and to about 3.4 percent at the inlet to the steam generator.

The two figure eight loops provide bidirectional flow through the core such that the flow is in opposite directions in adjacent channels.

Each loop will remove the heat from half of the 740 channels in the reactor core. Each loop will have one inlet and one outlet header at each end of the reactor core. D_2O will be fed to each of the 740 fuel channels through individual feeder pipes from the horizontal reactor inlet headers and heavy water will be returned from each fuel channel through individual outlet feeder pipes to the horizontal reactor outlet headers. Individual feeder piping sizes depend on the coolant flow through the particular channel.

D_2O will enter each reactor outlet header at an average temperature of 639.3°F and a quality of approximately 2.9 percent steam by weight.

One 30 inch inner diameter pipe will connect each reactor outlet header to one steam generator. As the heavy water passes through the four steam generators, the heat will transfer to the secondary side and the D_2O outlet temperature is 570°F at full power. Refer to Figure 5.1.3-4. Each steam generator will connect to the pump suction of one heat transport pump by a 24 inch inner diameter pipe. Each heat transport pump will deliver heavy water to one reactor inlet header through one 24 inch inner diameter pipe.

Steam Generators

Four identical steam generators with integral economizer transfer heat from the D_2O reactor coolant on the primary side to the feedwater (H_2O) on the secondary side. The steam generators, as shown in Figure 5.1.3-5 will consist of an inverted vertical U-tube bundle installed in a shell. Steam separating equipment will be housed in the upper end of the shell.

The primary side will consist of the steam generator head or channel cover, the primary side of the tubesheet and the tube bundle. A baffle will separate the inlet half of the steam generator head from the outlet half. The U-tubes will be welded to the primary side of the tubesheet. The steam generator head will be carbon steel clad with stainless steel and will be provided with two manways.

The primary side of the steam generators will be constructed in accordance with the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code. The primary side will be Safety Class 1 and Seismic Category I.

The secondary side of the steam generators will consist of the shell, the steam separating equipment, the tube bundle shroud, the secondary side of the tube plate, the secondary side of the tube bundle, the economizer baffles and the tube support plates. Carbon steel will be the principle material of construction. The shell will be provided with a manway and connections for chemical addition and steam generator blowdown. Incoming feedwater will be pumped into the baffled economizer section and flow over the D₂O outlet portion of the U-tube bundle. The feedwater will emerge from the preheater section at nearly saturation temperature and mix with the recirculated saturated water flowing over the other portion of the U-tube bundle.

The steam-water mixture rising from the upper end of the U-tube bundle will pass through the steam separators. The separated water will recirculate to the tube bundle. The steam with less than 0.25 percent moisture by weight will leave the steam generator through the outlet nozzle. The water level at the separators will be controlled by a combination of level measurement, steam flow measurement and feedwater flow measurement.

Saturated water will flow to the tube bundle from the steam separating equipment through an annulus between the tube bundle shroud and the inside surface of the shell. The water will pass through holes in the bottom of the shroud and flow over the D₂O inlet portion of the U-tube bundle.

The secondary side of the steam generators will be constructed to the requirements for Class 2 components in Section III of the ASME Boiler and Pressure Vessel Code and will be Safety Class 2 and Seismic Category I.

Orientation of the steam generators with respect to the reactor and pumps is shown in Figure 5.1.3-1.

Reactor Coolant Pumps

The Reactor Coolant Pumps will be vertical single stage, single suction, single discharge centrifugal pumps. A Schematic of a reactor coolant pump is shown in Figure 5.1.3-6.

Each pump will be driven by a vertical, totally enclosed, air-water cooled squirrel cage induction motor. The pump will be supported by the motor mount which is bolted to the top of the pump casing. A removable shaft coupling will connect the motor to the pump. Removal of the coupling will allow sufficient space for the pump seals and bearings to be removed without removing the motor.

The motor will be supplied with a flywheel so that the total rotational energy of the pump motor unit will prolong pump coastdown and the rate of decay of flow, after loss of motor power thus ensuring that fuel design limits are not exceeded during the incident.

The motor will be supplied with two removable radial bearings and a double acting thrust bearing. The motor bearings will be lubricated by cooled and recirculated oil. An oil lift system will supply high pressure oil to both sides of the thrust bearing simultaneously during start-up.

The cover, or stuffing box, will be mounted on top of the casing and sealed by a double gasket. Within the stuffing box will be the heavy water lubricated bearing, restriction bushing, shaft seals, thermal barrier and the shaft. The main impeller will be located at the end of the shaft below the bearing.

Shielding will be installed between the pump casing and the pump motor to reduce irradiation of personnel engaged on pump and motor maintenance and other maintenance tasks above the pump casing.

The shaft sealing arrangement will consist of two similar mechanical seals plus a third mechanical low pressure vapor seal in series. Each mechanical seal will be designed to withstand the full system differential pressure. Both seals will be housed in a removable cartridge located in the upper part of the stuffing box. The back-up seal will be located at the top of the stuffing box. In the event that both mechanical seals fail, the back-up seal will prevent leakage of D_2O to atmosphere while the pump is stopped or is coasting to a stop.

The gland seal circuit will supply cooled and filtered D_2O for lubricating and cooling the mechanical seals. The D_2O will be injected below the lowest mechanical seal at a pressure above pump suction pressure and a temperature of $120^{\circ}F$. Part of the gland seal flow will pass into the pump casing through a radial restriction bushing while the remainder will flow up through the mechanical seals.

Headers

There will be four reactor outlet headers, two at each end of the reactor. Each outlet header will receive flow from 185 outlet feeders and will direct the flow to the inlet pipe of each steam generator.

There will be four inlet headers, two at each end of the reactor. Each reactor inlet header will receive flow from one pump discharge pipe and distribute the flow to 185 inlet feeders.

The headers will be of carbon steel clad on the interior with stainless steel. The outlet headers will be 30 inch ID and the inlet headers will be 24 inch ID.

Feeders

The 740 feeders at each end of the reactor will run from the fuel channels vertically up the face of the reactor and then horizontally across and above the fueling machine area to the headers.

Figure 5.1.3-4 shows the arrangement of the feeders. The feeders will range in size from 3-1/2" schedule 160 pipe to 1-1/2" schedule 160 pipe.

The inlet header to outlet header pressure drop is set by the maximum pressure drop in an inlet feeder - fuel channel - outlet feeder combination. Each of the remaining feeder-fuel channel combinations will be designed for the same inlet header to outlet header pressure drop by selecting combinations of pipe sizes for the inlet feeder and the outlet feeder to obtain the required flow resistance. Each flow will be set by the fuel channel power so that approximately 2.9 percent quality will be obtained at the outlet header.

RCS Piping

The RCS piping will consist of twelve major pipe sections, plus other smaller piping. The twelve major pipe sections are: four hot legs, four discharge legs and four suction legs. The pipe sections will be of carbon steel clad on the interior with stainless steel. The diameters of these pipe sections (30 inch ID for the hot legs and 24 inch for the suction and discharge legs) were selected to obtain a reasonable balance between erosion/corrosion, pressure drop and system volume.

Instrumentation and Control

Instrumentation connected to this system will be used to annunciate operating parameters, provide safety actuations and for the control of systems such as charging and letdown pressure control, emergency core cooling and shutdown cooling. Instrumentation will be provided to monitor and/or control the following:

- (1) Operation of pumps.
- (2) Selected channel flow for generating gross low flow trip signal.
- (3) Reactor inlet and outlet header levels during maintenance of major main circuit components.
- (4) Reactor outlet header pressures for Heat Transport System pressure control and protection.
- (5) Reactor inlet and outlet header pressures to generate an emergency core cooling signal.
- (6) RCS temperatures for protection and control.

Four independent measurement channels will be provided for each parameter which initiates protective system action. Two independent signals will be required to initiate protective action thereby preventing spurious actions resulting from the failure of one measurement channel. This arrangement will result in a high degree of protective channel reliability in terms of initiating action when required and avoiding unnecessary action from spurious signals.

Each pump will have instrumentation such that a pump start is conditional on sufficient oil pressure being present from the oil lift pump. Each pump will be equipped with a vibration switch attached to the pump casing. Excessive vibration will be alarmed by the plant computer.

There will be no automatic pump trips other than those associated with the normal motor protection logic, and no interlocking logic between pumps.

Gross low flow will be detected by monitoring flow through 12 inlet feeders, 3 on each side of circuit 1 and 3 on each of the inlet sides of circuit 2. Shutdown System No. 1 will be tripped when certain combinations of feeder elements detect low flow.

Reactor inlet and outlet header levels will be monitored during maintenance of the Reactor Coolant System components in order that adequate cover on the reactor inlet feeders is maintained.

Reactor outlet header pressure will be used to control the Reactor Coolant System pressure. Two sets of four pressure measurements will be monitored. One set will be used for narrow range pressure control during normal operation, and the second will be used for wide range pressure control during startup.

The pressure measurements on the reactor headers will be used to initiate Shutdown System No. 1, Emergency Core Cooling and also to open the power operated relief valves on high primary system pressure.

System temperatures will be provided on the hot headers and cold headers to be used for protection and control. The protection channels will be two sets of four channels for core protection calculations and limits.

Remote Shutdown Instrumentation and Controls

In accordance with GDC-19 instrumentation and controls needed to achieve and maintain hot shutdown and also having a potential capacity for subsequent cold shutdown through the use of suitable procedures will be provided in an auxiliary operating station. RCS instrumentation on this panel include: (1) hot and cold header temperatures and pressures and (2) pressurizer level and pressure.

Reactor Coolant System Parameters

The parameters for the RCS are found in Table 5.1.3-1.

TABLE 5.1.3-1
 REACTOR COOLANT SYSTEM DESIGN DATA

<u>Item</u>	<u>Data</u>
1. Reactor Coolant Pump	
Quantity	4
Type	Vertical, centrifugal, single suction, single discharge
Duty	Continuous
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	70,300 gpm
Temperature	570°F
Head	810 ft
Design Pressure	2500 psia
Design Temperature	700°F
Motor Data:	
Type	Vertical, squirrel cage induction motor
Power Supply	12.68 MW (17,000 HP)
Operating Load	9.275 MW (12,435 HP)
Construction:	
Materials in contact with fluid	Stainless Steel
Applicable Code	ASME Section III, Class 1 for RCPB
2. Steam Generators	
Quantity	4
Type	Vertical U-Tube with integral steam drums and economizer
Duty	Continuous
Capacity	25%
Heat transferred	3820.5 MW(th) (total for 4)

TABLE 5.1.3-1 (Cont.)

<u>Item</u>	<u>Data</u>
Tube Side Data:	
Fluid	D ₂ O
Flowrate	112 x 10 ⁶ lb/hr (total for 4)
Inlet Temperature	637.3°F
Inlet Quality	3.4%
Outlet Temperature	570°F
Design Pressure	2500 psia
Design Temperature	700°F
Shell Side Data:	
Fluid	H ₂ O
Steam Flow (Total for 4)	17.24 x 10 ⁶ lb/hr
Feedwater Flow (Total for 4)	17.2745 x 10 ⁶ lb/hr
Blowdown flow (continuous)	34,500 lb/hr (total for 4)
Blowdown enthalpy	561 BTU/lb.
Feedwater inlet temperature	450°F
Pressure at drum nozzle	1120 psia
Temperature at drum nozzle	558.5°F
Quality at drum nozzle	0.25%
Design pressure	1320 psia
Design Temperature	580°F
Construction:	
Tube Material	Ni-Cr-Fe
Tube Outside Diameter	0.75 inch
Tube Wall Thickness	.042 inch
Shell Material	Carbon steel
Applicable Codes:	
Tube Side	ASME Section III, Class 1
Shell Side	ASME Section III, Class 2

TABLE 5.1.3-1 (Cont.)

<u>Item</u>	<u>Data</u>
3. Reactor Headers	
Quantity:	
Inlet Headers	4
Outlet Headers	4
Material	Carbon steel clad with stainless steel
Dimensions:	
Inside Diameter:	
Inlet Headers	24 inches
Outlet Headers	30 inches
Design Pressure	2500 psia
Design Temperature	700°F
4. Feeders:	
Material:	
Pipe	Stainless Steel
Dimensions:	
Inside Diameter:	
Size 1	3.354 inches
Size 2	2.900 inches
Size 3	2.323 inches
Size 4	1.939 inches
Size 5	1.500 inches
Design Pressure	2500 psia
Design Temperature	700°F
5. Piping:	
General:	
Material	Carbon steel clad with stainless steel

TABLE 5.1.3-1 (Cont.)

<u>Item</u>	<u>Data</u>
Dimensions:	
Steam Generator, Inlet Inside Diameter	30 inches
Pump Suction:	
Inside Diameter	24 inches
Pump Discharge:	
Inside Diameter	24 inches
Pressure, Design:	2500 psia
Temperature, Design	700°F

5.1.3.2 Integrity of Reactor Coolant Pressure Boundary

Function

To prevent an uncontrolled release of radioactivity into the containment with subsequent leakage to the environment, the integrity of the Reactor Coolant Pressure Boundary (RCPB) will be assured throughout the design lifetime. Such assurance depends on the following:

- 1) design and analysis
- 2) appropriate design margins
- 3) selection of proper material to prevent brittle fracture and corrosion
- 4) quality control
- 5) proper operation

Description

The Reactor Coolant System components will be constructed to the requirement of Section III of the ASME Boiler and Pressure Vessel Code for Class 1 components and to Seismic Category I and Safety Class 1 criteria.

In addition, precautions will be taken in the design to ensure that inadvertent operation of equipment will not increase the severity of transients. The effects of corrosion and other chemical effects, erosion, irradiation, vibration, earthquake, fire, deposition and immersion will also be considered.

The effects of critical speeds, shaft stability, and other aspects of the mechanical performance of pumps are considered in the design for operating conditions. The first reactor coolant pump will undergo a 500 hour endurance test run with 100 starts and stops prior to installation. All other reactor coolant pumps must meet 100 hour type tests with 20 starts and stops.

Overpressure Protection

Overpressure protection for the steam generators and Reactor Coolant System will be in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. Overpressure protection will be ensured by primary relief valves, the secondary safety valves, and the Plant Protection System. Analysis of all reactor and steam plant transients causing pressure excursions will be conducted and the worst case transient will be selected as the design basis for the primary relief valves. The primary relief valves, secondary safety valves, and Plant Protection System will be designed to maintain the Reactor Coolant System below 110 percent of design pressure during the worst case transient. The analysis will consider the effects of reactor coolant pump head, flow pressure drops and elevation heads. Steam generator pressure will be limited to less than 110 percent of steam generator design pressure during the worst case transients. Thus, integrity of the reactor coolant pressure boundary and steam generator will be assured.

Overpressure protection for the pressurizer when isolated will be provided by the pressurizer safety valve. This valve will be sized to protect the pressurizer and surge line from the effects of the worst case pressure transient in the pressurizer with the surge line valves closed.

Reactor Coolant Supports

The RCS will be provided with a complete and integrated system of supports. These supports allow thermal movement for normal plant operation while restraining the components during a reactor coolant pipe rupture in addition to a safe shutdown earthquake.

Supports will be provided for the steam generators, RC pumps, pressurizer, headers and major piping. In addition, provisions will be made to support the feeder tubes and pressure tubes which pass through the reactor. The supports will be in accordance with Regulatory Guide 1.45, "Protection Against Pipe Whip Inside Containment".

Environmental and Missile Protection

In accordance with GDC 4, the RCS is designed such that protection will be provided from the effects of both missiles and containment environment following a LOCA or SLB.

Missile Protection

The RCS, a potential source of missiles, will be, to the extent practical, either surrounded by barriers or restrained to prevent missiles from reaching other parts of the RCS. This missile protection will be in accordance with Regulatory Guide 1.67.

Environment

The components within the containment, used to perform a safety related function will be designed to withstand the effects of the environment which could exist within the containment following a design basis LOCA or a steam line break.

Transients

The RCS will be designed to withstand the effects of the transients of Table 5.1.3-2. This is in accordance with GDC 2.

Inservice Inspection

The RCS will be arranged such that access is provided to permit inservice inspection in accordance with GDC 32, also the ASME B&PV Code, Section XI.

The welds of the RCS will be accessible for inspection using volumetric examination procedures. To accomplish this, the biological shielding will be designed to afford access for examination and thermal insulation will be removable.

TABLE 5.1.3-2
TRANSIENT CYCLES

<u>Description</u>	<u>Design Cycles</u>
A. Normal Operation (COD I)	
1. Operation at 100% power	Continuous
2. Heatup from Cold Shutdown to Hot Standby	500
3. Power Cycles Between Hot Standby and 100% Power	15,500
4. Cooldown from Hot Standby to Cold Shutdown	500
5. Power Maneuvering at 100% Power	10,000
6. Normal Temperature Pressure Changes at Power	10 ⁶
7. Frequency Control Power Transients at 100% Power	1,000
B. Upset Condition (COD II)	
1. Reactor Trip from 100% Power	500
2. Turbine Trip from 100% Power	500
3. RCS Pump Trip at 100% Power	100
4. Power Cutback from 100% Power	2,000
5. Power Cutback due to Reactor Power	100
6. Loss of Class IV Power	100
7. Loss of Feedwater to a S.G. at 100% Power	100
8. One Half Safe Shutdown Earthquake at 100% Power	200
C. Emergency Condition (COD III)	
1. Rapid Cooldown and Depressurization	15
2. Rapid Pressurization at 100% Power followed by Reactor Trip	15
D. Faulted Condition (COD IV)	
1. Steam Line Break	1
2. Loss of Coolant Accident	1
3. Feedline Break	1

TABLE 5.1.3-2 (Cont.)

<u>Description</u>	<u>Design Cycles</u>
E. Test Condition	
1. Hydrostatic Test	10
2. Turbine Roll Test	10
3. Leak Test	10

5.1.3.3 Reactor Coolant Pressure and Inventory Control System

Function

The function of the pressure and inventory control system is to provide a means of pressure and inventory control for the pressurized D₂O, closed loops of the RCS as well as provide adequate overpressure protection.

The basic design requirements of the system are as follows:

- a. To provide a means of controlling RCS pressure at the appropriate values over the full range of operating conditions.
- b. To limit pressure increases and pressure decreases to acceptable values.
- c. To accommodate RCS coolant swells and shrinks associated with warmup, cooldown, power changes and various unit disturbances.
- d. To provide means of RCS pressure recovery following power reductions.
- e. To provide a cooling flow of purified D₂O to the reactor coolant pump seals.
- f. To provide adequate relief capacity for overpressure protection of the RCS as well as a means to contain any relief from it.
- g. To provide a means of degassing the RCS coolant.
- h. To provide a variable test pressure signal for use in testing the various protective systems employing pressure as a process parameter.

Description

The Pressure and Inventory Control System (PICS) during normal RCS operation maintains RCS pressure and inventory by maintaining pressurizer pressure and level. Pressurizer pressure is maintained by heaters and spray. An increase in pressure will cause the spray valve to open admitting subcooled water to the pressurizer which decreases pressure by condensing steam. A decrease in pressure causes the heaters to come on to create steam, thereby increasing pressure. Pressurizer level is maintained by charging and letdown. Level is maintained by positioning of the letdown valve. An increase in level above the setpoint causes the letdown valve to open a larger amount which decreases the pressurizer level. Conversely, for a level decrease, the letdown valve is partially closed to increase the pressurizer level.

The pressure and inventory control system flowsheet is shown in Figure 5.1.3-7. Basically, the system will consist of a pressurizer, a degasser-condenser, and a charging and letdown circuit. The pressurizer will be designed to provide sufficient steam volume to accommodate the swell in the heat transport system from hot zero power to full power. This permits the reactor power to be increased or decreased at a rapid rate without imposing severe operating requirements on the charging and letdown circuit. The charging and letdown circuit is designed to handle the shrink and swell rates that will take place during normal heatup and cooldown.

When the reactor is at power, pressure is controlled by the pressurizer with inventory being adjusted by the charging and letdown circuit. When the reactor is shutdown and the pressurizer isolated, pressure can be controlled by the charging and letdown circuit.

A. Pressurizer

The common pressurizer which will control the pressure in the two heat transport circuits will be a cylindrical vessel installed vertically and will be connected to lines linking the outlet headers at one end of the reactor. Figure 5.1.3-8 shows the pressurizer arrangement.

The liquid D_2O stored in the pressurizer will be heated by electric immersion heaters if its temperature falls below saturation temperature. The heaters will be sized to heat the water in the pressurizer at a rate great enough such that the loop remains subcooled during plant heatup. This subcooling will be sufficient to provide the minimum RCS pump net positive suction head requirements. A small portion of the heaters will be operated continuously to offset pressurizer heat losses to ambient. The remainder of the heaters will be normally off, but will be turned on by a low pressure signal.

The pressurizer will also be provided with a spray to prevent an insurge or any other pressure increasing operational transient from initiating a high pressure scram. In addition, the spray is available for pressurizer cooldown.

The pressurizer level set point will be power level dependent. It will increase to accommodate the swell between zero power hot and full power. The D_2O level will be automatically controlled at the setpoint by the D_2O charging and letdown system under the control of the unit computers.

The pressurizer will be constructed to the requirements for Class 1 Components in Section III of the ASME Boiler and Pressure Vessel Code.

B. Charging and Letdown Circuit

The charging and letdown circuit will be provided primarily for inventory control of the RCS. Letdown flow will be taken from the purification system return line as a cold and purified fluid. The letdown line will run to the D_2O storage tank which stores the swell from zero power cold to zero power hot. The charging flow will be taken from the D_2O storage tank. Three high pressure positive displacement charging pumps will be provided. Two of which will normally operate with the other on standby. The charging flow will be supplied to the RCS via the interconnection at the purification system return line (see Figure 5.1.3-7). If the purification system is isolated for any reason, charging and letdown can still be maintained; charging via the normal path and letdown via the degas line to the degasser-condenser, which will be connected to the purification return line downstream of the isolating valves. The charging and letdown control valves will be duplicated for reliability.

C. Overpressure Protection

Power Operated Relief Valves (PORV's) will provide overpressure protection for each of the two RCS main circuits during normal reactor operation in conjunction with the reactor protective systems and secondary safety valves. The pressurizer safety valve provides pressurizer overpressure protection when isolated. The PORV's will be connected to the surge line upstream of the pressurizer isolation valves and the safety valve will connect to the pressurizer steam space. (Refer to Figure 5.1.3-7). The PORV's will discharge a mixture of steam and water into the degasser-condenser when opened. The safety valve will discharge steam to the degasser-condenser when opened.

D. Degasser - Condenser

A condenser will be provided to receive discharge from the pressurizer safety and PORV's and degassing flow directly from the RCS. The condensed D_2O will be returned via a cooler to the storage tank and/or charging pumps. The D_2O level will be controlled by level control valves located downstream of the cooler. Component cooling water will be used as a cooling agent. The flow of component cooling water in tubes will be regulated to control the pressure in the condenser to a value which is sufficient to transfer fluid at the maximum required flow rate into the storage tank. The heavy water from the degas line enters the degasser-condenser as a flashing spray to promote degassing. Non-condensable gases can be vented to the waste gas management system.

E. Storage Tank

The storage tank is designed to accommodate the swell from the RCS from cold condition to the zero power hot condition. A cover gas blanket is maintained at approximately atmospheric pressures. Means are provided to vent the cover gas to the waste gas management system (refer to Figure 5.1.3-3).

F. Instrumentation and Control

Under normal power producing operation, the RCS pressure will be controlled by the charging and letdown pressurizer spray and the heaters in the pressurizer. This pressure control system will operate on a narrow range of pressure transmitter signals. Triplicated sets of pressure measurements taken from the outlet headers will be used. The control system will select the outlet header that has the highest

pressure and controls this pressure to the desired value. If the pressure is above the setpoint, the spray valve opens to reduce pressure and if below the setpoint heaters in the pressurizer tank will come on to raise pressure. There are 52 bottom mounted immersion electric heaters with total of 2600 kw output in the pressurizer, 600 kw variable and the remainder on-off.

A wide range pressure control system will be used when the pressurizer is isolated. This is a single channel analog system which will use signals from pressure transmitters located on the outlet headers. The highest of the outlet header pressures is selected and is compared to the setpoint in a pressure controller. The controller output is fed to the two letdown valves and the charging pumps which control the reactor coolant system pressure. Wide range will not be used when the plant is at power since it does not have the reliability or accuracy to provide adequate control. When the pressurizer is isolated in a standby condition, its pressure is controlled by a single channel pressure control loop. The PORV's being on the header side of the isolation valves are always available for RCS overpressure protection. Overpressure protection of the pressurizer is provided at all times by the safety valve.

The D₂O level in the pressurizer will be controlled by the letdown valves, and charging pumps which will be governed by the unit computer. A median is selected from among different level measurements and then compared to the programmed level setpoint. For high reliability, the level control program can be run from either of the two unit computers. Two separate single channel analog systems will be used to control the D₂O level and the pressure in the degasser-condenser. The level control valve which is located downstream

of the degasser cooler will respond to signals from the level controller. The process water control valve will be governed by the pressure controller.

Control of the degasser cooler outlet temperature will be performed by a single channel analog system. The D_2O outlet temperature will be limited because of the RCS pump seal requirements. High temperature protection will be provided by an over-ride control which will close the degasser-condenser level control valve.

The degasser flow control will be performed by adjusting the opening of the degasser flow control valve from a band controller on the panel. If the degasser-condenser pressure rises above its set point, by an arbitrary margin, the degasser valve will be automatically closed down by an over-ride control to prevent the waste gas system from being overpressurized.

G. Surge Line

The surge line will consist of two lines; one to a reactor outlet header of each RCS loop and a common line which connects the two lines to the pressurizer. Each of the separate lines will contain two, redundant isolation valves. Each set of isolation valves will be designed to close if a LOCA should occur in the RCS loop to which they connect. Thus, the pressurizer and intact loop will be isolated from the break. In this way the blowdown energy release and void reactivity insertion to the reactor will be restricted.

Design Parameters

Refer to Table 5.1.3-3 for design parameters.

TABLE 5.1.3-3
 REACTOR COOLANT PRESSURE AND INVENTORY CONTROL SYSTEM
 DESIGN DATA

<u>Item</u>	<u>Value</u>
Pressurizer:	
Quantity	1
Type	Vertical cylinder with electrical immersion heaters
Internal Volume (minimum)	2600 ft ³
Design Conditions:	
Operating Pressure	2050 psia
Operating Temperature	639°F
Design Pressure	2500 psia
Design Temperature	700°F
Material	Carbon steel clad on the interior with stainless steel
Safety Class	1
Code	ASME Section III, Class 1
Seismic Category	I
Heater Capacity	2600 KW
Degasser-Condenser:	
Quantity	1
Type	Vertical cylinder with inverted U-tube cooler
Duty	Intermittent
Heat Transferred	28.78 MW(th)
Tube Side Data:	
Fluid	H ₂ O
Flow Rate	3240 gpm
Inlet Temperature	77°F
Outlet Temperature	144°F
Design Pressure	150 psig
Design Temperature	200°F

TABLE 5.1.3-3 (Cont.)

<u>Item</u>	<u>Value</u>
Shell Side Data:	
Fluid	D ₂ O (Flashing)
Flow Rate	3240 gpm
Design Pressure	2500 psia
Design Temperature	700°F
Construction:	
Tube Material	Incoloy 800
Shell Material	Stainless Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 2
Shell side	ASME Section III, Class 1
Pressure and Inventory Control Cooler:	
Quantity	1
Type	Horizontal U-tube in shell
Duty	Intermittent
Heat Transferred	6.3 MW(th)
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	1.5 x 10 ⁵ lb/hr
Inlet Temperature	293°F
Outlet Temperature	120°F
Design Pressure	2500 psig
Design Temperature	700°F
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	1125 gpm
Inlet Temperature	77°F
Outlet Temperature	118°F
Design Pressure	150 psig
Design Temperature	200°F

TABLE 5.1.3-3 (Cont.)

<u>Item</u>	<u>Value</u>
Construction:	
Tube Material	Incoloy 800
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 1
Shell Side	ASME Section III, Class 3
D ₂ O Charging and Letdown:	
Letdown:	
Design Flowrate	275 gpm
Normal Flowrate	72.4 gpm
Charging:	
Design Flowrate	132 gpm
Normal Flowrate	88 gpm
Temperature	120°F
D ₂ O Charging Pumps:	
Quantity	3
Type	Positive Displacement
Duty	Two continuous, one on standby
Pump Data:	
Fluid	D ₂ O
Flow Rate (per pump)	44 gpm
Temperature (normal)	120°F
Design Head	2750 psig
Design Pressure	3025 psig
Design Temperature	250°F
Motor Data:	
Type	Squirrel case, induction motor
Power Supply	186 KW

TABLE 5.1.3-3 (Cont.)

<u>Item</u>	<u>Value</u>
Construction:	
Materials in contact with fluid	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 2
D ₂ O Storage Tank:	
Quantity	1
Type	Horizontal Cylinder
Capacity	2800 ft ³
Design Conditions:	
Operating Pressure	Atmospheric
Operating Temperature	100°F
Design Pressure	15 psig
Design Temperature	250°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 2

5.1.3.4 Gland Seal System

Function

The function of the gland seal system is to supply cooled and filtered D_2O at high pressure to the Reactor Coolant pump glands.

Description

The gland seal system flow diagram is shown in Figure 5.1.3-9. An adequate amount of D_2O is pumped by the charging pumps through the gland filters and supplied to the four reactor coolant pump seals at about 120°F and at high pressure. Part of this seal injection flows into the pump casing while the balance flows into the pump seals to form the seal controlled bypass plus seal leakage. The controlled bypass passes through flow restrictions which are designed so that each seal has the same differential pressure. The seal leakage and controlled bypass is piped to the D_2O storage tank. Any leakage past the third low pressure vapor seal is piped to the D_2O storage tank. Constant injection temperature is assured by the heat exchanger. During startup the heat exchanger is used to heat the injection water to 120°F. If the purification system is isolated during normal operation, the heat exchanger is used to cool the flow from the RCS to 120°F. Two filters are connected in parallel, downstream of the heat exchanger. Flow, temperature, and pressures are monitored (and alarmed if necessary) at various points in the system.

Low flow, measured in the common supply line to the pumps, will automatically close the gland return flow isolating valve. This will bottle up the system and prevent outflow of hot water from the RCS that could damage the pump seals. System reliability is ensured by using two charging pumps for normal supply and back up is available from the Reactor Coolant Pump discharge. The plant computers are used for the alarms and logic in the gland seal system.

TABLE 5.1.3-4
GLAND SEAL SYSTEM DESIGN DATA

<u>Item</u>	<u>Value</u>
Gland Seal Heater and Cooler:	
Quantity	1
Type	Shell and tube
Duty	Continuous
Capacity	100%
Heat Transferred	
Tube Side Data:	
Fluid	D ₂ O
Flow Rate (design)	30 gpm
Operating Flow	15.6 gpm
Case	(1) (2)
Inlet Temperature	60°F 570°F
Outlet Temperature	120°F 120°F
Design Pressure	2735 psig
Design Temperature	650°F
Shell Side Data:	
Case	(1) (2)
Fluid	Steam Water
Design Flow Rate	1400 lbm/hr 400 gpm
Normal Flow Rate	560 lb/hr 205 gpm
Design Pressure	75 psig
Design Temperature	350°F
Construction:	
Tube Side Material	Austenitic Stainless Steel
Sheet Material	Carbon Steel
Applicable Codes and Safety Class	
Tube Side	ASME Section III, Class 2
Shell Side	ASME Section III, Class 3
Seismic Category	I

TABLE 5.1.3-4 (Cont.)

<u>Item</u>	<u>Value</u>
Filters:	
Quantity	2
Type	Permanent vessel, disposable cartridges
Design Conditions:	
Design Flow	30 gpm
Operating Flow	15.6 gpm
Design Pressure	2735 psig
Design Temperature	250°F
Construction:	
Material	Austenitic stainless steel
Applicable Code	ASME Section III, Class

5.1.3.5 Reactor Coolant System Purification System

5.1.3.5.1 Functions

The Reactor Coolant Purification System limits the activity and corrosion product buildup in the reactor coolant by removing soluble and insoluble impurities and in maintaining the pH of the D_2O at the required value.

5.1.3.5.2 Design Requirements

The maximum purification flow rate is 1800 gallons per minute at a temperature of 150°F. This flow rate gives a purification half life of about 20 minutes. The time to reduce the concentration of dissolved solids by 50% is defined as the purification half-life.

5.1.3.5.3 System Description

One purification system, common to both reactor coolant loops, is provided. Flow is taken from one reactor inlet header of each loop via the shutdown cooling piping, passed through an interchanger, cooler, filters and returned to one pump suction in each circuit. The interchanger - cooler combination is used to first cool the heavy water and then recover heat to minimize heat losses. When the flow is at its maximum value, the cooler outlet temperature is 150°F with an interchanger return temperature of 445°F. The inventory control system is connected to the purification system.

A control valve located downstream of the purification ion exchange columns is provided for purification flow control. The purification system can be isolated for maintenance purposes by means of isolation valves located in the inlet and outlet lines of the

system. In the event of a loss of coolant accident, the purification system will be isolated automatically.

Filters

Two disposable paper cartridge filters connected in parallel are provided with a rated flow of 900 gpm each. Normally both filters are used but the cartridges in a filter can be changed while still maintaining flow through the other.

Ion Exchange Columns

Two ion exchange columns are provided, each rated at 900 gpm. The systems are designed so that the resin can be transferred in and out of one ion exchange column which the other is maintained in operation. Each column contains about 70 ft³ of resin and has a total internal volume of about 90 ft³.

Interchanger/Cooler

A heat interchanger and cooler combination is used to cool down the reactor coolant from reactor inlet temperature to the purification temperature of 150 F, and to recover about 90% of the heat. The material used for tubing in the heat exchanger and the interchanger is Stainless Steel.

Instrumentation and Control

Instrumentation is required to monitor temperature, flows, and pressures in the system as well as the pressure drops across the filters and ion exchangers.

Temperature control on the outlet of the purification cooler is performed by a single channel analog system. The temperature

control valve responds to signals from the controller to modify the flow rate of high pressure recirculated water. To prevent damage to the resin in the ion exchange columns, a temperature over-ride is provided to close the purification control valve if the temperature downstream of the cooler exceeds the set point by a predetermined margin.

The four isolation valves of the purification system are normally open and closed only when the loss of coolant logic is actuated. If these valves are actuated then the pressurizer isolating valves are also closed and two reactor coolant circuits are isolated from each other. Although the ability to control the system pressure is lost, overpressure is still provided on the reactor coolant system and the pressurizer.

TABLE 5.1.3-5
 REACTOR COOLANT PURIFICATION SYSTEM DESIGN DATA

<u>Item</u>	<u>Value</u>
Reactor Coolant Purification System	
Purification Flow Rate	1762.4 gpm
Ion Exchange Inlet Temperature	150°F
Clean-up Half-life	Approximately 20 minutes
Applicable Code	ASME Section III, Class 1
Safety Class	1
Seismic Category	I
<u>Reactor Coolant Purification Interchanger</u>	
Quantity	1
Type	Shell and Tube
Duty	Continuous
Capacity	100%
Heat Transferred (normal)	297.12×10^6 Btu/hr
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	1800 gpm
Inlet Temperature	570°F
Outlet Temperature	210°F
Design Pressure	2500 psia
Design Temperature	650°F
Shell Side Data:	
Fluid	D ₂ O
Flow Rate	1800 gpm
Inlet Temperature	150°F
Outlet Temperature	455°F
Design Pressure	2500 psia
Design Temperature	650°F

TABLE 5.1.3-5 (Cont.)

<u>Item</u>	<u>Value</u>
Construction:	
Tube Material	
Tube Outside Diameter	5/8 inch
Shell Material	Austenitic Stainless Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 1; Safety Class 1; Seismic Category I
Shell Side	ASME Section III, Class 1; Safety Class 1; Seismic Category I

Reactor Coolant Purification Cooler

Quantity	1
Type	Tube-in-shell
Duty	Continuous
Capacity	100%
Heat Transferred	60×10^6 Btu/hr
Overall Length	
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	1800 gpm
Inlet Temperature	210°F
Outlet Temperature	150°F
Design Pressure	2500 psia
Design Temperature	650°F

TABLE 5.1.3-5 (Cont.)

<u>Item</u>	<u>Value</u>
Shell Side Data:	
Fluid	H ₂ O
Flow Rate	2645 gpm
Inlet Temperature	95°F
Outlet Temperature	140°F
Design Pressure	150 psig
Design Temperature	450°F
Construction:	
Tube Material	Austenitic Stainless Steel
Tube Outside Diameter	5/8 inch
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section III, Class 1, Safety Class 1, Seismic Category I
Shell Side	ASME Section III, Class 2, Safety Class 2, Seismic Category I
Reactor Coolant Purification Filters:	
Quantity	2
Type	Removable cartridges housed in pressure vessel
Filter porosity	5 micron, nominal
Crud holding capacity	each approx. 60 lb
Clean ΔP at normal flow	1 psig
Dirty ΔP at normal flow	30 psig
Design conditions:	
Normal flow rate each	900 gpm
Maximum flow rate each	1800 gpm
Direction of flow in cartridges	Radially inward
No. of cartridges per filter	--

TABLE 5.1.3-5 (Cont.)

<u>Item</u>	<u>Value</u>
Cartridge material	Pleated paper
Operating Temperature	150°F
Design Pressure	2500 psia
Design Temperature filter vessel	650°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 1, Safety Class 1, Seismic Category I
<u>Reactor Coolant Purification Ion Exchange Columns</u>	
Quantity	2
Type	Vertical cylinder, dished top, conical bottom
Volume	90
Resin Capacity	70
Type of Screen	Wedge-wire, 0.008 in slot
Design Conditions:	
Maximum flow rate, each	900 gpm
Direction of flow	Downward
Operating pressure	1800 psig
Operating temperature	150°F
Design Pressure	2500 psia
Design Temperature	650°F
Construction:	
Material	Carbon steel with internal clad of austenitic stainless steel
Applicable Code	ASME Section III, Class 1, Safety Class 1, Seismic Category I

REACTOR COOLANT SYSTEM ARRANGEMENT

FIGURE 5.1.3-1

5.1-177

PLAN VIEW

SIDE VIEW

5.1-178

RCS SCHEMATIC
FIGURE 5.1.3-2

COMPOSITE FLOW SHEET

FIGURE 5.1.3-3

5.1-179

FIGURE 5.1.3-4

RCS TYPICAL FEEDER ARRANGEMENT

NO.	SERVICE	No. REQ'D
1	PRIMARY INLET	1
2	PRIMARY OUTLET	1
3	DOWNCOMER FEEDWATER	1
4	STEAM OUTLET	1
5	BOTTOM BLOWDOWN	1
6	LIQUID LEVEL	8
7	PRIMARY MANWAY	2
8	SECONDARY MANWAY	2
9	HANDHOLE	2
10	UPPER ECONOMIZER FEEDWATER	1
11	LOWER ECONOMIZER FEEDWATER	1

FIGURE 5.1.3-5
TYPICAL STEAM GENERATOR

FIGURE 5.1.3-6
REACTOR COOLANT PUMP (TYPICAL)

FIGURE 5.1.3-7
P.H.T.S. PRESSURE AND INVENTORY CONTROL SYSTEM

NOZZLE SCHEDULE		
NO.	SERVICE	NO. REQ'D
1	MANWAY	1
2	SURGE	1
3	SPRAY	1
4	SAFETY VALVE	4
5	INSTRUMENT	4
6	INSTRUMENT	2
7	TEMPERATURE	1
8	HEATER	52

FIGURE 5.1.3-8
PRESSURIZER ARRANGEMENT (TYPICAL)

FIGURE 5.1.3-9
GLAND SEAL SYSTEM

FIGURE 5.1.3-10
REACTOR COOLANT
PURIFICATION SYSTEM

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5.1.4 Engineered Safety Features Systems

The following systems comprise the Engineered Safety Systems:

- Containment Spray System (Section 5.1.4.1)
- Containment Isolation System (Section 5.1.4.2)
- Combustible Gas Control System (Section 5.1.4.3)
- Emergency Core Cooling System (Section 5.1.4.4)
- Shutdown Cooling System (Section 5.1.4.5)
- Control Room Habitability System (Section 5.1.4.6)

These systems are provided to mitigate the consequences of unlikely postulated accidents. Each of the above systems are described in the following sections.

5.1.4.1 Containment Spray System

Functions

The Containment Spray System is designed to preserve the integrity of the Containment boundary by removing thermal energy from the Containment in the event of a Loss-Of-Coolant Accident (LOCA) or Steam-Line Break (SLB). In addition the Spray System is designed to remove iodine from the containment following a LOCA.

Design Requirements

The containment spray system is designed to remove the energy released to the containment in the event of a loss-of-coolant accident or steam-line break to prevent the containment pressure from exceeding the design pressure. It also is required to assure that the containment leakage rate is reduced to less than one-half of the design leakage rate within twenty-four hours.

The spray system also provides for iodine removal to limit the consequences of a LOCA within the prescribed limits of 10 CFR 100.

Except for the spray headers and inside isolation valves, the containment spray system is located in the basement of the reactor service and fuel handling building.

The system is constructed to seismic Category I, ASME B&PV Code, Section III, Class 2 requirements.

System Description

The containment spray system consists of two redundant trains or subsystems, each of 100-percent capacity. Each subsystem consists of a spray pump, SCHX (Section 5.1.4.5) a 360-degree spray header, and the associated piping and valves. Each train draws independently from the demineralized water storage tank (Section 5.1.4.5).

The heat exchangers, and portions of the piping and valves are shared with the Shutdown Cooling System (Section 5.1.4.5).

On initiation of an ESFAS signal, as reported by high containment pressure from two out of three signals, the valves are properly aligned after which the pump motor will be started. Containment spray from the demineralized water storage tank will continue until a low-level set-point is reached. When the low water level is reached, the valves will be realigned and the water will be withdrawn from the containment sump.

The spray system functions to provide a rapid reduction in containment elemental iodine concentration following a LOCA. This greatly reduces the amount of iodine present in the containment atmosphere that could leak to the environs. Iodine removal by the spray

system reduces the offsite radiological consequences resulting from a highly improbable design basis event below allowable limits.

Sodium hydroxide is added to the spray water from a tank containing a solution of NaOH. This chemical will facilitate removal of radioactive iodine from the containment atmosphere. Additions of NaOH solution occurs when the containment spray is activated. Parallel and redundant motor-operated isolation valves in the transfer line between the NaOH storage tank and the spray eductors are opened by an ESFAS signal. The NaOH solution mixes with 10 percent of the containment spray flow in an eductor located in a side stream from the pump discharge and the mixture is injected into the pump suction. The spray eductor mixes the solution and meters for proper pH control.

The containment spray system consists of two independent subsystems, each containing:

1) Spray Pump

The pump has a capacity of 4000 GPM, and is furnished with a 450 HP electric motor drive. The pumps are designed to ASME III, Class 2, Safety Class 2 and Seismic Category I criteria.

2) Shutdown Cooling Heat Exchanger

This is normally used for shutdown cooling. The heat exchanger is a Pure counterflow type with sump water on the tube side during recirculation operation.* During the injection phase the exchangers do not require water. After the containment pressure has been reduced to atmospheric pressure, the valves are realigned, including shutting off the flow of NaOH, and the system is activated for long-term cooling of

*During normal plant operation the SCHX contains heavy water. The heavy water is isolated from the light water of the DWT.

the containment. Detail parameters are found in Table 5.1.4-2.

3) Sodium Hydroxide Storage Tank

This is a carbon-steel horizontal tank with ASME F&D heads, supported on two saddles, measuring 9'-6" OD by 20'-6" straight sides. The tank is located in the Reactor Service and Fuel Handling Building. The tank is sized to contain approximately 9000 gallons of 20% sodium hydroxide solution.

The tank is pressurized with a nominal 2 psig inert nitrogen atmosphere to prevent degradation of the sodium hydroxide during long term storage. Two combination vacuum breakers and relief valves are provided on the tank to prevent over-pressurization or creation of a vacuum during NaOH drawdown. A local sample connection is provided to permit sampling the solution on a periodic basis.

4) Spray Headers and Nozzles

Two redundant sets of spray headers are located as high as possible in the containment. Each set of spray headers have 160 nozzles: 96 in the lower header and 64 in the upper pipe header. The nozzles are oriented to provide maximum coverage of the containment atmosphere. The nozzles are one-piece castings with large unobstructed passageways to eliminate the possibility of clogging. The nozzle design produces uniform droplet size distribution.

5) Spray Additive Eductors

The spray additive eductors draw 20% sodium hydroxide into their suction by using water discharged by the pumps.

Approximately 10 percent of the flow is diverted through the eductor where it is redirected back into the pump suction after mixing with the NaOH solution.

All equipment is designed such that 100 percent heat removal capacity is provided by a single subsystem with its separate and redundant components.

The entire wetted surfaces of the containment spray system is constructed from austenitic stainless steel, where applicable. Other materials, such as brass for the spray nozzles, may be used as required.

5.1.4.2 Containment Isolation Systems

Functions

The Containment Isolation Systems establish and/or maintain isolation of the containment from the outside environment to prevent release of radioactive material, and ensure that the public is protected by not exceeding the site boundary dose limitations specified by appropriate federal regulations (10 CFR 100) under adverse conditions.

Design Requirements

The design of each piping system penetrating the containment, must consider the most adverse process requirements and containment isolation requirements.

The isolation valves of each system are required to be properly positioned, i.e. either opened or closed, automatically upon receiving an engineered safeguards features actuation signal (containment pressure and activity) (ESFAS) caused by either a LOCA or main steam line break.

In general, double barrier protection is provided for all lines that penetrate containment. A barrier may consist of a valve, a closed system, or a diaphragm depending on its location and application.

System Description

Containment isolation valves are located in process lines to the requirements of Appendix A of 10 CFR 50, Criteria 55, 56, and 57. Criteria 51, 52, and 53 requirements are incorporated into the containment isolation system materials and testing used in the components.

Containment isolation valves are provided with actuation and control components appropriate to the type of valve. Globe and diaphragm valves are normally furnished with air diaphragm operators designed to fail to the safe position on loss of control air pressure. Gate valves are fitted with motor operated components and are powered from emergency electrical buses. Motor operated valves fail in the "as-is" position. No manual operation is required for immediate containment isolation.

Containment isolation valves, actuators, and controls are protected against missiles, including those caused during a LOCA. Main steam isolation valves, pressure and safety relief valves, and boiler feedwater isolation valves are located in a seismic Category I Penetration Building.

Only isolation valves located inside containment are subject to high-pressure, high-temperature steam atmosphere resulting from an accident. Operability of these valves are assured by design and qualification tests.

5.1.4.3 Combustible Gas Control System

Functions

The Combustible Gas Control System is provided to prevent the accumulation of hydrogen from reaching combustible concentrations in the containment following a loss-of-coolant-accident. Hydrogen may accumulate in the containment as a result of:

- ° Metal-water reaction between zirconium and reactor coolant;
- ° Radiolytic decomposition of the post LOCA emergency coolant and containment spray solutions;
- ° Corrosion of metals by solutions used for emergency cooling or containment spray.

Design Requirements

The system is designed to maintain the hydrogen concentration in the containment to below 4.0 volume percent (4.0 v/o), which is the lower flammability limit in air.

System Description

The primary means of hydrogen control is provided by a hydrogen recombiner. A dilution and purge system is also provided as a backup hydrogen control feature.

The recombiner is designed to start operating in seven days after a LOCA. Initiating recombiner operation at this time limits the maximum hydrogen concentration to 2.0 v/o. If recombiner operation fails, then periodic sampling of the containment atmosphere is initiated. Dilution and purging of the containment is performed to limit the hydrogen concentration to below 4.0 v/o. The 4.0 v/o limit is attained in approximately 54 days.

The recombiner is located in a seismic Category I structure adjacent to the containment. The recombiner takes the air from the containment, removes the hydrogen via a catalyst bed, and returns hydrogen-free air to the containment. The recombiner is skid mounted and is powered from the emergency electric buses.

The purge system consists of a flow meter, throttle valve, and associated piping. The piping downstream of the flow meter enters the emergency exhaust filters located within the containment and provides a direct path to the plant vent. Components and piping upstream of the filter are in accordance with seismic Category I requirements. All pressure components upstream of the filter are constructed to ASME B&PV Code, Section III, Class 2 requirements.

Mixing of the containment air is provided by the redundant fans in the containment recirculation filter system.

Containment isolation is provided by locked closed valves outside of the containment and either check valves or electric solenoid operated valves inside the containment. The solenoid valves are normally open and are energized to close to provide a fail-safe feature.

5.1.4.4 Emergency Core Cooling System

Functions

The Emergency Core Cooling System (ECCS) provides the following:

- 1) Injection of water into the Reactor Coolant System (RCS) to cover and cool the core following a loss of coolant accident (LOCA), thus preventing a significant amount of cladding failure along with a subsequent release of fission products into the containment.

- 2) Removal of heat from the core for extended period of time following a LOCA.
- 3) Injection of borated water into the RCS to increase the shutdown margin following a rapid cooldown of the system due to a steam line break.

Design Basis Criteria

Sizing

The ECCS is sized to prevent significant fuel cladding damage for the full spectrum of RCS boundary break sizes up to and including a double-ended rupture of the largest pipe or header in the RCS.

Redundancy

The ECCS is designed to withstand any single active failure during injection and any single active failure or limited leakage passive failure during recirculation. All failures are to be taken in conjunction with a loss of the normal offsite power supply.

Separation

The trains* of the ECCS are physically separated from each other so that a failure in one train will not result in failure of the other train due to fire, flooding, steam impingement or missiles.

Overpressure Protection

The ECCS is provided with protection from overpressurization due to the maximum pressure encountered in any mode of operation, higher RCS design pressures, thermal expansion of fluid in isolated

*Or subsystem consisting of one high pressure safety injection pump, one low pressure safety injection pump plus the independent suction and discharge paths for each. Each subsystem is full capacity.

sections of piping, leakage past check valves and inadvertent charging pumps discharge into the high pressure safety injection (HPSI) headers.

Interlocks

Interlocks are provided to ensure safety injection tank (SIT) availability, when the plant is at power and to insure SIT depressurization prior to the initiation of shutdown cooling.

Classification

The ECCS will be constructed to Safety Class 2, ASME B&PV Code Section III, Class 2 and Seismic Category I criteria.

Design Basis Assumptions

The RCS is arranged in two heat transport loops connected to a common pressurizer and purification circuit. When an emergency condition exists isolation valves will be closed in the common lines thereby isolating one loop from the other (see Section 5.1.3.1).

In the event of a LOCA, the pressure in the intact RCS loop will be maintained above the shutoff head of the HPSI pumps by the charging pumps.

In the event of a LOCA, half of the reactor core decay heat will be removed by the steam generators in the intact loop by blowing steam through the secondary side safety valves.

System Description

The ECCS configuration is shown in Figure 5.1.4-1. The ECCS will consist of two redundant full capacity trains, four SIT's and associated valves, piping, power source* and instrumentation. Each train will contain one HPSI pump, one LPSI pump and its own suction and discharge paths. Depending on valve alignment, the pumps will be able to take suction from either the demineralized water tank or the containment sump. Discharge will be to the breached RCS loop with adequate flow to the core assured regardless of break location. Adequate flow is insured by providing a connection to each header (four hot and four cold) from each of the two trains. Motor operated valves and pumps in each train will receive power from either the normal offsite power source or the emergency diesel generators assigned to the individual train. Table 5.1.3-1 contains the design parameters for the ECCS.

Component Description

HPSI Pumps

The configuration for the high pressure portion of the SIS is illustrated in 5.1.4-1. The HPSI pumps will keep the reactor pressure tubes full following a small break LOCA. The HPSI pumps will match decay heat by boiloff at the minimum allowable time for the initiation of recirculation (20 minutes), following a large break LOCA. System sizing is based on the conservative assumption that only one of the HPSI pumps will be available following an accident and that 50% of the flow will be lost through spillage. A 50% loss of flow through spillage is assumed because the RCS configuration is such that there is no physical barrier in the RCS to prevent flow from reaching a rupture from both of the reactor headers in the ruptured loop. The shutoff head of the HPSI pumps is selected so that for small breaks, the RCS pressure can be maintained above the setpoint of the safety valves on the secondary side of the steam generators (SG's), thus

*On-site emergency power (diesel generator) and offsite power bus.

enabling decay heat to be initially removed by blowing steam through these valves.

LPSI Pumps

The configuration for the low pressure portion of the SIS is illustrated in Figure 5.1.4-1. The LPSI pumps will be used to prevent significant fuel cladding failure during the injection mode of operation following a large break LOCA. It is conservatively assumed that only one of the LPSI pumps will be available following an accident, and that 50% of the flow will be lost through spillage. Because the LPSI pumps are required to have the capability of delivering flow to all of the reactor fuel tubes, each LPSI pump will require eight discharge lines to deliver flow to each inlet and each outlet header. (See Section 5.1.4.5 for requirement on shutdown cooling).

Safety Injection Tanks

The configuration for the SIT's is illustrated in Figure 5.1.4-1.

The four SIT's will provide a means of rapidly refilling the core following a large break LOCA, and of keeping it filled until flow from the safety injection pumps has become available. It will be conservatively assumed that the inventory of two SIT's will be lost through spillage. No credit will be taken for flow delivery from the SIT's during blowdown.

Each SIT will contain borated light water and will be normally pressurized with nitrogen gas to 610 psig.

The SIT's will be provided with connections for filling, draining, pressurizing, venting, relieving and sampling. In addition, pressure and level instrumentation with appropriate alarms will be provided to assure that technical specifications are complied with during normal power operation.

As illustrated in Figure 5.1.4-1, the discharge line from each SIT will branch into two lines at a SIT header. Each of these two discharge lines will connect to a header injection line. These two lines will be to corresponding headers of the two difference RCS loops.

Each of the two SIT header discharge lines will contain a motor operated gate valve and a check valve in series. Whenever the RCS pressure is above 700 psia, the isolation valves will be key locked open and power to the motor operator will be removed to prevent spurious movement. In addition, the valves will be equipped with redundant and diverse position indicators and an alarm which will alert the operator whenever the valve is closed.

During shutdown cooling, SIT pressure will be reduced to 200 psig and the SIT isolation valves will be closed.

Demineralized Water Tank

The demineralized water tank (DWT) will be used as the light water source for the ECCS as well as the Containment Spray System (Section 5.1.3.7). The contents of the tank will be adequate to maintain maximum ECCS and Containment Spray system flow rates for a period of at least 20 minutes until recirculation from the containment sump can be initiated.

The DWT is constructed to Safety Class 2, and Seismic Category I, and ASME B&PV Section III, Class 2 requirements. The tank is reinforced concrete with an austenitic stainless steel liner located underground outside the containment.

System Arrangement

The ECCS will be located outside of containment except for the injection lines to each RCS header. All components will be located in seismic Category I structures.

The ECCS which is a low pressure system will be isolated from the high pressure RCS during normal operation by at least two series valves.

Any portion of the ECCS containing D_2O will be isolated from the light water portions of the system by at least two normally closed series valves.

ECCS Operation

Normal Operation

During normal plant operation the ECCS will be aligned for the injection mode, but will not be operating.

Injection Mode of Operation

The SIS will automatically go into operation upon indication that a significant breach in the boundary of one of the RCS loops has occurred. For very small breaks, the charging pumps will keep the RCS pressurized. If pressurizer pressure drops below 1600 psia during plant operation, a safety injection actuation signal (SIAS) will be generated and the SIS will become operational. The intact RCS loop will be isolated from the breached loop and will then be cooled without a significant drop in pressure by blowing steam from the steam generator safety valves. The flow from the SIS pumps and the SIT's will be received by the breached RCS loops.

Recirculation Mode of Operation

Injection will continue until an DWT low level produces a recirculation actuation signal (RAS). This will cause pump suction to be transferred from the DWT to the containment sump.

TABLE 5.1.4-1
ECCS DESIGN PARAMETERS

HPSI Pump Parameters

Quantity	2
Design Pressure	2050 psig
Design Temperature	350°F
Fluid	Borated Light Water
Design Flow	815 gpm
Design Head	2850 ft
Shutoff Head (max.)	4530 ft
Shutoff Head (min.)	4100 ft
Maximum Flow*	1130 gpm
Head at Max. Flow	1580 ft

*Does not include 35 gpm/pump bypass flow

LPSI Pump Parameters

Parameters are given in Shutdown Cooling System Component Parameters, Section 5.1.7.1.

SIT Parameters

Quantity	4
Design Pressure	700 psig
Normal Operating Pressure	610 psig
Design Temperature	200°F
Operating Temperature	140°F
Internal Volume	2400 ft ³
Liquid Volume	1858 ft ³
Pressurizing Gas	Nitrogen
Fluid	Borated Light Water

TABLE 5.1.4-1 (Cont.)

Demineralized Water Tank Parameters

Quantity	1
Design Pressure/Temperature	Atmospheric/150°F
Operating Pressure/Temperature	Atmos/60°F
Internal Volume (Minimum)	400,000 gals
Fluid	Borated Light Water
Construction	Austenitic Stainless Steel Lined rein- forced concrete

5.1.4.5 Shutdown Cooling System

Functions

The Shutdown Cooling System (SCS) is designed to:

Supplement other heat rejection equipment as necessary to reduce the temperature of the Reactor Coolant System (RCS) in post shutdown periods from normal operating temperature to a plant maintenance temperature and to maintain this temperature for an extended period of time.

Have the capability of operating under post loss of coolant accident (LOCA) conditions in conjunction with other heat rejection equipment to cool the plant and bring it to a cold shutdown condition.

Provide shutdown cooling heat exchangers (SDCHX's) for heat removal from the containment following a LOCA during the recirculation mode when the low pressure safety injection (LPSI) pumps are operating.

Design Requirements

The Shutdown Cooling System (SCS) employed in operating Candu plants is designed for operation at full system pressure and is sized to rapidly reduce primary system temperature and pressure after a shutdown. The capability for rapid cooldown is desirable since the primary circuit can be accessed more rapidly for repairs, thus reducing down-time and improving reactor availability. The SCS incorporated into the conceptual HWR is designed for operation at reduced system pressure, consistent with present LWR design practice, but the system has been sized to retain the rapid cooldown feature.

The SCS of the conceptual design is sized to cool the RCS from 350°F and 450 psia at 30 minutes after shutdown to 145°F and atmospheric pressure at 4-1/2 hours after shutdown. Cooling during the first 30 minutes and to 350°F is via the steam generators; the use of steam generators for initial cooling after a shutdown is standard operating procedure for both PWRs and HWRs. For comparison, the SCS of PWRs are typically sized to initiate cooling 3-1/2 hours after shutdown and to cool the RCS to 145°F and atmospheric pressure over a period of 24 hours.

The SCS is also sized to limit the temperature rise across the core (resulting from decay and LPSI pump heat) to 75°F at 30 minutes after shutdown and to maintain the RC temperature below 180°F with the RCS at atmospheric pressure at 4-1/2 hours after shutdown, when only one of the two SCS trains is available.

The SCS is designed such that no single failure of an active component during residual heat removal will result in a loss of core cooling capability, or prevent the initiation of shutdown cooling from the control room. All single failures are to be taken in conjunction with a loss of the normal offsite power supply.

Each train or subsystem of the SCS is physically separated from each other so that a failure in one train will not result in failure of the other train, due to fire, flooding, steam impingement or missiles.

Overpressure protection is provided for the SCS components having a design pressure and temperature that is less than RCS design pressure and temperature with overpressure protection devices and three motor operated suction-isolation valves in series. No single active failure will allow overpressurization of the SCS. Suitable provisions will be made to collect the discharges from overpressure protection devices.

The SCS will be constructed to Safety Class 2, ASME B&PV Code Section III, Class 2 and Seismic Category I System criteria.

Design Basis Assumptions

The SCS will not interface with the Containment Spray System except via double isolation valves.

Shutdown cooling will be initiated with the RCS at 350°F and 400 psia at 30 minutes after shutdown.

The total fuel channel heat generation at full power operation plus margin will be 4000 MW (th).

Reactor core decay heat is based on decay heat curves generated using the ORIGEN program for a two year average fuel residency time.

The maximum allowable cooldown rate will be 300°F/hr.

During shutdown cooling, all reactor fuel channels will require cooling flow.

No credit will be taken for removal of core decay heat by moderator cooling.

System Description

The SCS configuration is illustrated in Figure 5.1.4-2. During shutdown cooling, a portion of the reactor coolant will flow out the shutdown cooling nozzles located on the reactor outlet headers. Coolant will be circulated through the SDCHX's by the LPSI pumps and returned to the RCS through the eight LPSI lines which discharge to the reactor inlet headers. Normally the system will be full of D₂O and will be isolated from the RCS. The system pumps and heat exchangers will be located in the basement of the reactor service and fuel handling buildings.

Component Descriptions

LPSI Pumps

During shutdown cooling, each LPSI pump will take suction from one RCS outlet header from each RCS loop and will discharge through a SDCHX. The shutdown cooling flow will then be returned to each of the RCS loops by each SCS train through the low pressure safety injection headers to the RCS inlet headers. Because shutdown cooling flow will be required to cool all of the reactor fuel tubes each train will require four discharge lines to deliver flow to each reactor inlet header. (See Section 5.1.4.4 for requirement for low pressure safety injection). The LPSI pumps will be sized to provide adequate flow to the SDCHX's to cool the RCS to 145°F at 4-1/2 hours after shutdown and to preclude the temperature rise across the core from exceeding 75°F.

Shutdown Cooling Heat Exchanger

The SDCHX's will remove core decay heat, RCS sensible heat and LPSI pump heat during cooldown and subsequently maintain the RCS at 145°F. The SDCHX's will be sized as a pair to be able to cool the RCS to 145°F at 4-1/2 hours after shutdown.

Operation

During normal plant operation, the SCS will not be operated and will be isolated from the RCS. Its alignment will be for the injection mode of SIS operation.

Plant startup will be a series of manually controlled operations which will bring the RCS from the cold shutdown condition to hot standby. The SCS will be used during cold shutdown conditions to control the RC temperature. During the early stages of plant startup, core decay heat and LPSI pump heat will be able to be removed by the SCS. After one or more RC pumps have been started

and before the RCS state has reached 350°F, and 400 psia, the LPSI pumps will be stopped and will be re-aligned for automatic safety injection.

After the RCS state has been reduced to 350°F and 400 psia, at a minimum of thirty minutes after shutdown, the SCS will have the capability of continuing the cooldown until the RCS has been reduced to 145°F and atmospheric pressure. A maximum cool-down rate of 300°F/hr will be maintained during the initial phase of shutdown cooling by the operator limiting the amount of LPSI pump flow through the SDCHX's and channelling the remaining flow through a SDCHX bypass line. Flow will be controlled by means of a flow control valve in each of the SDCHX and bypass lines. After full flow through the SDCHX's has been achieved, further operator control of the LPSI pump flow will not be required. The RCS will be able to be drained by use of the SCS with flow taken from downstream of the LPSI pumps and sent via the purification system to the D₂O storage tank. Feed and bleed will be used to control the D₂O level in the RCS and SCS after the RCS level has been reduced. Shutdown cooling will be continued, as required, during plant shutdown to maintain a water temperature of 145°F or less.

TABLE 5.1.4-2
SCS DESIGN PARAMETERS

LPSI Pumps

Quantity	2
Design Pressure	650 psig
Design Temperature	400°F
Design Flow	9000 gpm
Design Head	300 ft
Material in Contact with D ₂ O	Stainless Steel

SDCHX

Quantity	2
Product of Effective Area and Overall Coefficient of Heat Transfer - UA	$3.77 \times 10^6 \frac{\text{BTU}}{\text{hr}^\circ\text{F}}$
Design Total Heat Load	
@ 0.5 hrs	675.4×10^6 BTU/hr (5% full power heat load)
@ 4.5 hrs	161.2×10^6 BTU/hr (1.2% full power heat load)

Tube Side

Fluid	D ₂ O Reactor Coolant
Design Pressure	650 psia
Design Temperature	400°F
Flow	9000 gpm
Design Inlet Temperature	145°F
Material	Austenitic Stainless Steel

TABLE 5.1.4-2 (Cont.)

Shell Side	
Fluid	Component Cooling Water
Design Pressure	150 psig
Design Temperature	250°F
Flow	11000 gpm
Design Inlet Temperature	105°F
Material	Carbon Steel
Codes	
Tube Side	
Industrial	ASME B&PV, Sect. III Class 2
Safety Class	2
Seismic Category	I
Shell Side	
Industrial	ASME B&PV, Sect. III, Class 3
Safety Class	3
Seismic Category	I
<u>SCS Volume (estimated)</u>	
Fluid volume per SCS train	1000 ft ³
Total SCS Fluid Volume	2000 ft ³

5.1.4.6 Control Room Habitability System

Functions

The Control Room Habitability System is provided to assure accessibility and habitability of the main control room and secondary control room subsequent to the occurrence of postulated accidents resulting in release of fission products or hazardous chemical releases such as chlorine.

System Description

The control room complex has a recirculating air system consisting of two air handling units, two independent refrigeration machines, and two independent air cooled condensers. Each air handling unit and its associated refrigeration equipment is sized to meet the normal and the design emergency conditions.

During normal operations, at least 30 cubic feet per minute of filtered outside air is provided for each occupant, and the air is then exhausted from the system. The control room complex is maintained at a positive pressure with respect to the outside environs by static pressure controls which limit the amount of air discharged from the system and building. Recirculated air and outside air is filtered prior to cooling and flowing into the control room complex.

The control room is supplied with conditioned air through a sheet metal duct system that is supported with seismic Category I supports. Air is distributed through diffusers as necessary to maintain design room temperature. Return air is drawn through return air registers in the plenum above the ceiling. The return air, together with the makeup ventilation air, flows through the air conditioning unit for recirculation.

The control room air cooling system consists of two 100 percent redundant trains or subsystems. One train is supplied electric power from the train A emergency bus and the other from train B emergency bus.

The design basis of the cooling system for normal operation and for emergency operation is to maintain the control room at 75°F when the outside air is 88°F or less and supplying the control room with 1100 cfm of outside air.

The control room air cooling system is constructed to seismic Category I requirements and to ASME B&PV Code, Section III, Class 3 for pressure boundary components of the systems.

The cable spreading rooms below and above the control room are ventilated with filtered outside air during the summer months. The ventilation system consists of a supply fan, return fan, and sheet metal ducts.

The emergency switchgear areas are ventilated in the summer with filtered outside air. Each ventilating system has redundant supply and return air fans, and sheet metal duct. The battery rooms have redundant exhaust fans, supply and exhaust sheet metal ducts. Battery rooms are maintained at a negative pressure to prevent any hydrogen generated by the batteries from infiltrating the emergency switchgear areas.

The control room complex emergency cleanup filter system consists of a filter train with redundant fans. The filters consists of a roughing filter, HEPA filter, and an impregnated charcoal filter. Redundant, remotely located, seismic Category I air intakes are provided for emergency control room operations.

The control room is provided with toxic chemicals detection equipment. On detecting a toxic substance, the fresh air intakes will automatically close and the emergency cleanup system will be placed in operation.

The control room is provided with radiation detectors that will sound an alarm if the level increases to a set point.

FIGURE 5.1.4-1
SIS CONFIGURATION

FULL SIS CONFIGURATION INDICATED BY DARK LINES.

FIGURE 5.1.4-2
SCS CONFIGURATION

FULL SCS CONFIGURATION INDICATED BY DARK LINES.

5.1.5 Instrumentation and Control

The reactor plant instrumentation and controls provide monitoring protection for plant, personnel and equipment and also provide controls to enable the operator to start up, operate, and shut down the reactor plant.

5.1.5.1 NSSS Control Board

The NSSS control board is in the form of a "U" or "L" shaped duplex control board containing the necessary controllers, switches indicating, recording and annunciator devices for remote operation of the generating unit. Reactor process systems and components which are monitored and controlled from this board are:

- a. High Pressure Safety Injection System
- b. Low Pressure Safety Injection System
- c. Component Cooling Water System
- d. Reactor Coolant System
- e. Moderator Cooling System
- f. Nuclear Instrumentation: Reactor Trip Status Monitoring
Lights, Rod Position Indication Display, Rod Control, Reactor
Power Indication
- g. Engineered Safety Features Actuation

5.1.5.2 Remote Shutdown Panel

A remote shutdown panel is provided for the orderly shutdown of the reactor to a cold condition from a location remote from the main control room. This panel makes shutdown possible when the main control room becomes inaccessible. The capability is required by 10 CFR 50, Appendix A, General Design Criterion 19.

Located on this panel are the necessary controls and instrumentation channels that are associated with the major systems in both the primary and secondary sides of the plant. These control and instrumentation devices prevent the reactor from achieving criticality in violation of the Technical Specification.

General considerations in the design of this panel are:

- a. Both the turbine and the reactor are tripped, locally or at the main control room
- b. All automatic systems continue functioning
- c. A selector switch is provided to transfer the shutdown controls to the local panel
- d. The necessary indications are: water level and pressure for each steam generator, pressurizer water level and pressure, service water and auxiliary feedwater system status.

Major equipment and systems controlled from this panel are:

- a. Reactor coolant pumps
- b. Moderator cooling pumps
- c. Auxiliary feedwater pumps

- d. Boric acid and gadolinium nitrate transfer pumps
- e. Charging pumps
- f. Service water pumps
- g. Component cooling pumps
- h. Residual heat removal pumps
- i. Controlled steam release and feedwater supply system
- j. Reactor coolant inventory control system
- k. Pressurizer pressure control system

5.1.5.3 Heating, Ventilation and Air Conditioning Panels

These panels provide monitoring and control of the HVAC systems for buildings which house the reactor plant systems. Typical HVAC systems controlled from these panels are the auxiliary building air handling system, containment air handling system, containment purge systems, diesel generator air handling system and fuel storage area air handling system.

5.1.5.4 Radwaste Panels and Racks

The complete waste management system consists essentially of three systems: Liquid waste system, gaseous waste system and solid waste system. Overall control of the complete system is exercised from radwaste panels in a local control room located near the waste management facility in the waste process building. For the liquid waste system, the panels control the collection, processing and directing of the processed waste either for re-use

or discharge from the site. For the gaseous portion of the system, the panels monitor the removal of gaseous fission products from the reactor coolant letdown and the primary drain tank. The radwaste panels also carry indicating lights to show the status of various sump pumps and equipment. The radwaste system alarm signals are reported through a common annunciation in the main control room. Racks are provided for mounting local instruments. For the solid waste system, solidification, handling and drumming for shipment and storage are performed through local panels nearby the equipment.

5.1.5.5 Logic Panels and Cabinets

The panels and cabinets provide mounting space for analog devices such as function generators, bistable modules, summers, dividers, analog controllers, selectors, etc. Typical reactor process control loops, which have analog devices in these panels and cabinets are: control rod reference signals, steam dump control, feedwater pump speed control, pressurizer pressure and level controls, boric acid blend control, volume control tank level control, steam generator level control, etc.

5.1.5.6 Instrument Racks

The instrument racks take the form of an open rack. They are used to mount local instruments such as pressure transmitters, manifolds, pressure switches, and other pneumatic instruments that connect directly with the process pipes. The rack has a rigid structure, suitably braced, to withstand all stress incidental to shipping, installation and operation without warping or twisting. Arrangement of instruments, conduits on racks, and electrical devices are placed out of the paths of condensation, or water drains from testing or calibrating instruments. Sufficient clearance is provided for maintenance of the instrumentation

without interruption of service. There is a provision to collect the drains when the instrument is removed. Suitable engraved plastic nameplates are required for each instrument.

5.1.5.7 Computer System

The plant is provided with two plant computers, which have duplicate capabilities. In the event of a malfunction of one of the computers, reliability is provided by automatically switching to the other computer. These computers in conjunction with the nuclear instrumentation permit automatic control of reactor power and flux shape and to monitor local core behavior.

The system combines state-of-the-art computer hardware with color CRT to assemble and display plant performance data and plant status information. Station signals are continually monitored and alarm messages are produced with an audible warning when limits are exceeded.

Typical functions of the computer system are:

- a) Reactor power control
- b) Plant load control
- c) Steam pressure control
- d) Steam generator level control
- e) Primary heat transport pressure control

5.1.5.8 Radiological Data Management System

The radiological data management system is designed to assure compliance with the applicable NRC licensing requirements by providing information concerning the radiological environment of the plant. The system enables the health physicist and plant operating personnel to maintain complete awareness, in real-time, of plant radiation levels. Permanent records are automatically produced for regulatory requirements. The system consists of:

- a. Process Radiation Monitoring System
- b. Effluent Radiation Monitoring System
- c. Area Radiation Monitoring System
- d. Data Acquisition and Processing System

Each monitor channel is complete with detector, preamplifier, digital buffer, microprocessor with alarm outputs, and readout modules. The data acquisition and processing system is a computer-based system that collects the available information from field-mounted detectors, performs the necessary calculations and displays the results on the CRT as required. The system assists in the generation of required NRC reports.

5.1.5.9 Neutron Monitoring System

The neutron monitoring system consists of:

- a. Out-of-core flux detectors
- b. Fixed-in-core thermocouples

c. Fixed-in-core flux detectors

The out-of-core detectors protect the reactor core by monitoring the neutron flux and generating appropriate trips and alarms for various phases of reactor operating and shutdown conditions. They also provide a secondary control function by indicating reactor status during startup and power operation.

The in-core instrumentation provides information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. The thermocouples are positioned at preselected locations in the outlet nozzle assemblies to measure coolant outlet temperature for use in monitoring the core radial power sharing and coolant enthalpy distribution. The vertical in-core detectors are placed through the calandria such that they are not near any control rods or liquid zone control units to minimize interaction effects. The assemblies have single-ended platinum detectors wound on them to provide high neutron power alarms to the shutdown systems. Provisions are made for spares.

5.1.5.10 Post Accident Monitoring

The post accident monitoring system is designed to assist the operator in accident surveillance. The accident monitors help to determine the nature of the accident, to predict the trend and to ascertain if corrective actions of the engineered safeguards system are functioning as required. In addition, the system provides the plant operator with information necessary to assess possible fuel or system damage. Finally, it provides material evidence for post-accident investigation into the causes and consequences of the event.

5.1.5.11 Reactor Diagnostic System

The reactor diagnostic system consists of:

- a. Loose parts monitoring
- b. Vibration monitoring
- c. Neutron noise threshold monitoring
- d. Data handling and analysis

Loose parts monitoring is accomplished with an array of strategically located accelerometers mounted external to major reactor components. The system is designed to alarm the presence of unusual noises in the reactor coolant system and moderator system and to assist in determining the location and energy level of these noises. The information is used to decide the feasibility of continuing plant operation if an abnormal condition occurs.

Vibration monitoring is accomplished by monitoring the acoustic signals and vibration generated by rotating parts. Permanently mounted accelerometers and non-contact probes are used to monitor and alarm excessive vibration of reactor coolant pumps and motors. A portable system is provided for periodic monitoring of all other accessible rotating equipment in both the primary and secondary plant. The vibration monitor provides a warning of impending equipment failure and helps measure vibration frequency trends useful for preventive maintenance and outage planning.

Neutron noise threshold monitoring is accomplished by monitoring signals from power range neutron detectors. An adjustable threshold detector provides an alarm for large cyclical variations in neutron flux on each of the detector channels. Periodic signatures

conducted by plant staff using data analysis equipment are used to verify the structural integrity of reactor internals.

Input signals from system sensors are available for audio interpretation by the maintenance personnel and for audio comparison against pre-recorded baseline information. Recording permits retention of selected audio signals for subsequent playback to help interpret and analyze data.

5.1.5.12 Containment Atmospheric Monitoring System

The containment atmospheric monitoring system monitors the hydrogen content, gamma radiation levels and fission products in the containment atmosphere. Alarms are provided for high hydrogen or high radiation level conditions. Permanent records of the measurements are provided on control room recorders.

5.1.5.13 Containment Leakage Monitoring System

The containment leakage monitoring system measures the containment overall integrated leakage rate at the required periodic intervals. The system measures and records the absolute pressure, the dewpoint temperature, and the dry-bulb temperature. Sensors having high accuracy and resolution are utilized, since changes in the measured parameters are small. A data acquisition system is provided to compute the containment leakage based on the measured inputs.

5.1.5.14 Failed Fuel Detection System

The failed fuel detector is a gamma activity monitoring device. It monitors reactor coolant for fission product gamma activity as a means for detecting leaks in the fuel rod cladding. Output from the radiation channels are recorded to provide trends, and alarmed to alert operations personnel to abnormal radiation

levels. Sampling and radiochemical analysis of reactor coolant water is used to verify the fission product activity.

5.1.5.15 Reactor Power Control System

The reactor power control system is designed to follow load changes as they are made on the turbine. The system automatically adjusts the power level of the reactor to match the power demanded by the turbine. The turbine load determines a reference temperature which corresponds to the reactor power level required to meet the turbine demand. For large and rapid variations, the turbine load signal is also compared to a nuclear power (flux) signal, resulting in a signal which corresponds to the rate at which the load is decreasing or increasing. The rate signal and the reference temperature signal are compared to the auctioneered average temperature generated in the reactor. This comparison results in an error signal proportional to the reactor power increase or decrease required. This error signal is inputted to the rod control system which performs the load change in the reactor.

5.1.5.16 Reactor Protection System

The reactor protection system receives signals from nuclear instrumentation and process instrumentation bistables, control board pushbuttons, and field-mounted devices, and combines these signals according to prescribed logic to produce actuation signals for reactor trip and engineered safeguards operation. By tripping the reactor, the system automatically keeps the reactor operating within a safe region whenever the limit of the region is approached. The safe operating region is defined by several considerations such as mechanical/hydraulic limitations on equipment, and heat transfer phenomena. Therefore, the protection system keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer

water level (to prevent water discharge through safety valves, and uncovering heaters), and also on variables which directly affect the heat transfer capability of the reactor (e.g., flow and reactor coolant temperature). Typical causes of reactor trips are:

- a. Nuclear overpower
- b. Core thermal overpower
- c. Pressurizer low pressure and high pressure
- d. Reactor coolant system low flow
- e. Low feedwater flow
- f. Low steam generator water level
- g. Low moderator level, high moderator pressure and/or temperature
- h. Manual trip

5.1.5.17 Engineered Safety Features Actuation System

The engineered safety features actuation system (ESFAS) senses selected plant parameters, determines whether or not predetermined safety limits are being exceeded and, if they are, combines the signals into logic matrices sensitive to combinations indicative of primary or secondary system boundary ruptures. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the accident. The specific functions which rely on the ESFAS for initiation are:

- a. Proper sequencing of power demands on the engineered safety features buses
- b. Containment isolation and cooling to prevent fission product release to the environs
- c. Main steam line isolation
- d. Starting the emergency diesels to assure a backup supply of power to emergency and supporting systems components
- e. Control room intake ducts isolation
- f. Containment spray actuation
- g. High and low pressure injection to the primary coolant system to maintain core cooling in event of small and large system breaks

The ESFAS is designed to comply with requirements of IEEE 279, 1971.

5.1.5.18 Reactor Plant Instrument Tubing and Fittings

The scope of supply of instrument tubing begins at the first accessible root valve at the piping and extends to the instrument shutoff valve. Materials and certification of instrument lines which are part of the pressure boundary are in accordance with the Instrument Society of America Standards.

5.1.6 Electrical Systems

The electric plant equipment conveys the electric power generated in the plant to the low voltage bushings of the generator step-up (GSU) transformers, controls and meters the electric energy, and protects the components through which the power flows. It is the source of power for the plant auxiliaries and the plant control, protection and surveillance systems during normal operation, and for the plant protection system and engineered safety features during normal operation, abnormal conditions, and accident conditions. A one line diagram for the unit electrical distribution system is shown on Figure 5.1.6-1.

The electric plant design features are as follows:

- a. The main generator, the three single phase generator set-up (GSU) transformers and the five three phase unit auxiliary transformers (UAT) are interconnected with an isolated phase bus.
- b. The generator is provided with a load break switch in the mains between the generator and the UAT tap to disconnect the generator from the offsite power system.
- c. Five (four 3-winding and one 2-winding) unit auxiliary transformers (UAT), each 24.5 kV to 13.8 or 4.16 kV, are connected to the generator main leads between the generator load break switch and the GSU transformers.
- d. Three 3-winding reserve auxiliary transformers (RAT), each 230 kV to 13.8 or 4.16 kV, are connected to an offsite transmission system.

- e. The balance-of-plant (BOP) medium voltage a-c distribution system is nominally 13.8 kV and 4.16 kV. Five separate and independent 13.8 kV buses and two separate and independent 4.16 Kv buses are provided. The five BOP 13.8 kV buses are fed from two 3-winding and one 2-winding UAT and a 3-winding RAT. The two BOP 4.16 kV buses are supplied from one low voltage winding on each of two 3-winding UAT's and two 3-winding RAT's.
- f. The Class 1E medium voltage a-c distribution system is a nominal 4.16 kV. Two separate and independent Class 1E buses are provided. Each Class 1E bus is fed from one low voltage winding on each of the UAT's and RAT's.
- g. The low voltage a-c distribution systems are a nominal 480 volts. Nineteen buses are provided for the BOP systems and eight buses are provided for the Class 1E systems.
- h. Four separate and independent Class 1E, 120 volt nominal, uninterruptable a-c instrumentation and control power supplies and distribution buses fed from the Class 1E 480 volt buses are provided.
- i. Two separate and independent Non-Class 1E, 120 volt nominal, uninterruptable power supplies fed from the BOP 480 volt buses are provided. One supplies power to BOP instrumentation and control and the other to the plant computer.
- j. The Class 1E d-c distribution and supply systems are nominally 125 volts. Four separate and independent station barriers and distribution buses are provided.
- k. The Non-Class 1E d-c distribution and supply system is nominally 125/250 volts, with center-tapped battery systems.

One center-tapped station battery and distribution system is provided.

1. One Class 1E 125 volt battery charger is provided for each Class 1E battery. One Non-Class 1E 125 volt battery charger is provided for each of the two 125 volt sections of the 125/250 volt Non-Class 1E center-tapped battery.
- m. Two redundant Class 1E, 100 percent, 6083 kW diesel generator units are provided as the standby power supply for the Class 1E buses, and are automatically connected to their respective buses when the normal and preferred power supplies are not available.

Motor starting voltage and frequency and allowable operational variations, at which the required starting and operating torques are developed, are as follows:

- a. Continuous operation of a-c motors
 - 1) Voltage: $\pm$ 10 percent of rated
 - 2) Frequency: $\pm$ 5 percent of rated
- b. Starting and short time (approx. 30 seconds) operation of a-c motors:
 - 1) Class 1E (Voltage): 75 percent of rated
 - 2) Non-Class 1E (Voltage): 80 percent of rated
- c. d-c Motors (Voltage): 210 to 280 volts

All Class 1E loads are furnished with a-c or d-c power from one of the following: the Class 1E a-c emergency buses, the Class 1E uninterruptible instrumentation and control a-c power supply and the Class 1E d-c buses.

The normal power supply for the plant electric auxiliaries is from the main generator through the unit auxiliary transformers. The preferred emergency power supply is from the 230 kV offsite power supply via the reserve auxiliary transformers. The alternate access power supply is from the 500 kV offsite power supply via the generator step-up transformers and the unit auxiliary transformers, when the generator load break switch is open. The standby emergency power supply is from one of the two redundant diesel generator units to the corresponding Class 1E medium voltage bus.

The power and control circuits, including circuit breakers and cabling, to all Class 1E loads are qualified, channeled and separated to meet Class 1E requirements. Protective devices are coordinated on the basis that protection of the Class 1E systems is the primary goal.

The safety related design bases for the electric power system are tabulated in Table 5.1.6-1.

Table 5.1.6-2 presents allowable ranges of temperature and limits for exposure to radiation for electric equipment. Design ambient conditions for spaces housing electric equipment are based on these ranges and limits plus a minimum of 5 percent for margin.

Physical and electrical separation of equipment and systems is provided to assure the availability of the minimum required equipment during any design basis event. Table 5.1.6-3 presents acceptable types of physical and electrical separation and their application. Physical separation of equipment and circuits is

done in such a way that the single failure criterion is met and that the independence achieved cannot be compromised by adjacent or supporting mechanical and structural systems.

5.1.6.1 Switchgear

A main generator load break switch is provided to disconnect the generator rapidly from the offsite system when no fault is present. This allows rapid connection to the alternate access power supply (GSU transformers and UAT). The switch comprises three single-pole, water-cooled load break switches with a spare single pole and an independent cooling system. The cooling system has redundant active components. The medium voltage metal-clad switchgear comprises five Non-Class 1E 13.8 kV buses, two Non-Class 1E 4.16 kV buses and two Class 1E 4.16 kV buses. Motors rated 3000 hp and above are rated 13.2 kV and motors rated 250 hp to 2500 hp are rated 4.0 kV. Transfer schemes are provided for automatically and manually transferring each train between the preferred emergency power supply and the normal power supply. Load shedding schedules are provided for automatically stripping all Class 1E buses whenever there is an offsite power failure. Sequential loading schemes are provided for the Class 1E buses to automatically connect selected engineered safety features to those buses after load shedding, and re-energizing the buses from the diesel generator units. Overcurrent protection is provided for all circuits. Differential protection, overload protection and zero sequence overcurrent ground protection is provided for all medium voltage motor circuits.

Non-Class 1E and Class 1E 460 volt motor control centers are provided for power distribution to motors 100 hp and below, lighting loads and miscellaneous loads such as motor-operated valves, resistance heaters, heat tracing and motor space heaters.

5.1.6.2 Station Service Equipment

Five auxiliary transformers (UAT) and three reserve auxiliary transformers (RAT) are provided to furnish power to the plant auxiliary power system. Each transformer is sized with sufficient margin to carry the plant auxiliary load under the heaviest load conditions. Each transformer is protected with differential protection schemes and sudden internal overpressure devices.

Unit substations are provided to transform the medium distribution voltages to the low distribution voltage for Non-Class 1E and Class 1E low voltage loads. Motors rated 125 hp through 200 hp are connected to the unit substations.

Overcurrent protection is provided for all circuits. Overload protection is provided for motor circuits. The unit substations for the cooling towers are fed from a loop feeder.

The battery systems comprise the plant Non-Class 1E and Class 1E batteries and battery chargers. Each Class 1E d-c bus is supplied from a Class 1E battery and a Class 1E battery charger. The Non-Class 1E 125/250 volt d-c bus is supplied from a 125/250 volt center tapped battery and two 125 volt battery chargers, one for each 125 volt section of the 125/250 volt battery. During normal operation d-c power is supplied from the battery chargers. During emergency operation d-c power is supplied from the batteries. During startup and shutdown d-c power is supplied from whichever source is available.

Two redundant diesel generators are provided to furnish the onsite source of emergency a-c power to the Class 1E 4.16 kV buses.

Each diesel generator unit is provided with redundant automatic air starting systems that are initiated when loss of offsite power, loss of power to engineered safety features or when reactor trip occurs. The rating for the diesel generator units was chosen so that each unit has the capability to continuously operate all protection systems and engineered safety features that are necessary for a safe and orderly shutdown following a loss of coolant accident concurrent with loss of offsite power. Minimum voltage that can be experienced at the diesel generator terminals during motor starting is 80 percent. Rating, configuration and switching of the diesel generator units are designed to prevent their use for any purpose other than that of standby power supply in accordance with preferred practice.

Four Class 1E and two Non-Class 1E dual input solid state inverters are provided to serve as uninterruptible power sources for miscellaneous vital and non-vital a-c and plant instrumentation loads. The inverters are supplied with power from the a-c buses through regulating transformers or directly from the station batteries.

5.1.6.3 Switchboards

Four Class 1E and two Non-Class 1E a-c power distribution panels are provided to distribute a-c power from the inverters to the 120 volt or 120/240 volt uninterruptible loads. They are configured as one panel per inverter for both Class 1E and Non-Class 1E equipment.

Four Class 1E and Non-Class 1E d-c power distribution switchgear lineups are provided to distribute d-c power from the batteries and their associated chargers. There is one lineup per station battery/charger combination.

Sixteen feet of control benchboard is provided in the main control board lineup for control and data acquisition of the main generator and the auxiliary electric power system.

One electric system relay panel lineup is provided for protection and metering of the main generator, the generator step-up transformers and the unit and reserve auxiliary transformers. The main generator is protected by high speed differential, ground current, loss-of-field, negative sequence overcurrent, and voltage restrained overcurrent relays. The main generator, the generator step-up transformers and the unit auxiliary transformers are protected by power directional overall differential relays. The reserve auxiliary transformers are protected by power directional differential relays.

5.1.6.4 Power and Control Wiring

Isolated phase bus is provided to interconnect generator terminals, GSU transformer low voltage terminals and UAT high voltage terminals. This bus is force-cooled with redundant active components in the cooling unit.

The plant wire and cable consists of three conductor and triplexed single conductor power cable, multi-conductor cable, coaxial, triaxial, shielded twisted pair and multi-shielded twisted pair and shielded quad instrument wire and containment electrical penetrations. Materials for insulation systems are selected to provide optimum system performance in the areas of physical stability, tensile strength, flexibility, aging characteristics, resistance to abrasion, ozone (where required), water absorption, heat distortion, solvent extraction, irradiation, self-extinguishing and non-propagating fire characteristics and resistance to corona effects where required. Wires and cables are assigned to load groups, whether safety related or not, and the same cable that

is qualified for use in Class 1E systems is used in Non-Class 1E systems to reduce the hazard of unqualified cables being inadvertently routed between two redundant load groups or installed in Class 1E systems. In addition to separation by load groups, wire and cable is also separated by energy level to reduce heating and arcing fault problems.

Wire and cable routing is governed by the following:

- a. Requirements for the power supply, control network and/or instrumentation signals
- b. Requirements for loading
- c. Requirements for physical separation of redundant criteria
- d. Avoidance of high hazard areas (e.g., areas subject to high ambient temperatures, missiles, fires, and irradiation)
- e. Areas having high ambient temperature (e.g., areas near uninsulated, main-steam pressure and relief valves)
- f. Protection from missiles, fire, and/or irradiation, when required
- g. Single failure criterion and the effects of common failure modes
- h. Simplicity of layout
- i. Ease of insulation
- j. Ease of access

TABLE 5.1.6-1

SAFETY RELATED DESIGN BASES FOR THE ELECTRIC POWER SYSTEM

1. Subsystems, equipment and components which are required to achieve a protective function, to perform a protective action or to provide power for engineered safety features are Class 1E design as defined below.
2. Class 1E Design:
 - a. Considers interactive effects of plant conditions and natural phenomena to the extent that power required by the plant protection system and engineered safety features are available during abnormal and accident conditions.
 - b. Is based on the safe shutdown earthquake (SSE) in order to assure safe reactor shutdown and removal and dissipation of reactor stored energy and decay heat for an indefinite period in the event of an SSE.
 - c. Includes provisions to minimize failures due to flame or fire damage and to detect, confine and promptly extinguish any fire which might occur.
 - d. Includes provisions to allow periodic inspection and testing of systems and equipment.
3. Class 1E power sources, power supplies, distribution systems and load groups, have sufficient separation and independence so that loss of any group does not prevent the minimum safety actions from being performed.

4. Class 1E load groups are provided with separate and independent power supplies in order to prevent a common failure mode from being established among the load groups.
5. The degree of physical separation required for redundant Class 1E equipment, including cable and raceways containing cable, is based upon the hazards which exist in the vicinity of the equipment and which would constitute a common failure mode between the redundant equipment if no separation existed.

TABLE 5.1.6-2
DESIGN AMBIENT CONDITIONS FOR ELECTRIC EQUIPMENT

<u>Type of Equipment</u>	<u>Limit</u>	<u>Ambient Temperature Limit (°F)</u>		<u>Ambient Radiation Limit (Rads)</u>	
		<u>Equipment</u>	<u>Equipment Space</u>	<u>Equipment</u>	<u>Equipment Space</u>
Battery	Max	90	N/A	N/A	N/A
Battery	Min	77	80	N/A	N/A
Cable	Max	104	100	1×10^8	1×10^7
Cable	Min	N/A	N/A	N/A	N/A
All Other**	Max	104	100	N/A	N/A
All Other**	Min	40*	50*	N/A	N/A

* Or above dewpoint temperature, whichever is higher.

** Sensitive relays and other electrical devices are placed in controlled environment spaces such as the control room, as applicable.

TABLE 5.1.6-3

ACCEPTABLE PHYSICAL AND ELECTRICAL SEPARATION

<u>Type</u>	<u>Designation</u>	<u>Used for Protection From:</u>
A	Physical	Low Energy Missiles Mechanical Protection
B	Physical	High Energy Missiles Radiation
CDE	Physical	Heat, Fire
FGH	Electrical	Common Failure Mode

DETAILED DESCRIPTION

<u>Type</u>	<u>Barrier</u>	<u>Example</u>
A	Metal	Sheet metal on rigid frame Rigid steel sleeve or plate Rigid steel conduit
B	Concrete	Concrete block wall Concrete foundation Reinforced concrete wall
C	Flame Resistant	Mineral, wool or fiberglass Ablative coating Asbestos cement sheet support on light steel frame
D	Air Space	1'-0"*** or 3'-0" in horizontal direction 3'-0"*** or 5'-0" in vertical direction
E	Minimum Air Space	6" in horizontal direction* 6" in vertical direction*
F	Prohibition of sharing among redundant circuits unless electrical isolation is provided by buffers.	
G	Prohibition of automatic transfer of redundant safety circuits between Class 1E power supplies or power sources.	
H	Prohibition of automatic transfer of Class 1E power supplies or power sources between redundant safety circuits.	

* In conjunction with a barrier of Type A, B or C.

** In cable spreading areas only.

5.1.7 Auxiliary Systems

The auxiliary systems of the HWR plant include fuel storage and handling, water systems process auxiliaries, HVAC, and other auxiliaries.

A description of the various auxiliary systems is provided in the sections that follow. The systems are:

- New Fuel Storage (Section 5.1.7.1)
- Spent Fuel Storage and Handling (Section 5.1.7.2)
- Spent Fuel Storage Pool Cooling and Cleanup System (Section 5.1.7.3)
- Primary Component Cooling Water Systems (Section 5.1.7.4)
- Demineralized Water Makeup System (Section 5.1.7.5)
- Ultimate Heat Sink (Section 5.1.7.6)
- Moderator Collection System (Section 5.1.7.7)
- Moderator Deuteration and De-Deuteration System (Section 5.1.7.8)
- Moderator D₂O Sampling System (Section 5.1.7.9)
- Moderator Cooling System (Section 5.1.7.10)
- D₂O Supply System (Section 5.1.7.11)
- D₂O Vapor Recovery System (Section 5.1.7.12)
- D₂O Cleanup System (Section 5.1.7.13)
- D₂O Upgrading System (Section 5.1.7.14)
- Compressed Air System (Section 5.1.7.15)
- Air Conditioning, Heating, Cooling and Ventilating Systems (Section 5.1.7.16)
- Fire Protection System (Section 5.1.7.17)
- Auxiliary Steam System (Section 5.1.7.18)
- Standby Power Supply and Auxiliaries (Section 5.1.7.19)
- Primary Heat Transport D₂O Sampling System (Section 5.1.7.20)
- Primary Heat Transport D₂O Collection System (Section 5.1.7.21)

- ° Primary Heat Transport Deuteration and De-Deuteration System (Section 5.1.7.22)
- ° Containment Air Purification and Cleanup Systems (Section 5.1.7.23)

5.1.7.1 New Fuel Storage

Functions

New fuel storage and handling equipment are provided in the Reactor Service and Fuel Handling Building. The handling facilities provide for the safe transfer of new fuel to the containment for manual loading of new fuel bundles into the motorized new fuel ports.

Design Requirements

The new fuel storage room is sized to accommodate a nine month fuel inventory and is able to temporarily store the fuel for the initial core loading.

System Description

The new fuel storage room is sized to accommodate a nine month supply of fuel elements, equivalent to approximately 7500 bundles. Since criticality in air is not a problem with this low-enriched fuel, no special precautions are taken in fuel bundle spacing.

The fuel bundles are stored in such a manner to preclude damage from dropped objects or from seismic events.

The new fuel bundles are loaded onto pallets for transferring from storage to the fuel loading area in the containment. The fuel pallets are passed into the containment through the containment

equipment/personnel air lock. The fuel pallet is then placed on an elevator that lowers the fuel to the new fuel loading area.

The fuel is manually loaded into the magazines of the two new fuel ports which penetrate into the fueling machine maintenance locks. Handling equipment is available and used to avoid damage to the fuel bundles during manual loading. Each fuel bundle is visually examined as well as performing other examinations, if required, prior to loading into the ports.

The fueling machine locks onto the loading port and accepts up to 10 fuel bundles into a magazine in the machine.

5.1.7.2 Spent Fuel Storage and Handling

Functions

The flow of fuel through the plant is shown schematically in Figure 5.1.7-1. New fuel stored in dry racks in the new fuel storage room of the reactor service and fuel handling building is transferred through the equipment lock into the new fuel loading area in the containment. In the loading area of the containment, the fuel bundles are given final visual inspection and are manually loaded into the magazine of the new fuel ports which penetrate into the fueling machine maintenance locks. Refer to Figure 5.1.7-1. Mechanisms of the ports are motorized and can be controlled remotely.

One of the two fueling machines locks onto the new fuel port where it can accept as many as 10 new fuel bundles into five magazine positions, by loading 2 bundles into each magazine.

Design Requirements

Spent Fuel Storage

Spent fuel storage facilities are provided in the reactor service and fuel handling building to accommodate approximately one year plus one full core of irradiated fuel. In addition, a separate reception/discharge compartment is provided for receiving spent fuel from the containment and for loading spent fuel into a shipping cask.

Fuel Handling Systems

Fuel handling systems are provided to supply new fuel to the reactor core and remove spent fuel in such a manner that shutdown of the reactor is unnecessary.

Fuel Changing

The on-power fuel changing equipment is located in the containment and consists of two identical, unshielded fueling machines, which are operated remotely. The fueling machines are normally stored in two fueling machine maintenance locks and are suspended from tracks. Each set of tracks connects with a bridge at each face of the reactor. Powered shielding doors separate the maintenance locks from the reactor and, when closed, allow access to the fueling machines while the reactor is at full power.

While in the maintenance locks, the fueling machines can lock on to the new fuel port to accept new fuel, to the service port for maintenance or service, or onto the spent fuel port to discharge spent fuel.

Operations on the reactor are performed under remote control. One of the fueling machines locks on to the new fuel port and accepts up to 10 new fuel bundles, with the other machine accepting 8 for a refueling operation leaving one empty magazine to accept spent fuel. The shielding doors are opened and the two machines travel along the tracks onto the fueling machine bridges at each face of the reactor. The bridges are then raised and the fueling machines are positioned on the bridge so that the machines are located at each end of the pre-selected fuel channel.

A description of the fueling machine is given later in this section.

Both fueling machines move forward and lock on to the fuel channel end fittings to form a secure and leak-tight joint with the channel. Each machine is pressurized to approximately the same pressure that exists in the end fitting and the plugs from the snouts of the machines are removed. The fueling machines remove the fuel channel closures from the pressure tube and store them in a magazine. The machine at the upstream end of the fuel channel removes the shield plug and inserts a new fuel bundle into the channel from a new fuel magazine. The downstream machine removes the shield plug and receives a spent fuel bundle which is also stored in the fuel magazines. Two fuel bundles can be stored in each magazine position. Either machine can load or receive fuel, depending upon the direction of flow of the D_2O in the channel being fueled, fueling normally being with the direction of flow.

When the required number of fuel bundles have been inserted, the shield plugs and channel closures are replaced and the closures checked for leaks. The machines replace the plugs in the snouts and unlock from the end fittings. They are then retracted from the end fitting and returned to the fueling machine maintenance

locks, or can be moved to another fuel channel for further fueling operations.

Upon returning to the maintenance lock, the fueling machine containing the spent fuel discharges it through the spent fuel port.

Spent Fuel Handling

The spent fuel handling equipment consists of discharge and transfer equipment in the containment and spent fuel reception and storage equipment and bays in the reactor service and fuel handling building. The transfer of spent fuel between buildings is carried out under water through a transfer canal. Handling and storage space is also provided under water for irradiated parts of the reactivity control mechanisms, which may be removed from the reactor.

The discharge and transfer operations are controlled remotely, while operations in the storage bays are carried out manually under water using long tools, and aided by powered cranes and hoists.

The equipment incorporates devices for canning defective fuel bundles and is arranged to reduce the radiation exposure to acceptable levels to personnel when handling defective fuel.

The discharge equipment comprises two valved spent fuel ports located above the water canal, while the transfer equipment is located in a shielded room and extends down under the water into a reception bay in the RS&FH building. In operation, the fueling machine seals to the spent fuel port in the fueling machine maintenance lock and the heavy water level in the head is lowered below the snout. The snout plug is then removed and the machine

discharges the spent fuel, two bundles at a time, through the spent fuel ports onto the transfer mechanism. The mechanism lowers the fuel into the light water of the fuel transfer canal and transports it into the reception bay in the RS&FH building. Defective fuel is manually inserted under water into cans before transfer, to limit the spread of contamination.

The reception bay connects under water with the main storage pool on one side and with a small defective fuel pool on the other side. In the reception pool, the normal spent fuel is remote manually off-loaded under water from the transporter and placed onto storage trays. In addition to the trays of spent fuel, irradiated elements of the reactivity control mechanisms received in the reception pool are moved into the storage pool for under water storage. Space is also allocated for a shipping cask which would be lowered into the water of the pool for loading the fuel.

Fueling Machines

For descriptive purposes, the fueling machine consists of a head, a suspension and a self-propelled carriage. The machine travels on an elevated track when in the maintenance lock and on the fueling machine bridge when at the face of the reactor. Heavy water, air, electric power and oil hydraulic connections and control signals are supplied to the fueling machine through a flexible catenary which connects the mobile machine to the stationary auxiliary systems.

The function of the fueling machine head is to pick up new fuel from the new fuel port, load it into the reactor, receive spent fuel from the reactor and discharge it into the spent fuel port.

The snout assembly is located at the front end of the fueling machine head. It consists of an antenna assembly which ensures

that the fueling machine head is in adequate alignment with an end fitting before the head clamps on to it. The snout clamping mechanisms latch the fueling machine head on to an end fitting to form a pressure tight seal. When not attached to a fuel channel or a port, the snout is sealed by a snout plug. When attached to a fuel channel, the snout plug is stored in one of the 12 magazines.

The magazine assembly is located behind the snout assembly and the separators. The magazine housing and end cover are forgings which take the form of a cylindrical pressure vessel with flat heads. The housing and the end cover are fastened together by a two-piece Grayloc clamp and seal ring.

A twelve-position rotatable magazine is located concentrically inside the magazine housing and is driven by an oil hydraulic motor. The twelve magazine positions provide for five fuel stations (holding two bundles each), two channel closures, two shield plugs, one snout plug, one ram adapter and one guide sleeve and tool.

The guide sleeve is a tube with an internal diameter nominally equal to that of the coolant channel. It is inserted into the snout and the end fitting to facilitate the passage of fuel bundles into the magazine.

Two separator assemblies are located between the snout and the magazine assemblies. The separators are used for sensing the passage of fuel bundles and shield plugs into the machine and for separating the spent fuel bundles from the remainder of the fuel string.

The ram assembly is located immediately behind the magazine assembly. The telescopic rams operate all the plugs, the ram adapter, the guide sleeve and tool, and push fuel bundles into

the fuel channel. The rams are enclosed in a cylindrical housing and are driven by two pairs of water-lubricated ball screws arranged within the housing. The ball screws penetrate the pressure boundary at the rear of the housing and are sealed by self-energized hydrostatic face seals. An oil lubricated gearbox and two oil hydraulic motors drive the ball screws.

The head is secured to the carriage through a suspension. This is an assembly which allows the head to align itself with a channel end fitting to reduce the forces exerted on the end fitting by the head. The catenary hoses and cables are connected to the head by quick-disconnect couplings. These, in conjunction with a mechanical disconnect, enable the head to be removed remotely in the event that the spent fuel is stuck in the head and cannot be discharged by normal methods. The same couplings and mechanical disconnect are also utilized for removal of the head for routine maintenance.

The fueling machine carriage is equipped with eight wheels which run on the tracks in the fueling machine maintenance lock and on the fueling machine bridge. It is electrically driven for horizontal traverse, the eight wheels enable it to cross the gaps between the tracks and the bridge. Slow speed is provided for homing motion. The fueling machine bridge is elevated by 4 ball screws along 2 vertical columns.

Auxiliaries and Catenaries

D₂O, air and oil hydraulic systems in addition to electrical power are provided for operation of the fueling machines and are located below the maintenance locks and are accessible at full reactor power. The D₂O system comprises equipment to supply both fueling machines with filtered heavy water at the required temperatures and pressures. D₂O is drawn from and returned to

the heat transport system storage. Pressurized air is provided from the plant services and two oil hydraulic power units supply the oil hydraulic devices on the machines. The operating valves and gauges for all three systems are centralized in a separate set of panels for each machine.

The catenary, connecting the auxiliaries to each fueling machine, consists of a power track and a free loop supported by a trolley.

5.1.7.3 Spent Fuel Storage Pool

Cooling and Cleanup System

The spent fuel storage pool cooling and cleanup system removes heat from the spent fuel in the storage pool as well as the spent fuel reception bay. The system provides clarification of the water for visibility and to reduce the activity of the water to acceptable conservative levels.

Design Requirements

The spent fuel storage pool cooling and cleanup system is designed to:

- ° Maintain the spent fuel pool water at a temperature less than 100°F when the pool contains an accumulated storage of one year of spent fuel and 140°F during abnormal loading when one full reactor core fuel load is added to the pool.
- ° Maintain purity and clarity of the water to permit the underwater observation of fuel handling operations.

- Capability to remove suspended and dissolved radionuclides and maintain pool water activities to acceptable conservative values.
- Capacity to maintain a minimum water level for shielding during all normal phases of operation and for abnormal plant operating conditions.

System Description

Decay heat is removed from the spent fuel stored in the spent fuel pool by circulating the water through heat exchangers. Cooling water to the heat exchangers is supplied from the primary component cooling water system.

The system consists of a main fuel pool loop from the pool to the pumps and heat exchangers and back to the pool. A side stream through the spent fuel pool skimmers is directed to pumps and heat exchangers then through the ion exchanger. Cooling water for the reception bay is taken off the system downstream of the heat exchangers, where it flows through the pools and then flows to the purification pump, ion exchangers, and the heat exchanger. By adjusting the line valves, the flow to any part of the system may be varied to suit the purification requirements. A vacuum cleaning system is provided for cleaning the bay floors.

Piping that penetrates the spent fuel storage pool is located to provide a sufficient depth of water over the stored fuel in the event of a pipe break in the system. Siphon breakers or vents are provided in the system to prevent the loss of fuel pool water.

The cooling and cleanup system components are furnished to meet the requirements of the ASME B&PV Code, Section III, Class 3.

The material for the components is either austenitic stainless steel or other material having acceptable corrosion-resistance properties.

Two circulation pumps are provided for the spent fuel pool cooling loops. Each pump is rated at 100% of the normal cooling load requirements. The second pump is normally used as a redundant spare but may be used should it become necessary to temporarily store partially spent fuel from the reactor.

A pump and ion-exchangers are provided for fuel pool water clarification. Normally, one pump will operate; however, one pump from the cooling loop may be used if the demand for water clarification exceeds the capacity of the normal pump flow. The ion-exchangers are sized to handle 50 percent of the combined flow from the clarification and cooling loop pumps.

The demineralized water distribution system provides the initial fill for the pools, for transfer cask wash-down, fuel pool water makeup, and for spent resin transfer. Water discharged from the pools and any leakage from the cooling and cleanup system is collected and discharged to the radioactive liquid waste system. Emergency makeup water is provided from the demineralized water storage tank and from the plant fire protection water system.

Instrumentation for the spent fuel storage pool cooling and purification system includes:

- ° Control room indication of water level, temperature and alarms for high temperature and high/low radiation levels of the spent fuel pool and the fuel reception and inspection pool.

- ° Local indication of water flow in the cooling loop and the purification system.
- ° Control and status indication of the pool cooling water pumps from the control room.

Control of the pool water temperature is attained through adjusting valves in the cooling water system.

5.1.7.4 Primary Component Cooling Water System

Functions

The Primary Component Cooling Water System (PCCW) circulates cooling water to those heat exchangers required to perform a safety function or for the heat exchangers containing tritiated heavy water. The PCCW System operates as a closed loop during both normal and emergency operating conditions. During emergency conditions the heat exchangers not required to perform a safety function are valved-off.

Design Requirements

The Primary Component Cooling Water System is designed to remove heat generated through the heat exchangers of systems either required to perform a safety function or containing tritiated heavy water. The heat thus removed is transferred to the Circulating Water System through primary component cooling water system heat exchangers during normal plant operation conditions. During emergency plant operating conditions various valves in the piping system are realigned, non-essential heat exchangers are valved-off, and the heat removed from the essential heat exchangers of the PCCW System is transferred to the ultimate heat sink. A pump is provided to circulate the water from the ultimate heat

sink through the PCCW System heat exchanger, then back to the ultimate heat sink.

System Description

The Primary Component Cooling Water System is a closed cooling water system designed to remove heat from various systems provided for safeguard functions or from systems containing tritiated heavy water. Rejected heat from the system is transferred to the Circulating Water System during normal plant operation, or to the ultimate heat sink during emergency plant operating conditions. During emergency plant operating conditions, systems not required for safeguard functions are valved-off from the PCCW System.

The system consists of two 100 percent redundant closed cooling water loops. Each loop is furnished with a heat exchanger for transferring from the loop to the Circulating Water System, redundant pumps for circulating water through the loops and component heat exchangers, and an expansion tank having provisions for adding water treatment chemicals. The redundant pumps are furnished with electrically driven motors, each connected to separate emergency diesel-generator supply busses.

In an emergency condition requiring plant shut-down, the cooling water for the Nuclear Service Water System is valved from the Component Cooling Water System to the Ultimate Heat Sink.

5.1.7.5 Demineralized Water Makeup System

Functions

The demineralized water makeup system provides and maintains a supply of treated water for makeup to the power conversion system and for other various plant process operations.

Design Requirements

Water for the makeup system is obtained from the raw water supply. Two demineralizer trains, each having a capacity of 240,000 gallons per day for a total capacity of 480,000 gallons per day, form the system.

The demineralizers will produce water of the following quality:

Specific conductivity at 25°C	1 micromho/cm
pH at 25°C	6-8
Silica (as SiO ₂)	0.02 ppm
Chloride (as Cl ₂)	0.02 ppm

The demineralized water makeup system is not a safety related system.

System Description

The raw water passes through a train of clarifiers, sand filters, and a vacuum deaerator for turbidity control and removal of oxygen. From there the water passes through two parallel lines of demineralizers. Each line of demineralizers consists of a strong acid (cation) exchanger, a strong base (anion) exchanger and a mixed-bed exchanger. The treated water is then pumped to the condensate storage tank and demineralized water storage tank.

Each demineralizer line has a capacity of 240,000 gallons per day (166.7 gpm) or a total of 480,000 gallons per day for the system (20 hours in service and 4 hours in regenerating mode). During normal operation, one demineralizer line can supply normal plant makeup requirements.

The system includes in-line acid and caustic systems for regeneration requirements. The regeneration system includes pumps, metering equipment, dilution tanks and dilution control. Storage tanks are provided for the concentrated sulfuric acid and 50 percent sodium hydroxide each having a capacity of 5,000 gallons.

Buried concrete, stainless steel-lined, rectangular tanks, each measuring 50'-0" x 35'-6" x approximately 28'-0" free height, are provided for the demineralized water storage and the condensate storage. The tanks have a nominal capacity of 400,000 gallons. The storage tanks are designed to seismic Category I requirements and are covered with a reinforced concrete slab for missile protection.

5.1.7.6 Ultimate Heat Sink

Functions

The Ultimate Heat Sink is provided to supply cooling water to remove heat from the nuclear plant equipment which is essential for a safe and orderly shutdown of the reactor and to maintain it in a safe condition.

Design Requirements

The Ultimate Heat Sink must be capable of accomplishing its safety function for a normal cooldown or an emergency cooldown following a loss-of-coolant-accident (LOCA) without the availability of off-site power. The Sink shall provide this cooling capability without supplemental water makeup for a period of 30 days. Provisions are to be incorporated to furnish supplemental water, if needed, from outside sources for continued operations after 30 days.

The Ultimate Heat Sink will be designed to seismic Category I requirements to assure that it will provide its safety functions under abnormal environmental conditions.

System Description

The Ultimate Heat Sink provides cooling water to essential nuclear plant equipment during abnormal shutdown of the reactor. The normal operating mode of the cooling systems is described in the System Design Description for the Primary Component Cooling Water System (Section 5.1.7.4).

During abnormal operations, cooling water is furnished to one of two circulating water loops that removes heat from one of the redundant primary component cooling water heat exchangers. Each of the redundant loops are furnished with redundant pumps having electric motors connected to the redundant emergency electric power buses.

Should the plant cooling towers become unavailable, either due to normal or abnormal occurrences, the essential heat removal loads will be transferred to the Ultimate Heat Sink Cooling towers.

The Ultimate Heat Sink consists of a concrete basin with a cooling tower and a pumphouse. All components are designed to seismic Category I requirements.

The concrete basin is sized to provide adequate cooling for 30 days without makeup. The pond is not used for cooling water purposes during normal plant operations. Some water loss from the basin is expected during standby due to normal evaporation. This evaporation loss will require occasional blowdown to maintain the basin water chemistry. A gravity makeup water line is provided from the circulating water pumphouse to the cooling tower basin

to maintain the proper water elevation. Each of the 100 percent redundant loops circulates 16,470 gallons per minute through the primary component cooling heat exchanger then through the cooling tower and into the storage basin.

The cooling tower is constructed from three cells, each cell will handle 50 percent of the full circulating water flow. Two cells are sized to handle the full loop flow, using the center cell as a common cell to both loops. Should it become necessary to operate both loops simultaneously, the pipelines supplying water to the tower will be valved to evenly distribute the water to each cell.

5.1.7.7 Moderator Collection System

Functions

The Moderator D₂O Collection System is provided to recover any heavy water that may leak from the moderator process auxiliaries.

Design Requirements

The collection tank and pump are located in the reactor service and fuel handling building (RS&FH). Heavy water is collected from the moderator process auxiliaries located in the containment, penetrations building, and in the RS&FH building. Heavy water from the collection points flow by gravity to the collection tank.

This is not an engineered safeguards system.

Design Description

The Moderator D₂O Collection System consists of a collection tank and transfer pump. Collection points, from which the collected water flows to the central collection tank, are located at moderator process auxiliaries located in the containment, RS&FH building, and the penetrations building. Design parameters for the system are given in Table 5.1.7-1 and a schematic is shown on Figure 5.1.7-2.

Water flows by gravity from the collection point to the collection tank through pipes which are equipped with flow indicators. The flow indicator permits the plant operator to identify the source of heavy water and the flow rate.

A transfer pump is provided to pump the contents of the collection tank to the moderator system or to drums. If samples of the collection tank contents show it to be of reactor grade it is returned to the moderator system. If the sample shows the water to be degraded, it is pumped into drums, after which it will be upgraded.

The collection tank is provided with both local and control room indications of the water level. High water levels in the tank are annunciated. The pump is provided with pressure gauges on the suction and discharge lines. A motorized valve is provided on the pump discharge. Pipe lines penetrating containment are provided with containment isolation valves.

TABLE 5.1.7-1
MODERATOR D₂O COLLECTION SYSTEM
DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
General:	
Expected Collection Flow Rate	1.5
Return Flow Capacity	18 gpm
Pumps:	
Name	D ₂ O Collection Pump
Quantity	1
Type	Canned Centrifugal
Duty	Intermittent
Pump Data:	
Fluid	D ₂ O
Flow Rate	18 gpm
Operating Temperature	100 to 200°F
Head	100 ft.
Design Pressure	150 psig
Design Temperature	250°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Construction:	
Material	Austenitic Stainless Steel, Type 304
Applicable Code	ASME Section VIII
Tanks	
Name	D ₂ O Collection Tank
Quantity	1
Type	Horizontal Cylinder with ASME F&D Heads with Saddle Supports
Capacity	11 cubic feet

TABLE 5.1.7-1 (Cont.)

<u>Item</u>	<u>Value</u>
Design Conditions:	
Operating Pressure	2 psig
Operating Temperature	100 to 200°F
Design Pressure	30 psig
Design Temperature	250°F
Construction:	
Material	Austenitic Stainless Steel, Type 304
Applicable Code	ASME Section VIII

5.1.7.8 Deuteration and De-Deuteration System

Functions

The Moderator Deuteration and De-deuteration System is used to remove corrosion products, ionic impurities and liquid poisons (boron and gadolinium) that may be injected into the moderator for reactivity control purposes. In addition, the system is used to remove light-water from resins before the resins are introduced into the system, and to remove heavy-water from spent resin before it is discharged to resin disposal tanks.

Design Requirements

The system is designed to maintain the moderator water within specification limits under all planned plant operating conditions. Components of the system are designed to comply with seismic Category I requirements to assure containing the heavy water during abnormal plant or environmental conditions. Materials of the wetted surfaces are compatible with the fluid contained and will not contaminate the fluid.

Shielding is provided around the de-deuteration tank to protect plant personnel from the radiation contained in the spent resin.

This is not an engineered safeguards system but is constructed to seismic Category I, ASME B&PV Code, Section III, Class 3 requirements. Design parameters for the system are given on Table 5.1.7-2 and a schematic is shown on Figure 5.1.7-3.

Design Description

The deuteration and de-deuteration tanks are provided with removable screens. These screens are capable of withstanding high differential

pressures and are designed to minimize plugging. Both the D_2O and H_2O tanks incorporate a conical bottom to facilitate resin removal after slurring the resin. The tanks are approximately 2 feet in diameter and have overall height about 5-1/2 ft. The tanks are designed to hold approximately 7 cubic feet of ion exchange resin.

Both the D_2O tank and the H_2O tank have a capacity of approximately 15 ft³. These tanks are provided with multiple position (4 for the D_2O head tank and 5 for the H_2O head tank) level switches and alarms. Both tanks are located above the deuteration and de-deuteration tanks so that gravity flow can be used for deuteration and de-deuteration. The level switches in the tanks will operate the automatic ON-OFF control valves located in the deuteration tank outlets.

The pump in the system has a capacity of about 25 gpm and is the canned type in order to eliminate seal leakage problems. The pump is used for D_2O service only. The strainers at the pump suction and discharge lines prevent resin from entering the pump. The strainers in the system, as well as the wedge-wire screens, retain any particles larger than 0.2 mm. The resin head size is 0.38-0.60 mm.

The material of construction for the tanks, pump, and other components of the system is Type 304 austenitic stainless steel or appropriate synthetic elastomer.

The valves in the system, except the two located in the main lines to the storage tanks in the D_2O cleanup system, are manual valves. These two valves are ON-OFF automatically operated control valves. Diaphragm valves are used for D_2O service and ball valves where resin transfer is involved.

The deuteration and de-deuteration processes can be carried on at the same time, independently of each other.

At the end of the deuteration and de-deuteration operation, the deuteration tank will be full of heavy water and the de-deuteration tank will be full of light water. The heavy water from the deuteration tank will be transferred to the de-deuteration tank using instrument air. The H_2O in the de-deuteration tank is transferred to the resin disposal vault using instrument air, before D_2O is transferred to it.

Generally, flexible hose connections are provided at every point where it is possible to mix D_2O with H_2O in order to eliminate inadvertent downgrading of D_2O .

The moderator deuteration and de-deuteration system is constructed to conform with the ASME Boiler and Pressure Vessel Code, Section III, Class 3.

The de-deuteration tank is shielded with light concrete 4.6 ft. thick. One side of the shielding is removable to allow access to the vessel.

The resin slurry lines from the ion exchange columns to the de-deuteration tank and from the de-deuteration tank to the resin disposal value are shielded with 1 ft. thick concrete.

The screens in the tanks can be removed for cleaning in the event they plug. Flanged connections are provided for this purpose. The strainer screens must be cleaned periodically.

TABLE 5.1.7-2
MODERATOR DEUTERATION AND DE-DEUTERATION SYSTEM
DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
General:	
Flow Rate	1.3 to 2.5 gpm
D ₂ O Required to Deuterate 7 ft. of Resin	9.5 ft ³
D ₂ O/H ₂ O Interface Velocity	0.01 to 0.02 ft/min
Applicable Code	ASME Section III, Class 3
Pump:	
Name	Resin Transfer Pump
Quantity	1
Type	Centrifugal
Duty	Intermittent
Pump Data:	
Fluid	D ₂ O
Flow Rate	25 gpm
Operating Temperature	Ambient
Head	210 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	3 Phase, Squirrel Cage Induction Motor
Construction:	
Materials in Contact with D ₂ O	Austenitic Sainless Steel, Type 304
Applicable Code	ASME Section III, Class 3
Tanks:	
Name	D ₂ O and H ₂ O Head Tanks
Quantity	2 (1 for each service)
Type	Vertical Cylinder with Dished Heads
Capacity	10.9 ft ³

TABLE 5.1.7-2 (Cont.)

<u>Item</u>	<u>Value</u>
Design Conditions:	
Operating Pressure	Atmospheric
Operating Temperature	100°F
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Materials	Austenitic Stainless Steel, Type 304
Applicable Code	ASME Section III, Class 3
Name:	Resin Deuteration and De-deuteration Tanks
Quantity	2 (1 for each service)
Type	Vertical cylinder with dished top and conical bottom
Capacity	9 ft ³
Design Conditions:	
Operating Pressure	0 to 150 psig
Operating Temperature	Ambient
Design Pressure	175 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel, Type 304
Applicable Code	ASME Section III, Class 3

5.1.7.9 Moderator D₂O Sampling System

Functions

The Moderator D₂O Sampling System is used to collect samples of heavy water from the system.

Design Requirements

Samples of heavy water are obtained by hypodermic sampling from the moderator system, moderator D₂O collection system, moderator purification and cleanup systems. Samples may also be taken from the poison mixing tanks of the liquid poison system. Design parameters are given on Table 5.1.7-3.

This is not an engineered safeguards system.

System Description

The Moderator D₂O Sampling System consists of a series of hypodermic sampling points in sample lines from various points from the moderator purification system, D₂O cleanup system, moderator D₂O collection system, and from the main moderator system. Each hypodermic sampling point is provided with a series of valves to control the flow from the particular point from which a sample of D₂O is desired. Conductivity cells are provided in the sampling loops to check the D₂O from the main moderator system and from the moderator purification system.

TABLE 5.1.7-3
MODERATOR D₂O SAMPLING SYSTEM
DESIGN PARAMETERS

<u>Item</u>	<u>Value</u>
General:	
Type of Sample	Hypodermic
Points Sampled	<ul style="list-style-type: none"> ° Main moderator system ° Purification system return ° Individual ion exchanger outlets ° Moderator D₂O collection system ° D₂O cleanup system
Sample Pots:	
Number	6
Type	Hypodermic
Material:	
Body	Austenitic Stainless Steel
Septum	Neoprene
Conductivity Cell Holders:	
Number	2
Purpose	To contain in-line conductivity cells for continuous monitoring of moderator conductivity.
Conductivity Range	0 to 5 microhm/cm
Applicable Code	ASME Section III, Class 3

5.1.7.10 Moderator Cooling System

Functions

The Moderator Cooling System is an ammonia absorption refrigeration system provided to remove the heat generated in the moderator fluid.

Design Requirements

The Moderator Cooling System is designed to cool 27,400 gpm of moderator fluid (D_2O) from 200°F to 150°F. A flow diagram is given on Figure 5.1.7-4 and system data is presented in Table 5.1.7-4. The total heat absorption load on the refrigeration system is 748.2×10^6 Btu/hr. A description of the Main Moderator system is found in Section 5.1.1.2.

The system is a non-safety related system but is designed to seismic Category I requirements.

Storage tanks provided for storing the system volume of ammonia-water mixture and for the anhydrous ammonia are located outside of the building.

System Description

An ammonia absorption refrigeration system is provided to remove heat from the moderator fluid to preclude pressuring the calandria.

The absorption system is used in lieu of a mechanical refrigeration system to reduce operating costs, eliminate possibility of contaminating the refrigeration system with lubricating oil, and to reduce maintenance.

The compressor function of a mechanical refrigeration system is replaced by two heat exchangers operating at different pressure and temperature levels. In one of the heat exchangers, the GENERATOR, a solution of water and ammonia (the strong liquor) is heated with process steam causing part of the contained ammonia to be driven off as a superheated vapor mixed with dry saturated steam. The steam is then condensed in the second heat exchanger, the DEHYDRATOR, and returned to the generator. The ammonia vapor, in a dry saturated state, is delivered to a CONDENSER. The condenser is furnished with cooling water from the main mechanical draft cooling towers. Further cooling of the ammonia is accomplished in a liquid suction heat exchanger (LSHE) such that the ammonia liquid is subcooled about 10°F. The subcooled liquid ammonia flows through a combination of float and expansion valve where it expands at constant enthalpy into a flooded evaporator. The liquid ammonia in the evaporator is maintained at a constant level where it absorbs heat from the moderator fluid, a subcooled nucleate boiling process, cooling it from 200°F to 150°F. The dry saturated ammonia vapor from the evaporator passes into the shell side of the LSHE, absorbs additional heat from the liquid ammonia heat exchanger section, then flows to the absorber as a superheated ammonia vapor.

The superheated ammonia vapor from the LSHE is mixed in the ABSORBER with the lower concentration ammonia-water solution (weak liquor) formed when the ammonia vapor is driven off in the generator. By removing heat in the absorber and transferring it to cooling water, the absorber causes the vapor to go into solution with the weak liquor and produce a more highly concentrated solution.

For economy of both process steam in the generator and cooling water in the absorber, it is customary to install a heat exchanger (SL/WL HE) in which the weak liquor transfers a portion of the

heat to the strong liquor. A STRONG LIQUOR PUMP is usually installed between the absorber and this heat exchanger, sized with sufficient pressure rise capability to allow the SL/WL HE generator and absorber to operate under optimum conditions. This pump requires approximately 12% of the power needed to operate a conventional vapor compression mechanical refrigeration system.

The system consists of one generator and one dehydrator then splits into two 50% capacity parallel lines. These parallel lines each contain a condenser, receiver, liquid suction heat exchanger, flooded evaporator, absorber, strong liquor pump, and an intercooler. The two parallel lines then feed into the common generator.

A spherical pressure vessel containing liquid anhydrous ammonia is located in the yard in the vicinity of the cooling towers. A horizontal storage tank is provided and located in the yard near the penetrations building. This storage tank is sized to permit draining the complete inventory of ammonia-water mixture from the refrigeration system.

The components of the refrigeration system are constructed to the requirements of the ASME B&PV Code, Section VIII, Division 1. Components are fabricated from carbon steel which meets impact test requirements when used for low temperature services. The tube side of the two evaporators, through which the moderator fluid (D_2O) flows, is fabricated from austenitic stainless steel. The components and piping of the refrigeration system located in seismic Category I structures are designed so as not to fail their pressure boundaries or supports.

A ventilation system is provided in structures containing components of the refrigeration system to assure that ammonia vapors cannot accumulate in the event of a system leak.

All electrical equipment within the confines of the moderator cooling system equipment is furnished with explosion-proof enclosures.

TABLE 5.1.7-4
MODERATOR COOLING SYSTEM DATA

<u>Item</u>	<u>Data</u>
<u>General</u>	
Moderator Fluid	
Fluid	D ₂ O
Flow Rate	28,000 gpm, total
Temperature in,	200°F
Temperature out,	150°F
Heat Transferred	748.2 x 10 ⁶ Btu/hr
Coolant Fluid	
Fluid	NH ₃ and NH ₃ -H ₂ O solution
<u>Heat Exchangers</u>	
Evaporator	
Quantity	Two-50 percent
Type	Horizontal Shell and Tube, Double Tube Sheet
Duty	Continuous
Heat Transferred	374.1 x 10 ⁶ Btu/hr, each
Tube Side Data:	
Fluid	D ₂ O
Flow Rate	14,100 gpm, each
Inlet Temperature	200°F
Outlet Temperature	150°F
Design Pressure	300 psig
Design Temperature	250°F
Shell Side Data:	
Fluid	NH ₃ Liquid
Flow Rate	8.05 x 10 ⁵ lbs/hr each
Temperature	32.6°F
Design Pressure	300 psig
Design Temperature	(-) 20°F

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
Construction:	
Tube Material	Type 304 Stainless Steel
Shell Material	Carbon Steel-Impact Tested
Applicable Codes:	
Tube Side	ASME Section III, Class 3
Shell Side	ASME Section III, Class 3
<u>Absorber</u>	
Quantity	Two-50 percent
Type	Horizontal Shell and Tube
Duty	Continuous
Heat Transferred	$312. \times 10^6$ Btu/hr each
Tube Side Data:	
Fluid	H ₂ O
Flow Rate	21,500 GPM each
Inlet Temperature	97°F
Outlet Temperature	126°F
Design Pressure	300 psig
Design Temperature	450°F
Shell Side Data:	
Fluid	NH ₃ -H ₂ O Solution
Flow Rate	6.67×10^6 lbs/hr each
Inlet Temperature	146.3°F
Outlet Temperature	134°F
Design Pressure	300 psig
Design Temperature	450°F
Construction:	
Tube Material	Carbon Steel
Shell Material	Carbon Steel

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
Applicable Codes:	
Tube Side	ASME Section VIII
Shell Side	ASME Section VIII
<u>Generator</u>	
Quantity	One-100 percent
Type	Horizontal-Coil in Shell
Duty	Continuous
Heat Transferred	750×10^6 Btu/hr
Tube Side Data:	
Fluid	Steam
Flow Rate	8.64×10^5 lbs/hr
Inlet Temperature	300°F
Design Pressure	300 psig
Design Temperature	450°F
Shell Side Data:	
Fluid	NH ₃ -H ₂ O Solution
Flow Rate	
From Dehydrator	1,671,400 lbs/hr
To Dehydrator	1,633,800 lbs/hr
To Interchanger	37,600 lbs/hr
Temperature	
Design Pressure	300 psig
Design Temperature	450°F
Construction:	
Tube Material	Carbon Steel
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section VIII —
Shell Side	ASME Section VIII

TABLE 5.1.7-4 (Cont.)

Item

Data

Dehydrator

The dehydrator is a multiple effect shell and tube unit for producing anhydrous ammonia vapors from a mixture of ammonia and water vapor.

Quantity One-100 percent

Strong Liquor-Vapor

Mixture Heat Exchanger Section

Duty Continuous
Heat Transferred 62.33×10^6 Btu/hr

Tube Side Data:

Fluid $\text{NH}_3\text{-H}_2\text{O}$ Solution
Flow Rate 1,671,400 lbs/hr
Design Pressure 300 psig
Design Temperature 450°F

Shell Side Data:

Fluid $\text{NH}_3\text{-H}_2\text{O}$ Vapor Mixture
Flow Rate 1,633,800 lbs/hr
Design Pressure 300 psig
Design Temperature 450°F

Construction:

Tube Material Carbon Steel
Shell Material Carbon Steel

Applicable Codes:

Tube Side ASME Section VIII
Shell Side ASME Section VIII

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
<u>Water Vapor Mixture</u>	
<u>Heat Exchanger Section</u>	
Duty	Continuous
Heat Transferred	177.5×10^6 Btu/hr
Tube Side Data:	
Fluid	Water
Design Pressure	100 psig
Design Temperature	450°F
Shell Side Data:	
Fluid	NH ₃ -H ₂ O Vapor Mixture
Flow Rate	1.635×10^6 lbs/hr
Design Pressure	300 psig
Design Temperature	450°F
Construction:	
Tube Material	Carbon Steel
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section VIII
Shell Side	ASME Section VIII
<u>Ammonia-Vapor Mixture</u>	
<u>Heat Exchanger Section</u>	
Duty	Continuous
Heat Transferred	53.87×10^6 Btu/hr
Tube Side Data:	
Fluid	NH ₃ Liquid
Flow Rate	37,175 lbs/hr
Inlet Temperature	32.6°F
Outlet Temperature	32.6°F

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
Design Pressure	300 psig
Design Temperature	(-) 20°F
Shell Side Data:	
Fluid	NH ₃ -H ₂ O Vapor Mixture
Flow Rate	1.53 x 10 ⁶ lbs/hr
Design Pressure	300 psig
Design Temperature	450°F
Construction:	
Tube Material	Carbon Steel
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section VIII
Shell Side	ASME Section VIII
<u>Condenser</u>	
Quantity	Two-50 percent
Type	Horizontal Shell and Tube
Duty	Continuous
Heat Transferred	457.5 x 10 ⁶ Btu/hr
Tube Side Data:	
Fluid	H ₂ O
Design Pressure	100 psig
Design Temperature	450°F
Shell Side Data:	
Fluid	NH ₃
Flow Rate	1.457 x 10 ⁶ lbs/hr
Design Pressure	300 psig
Design Temperature	450°F

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
Construction:	
Tube Material	Carbon Steel
Shell Material	Carbon Steel
Applicable Codes:	
Tube Side	ASME Section VIII
Shell Side	ASME Section VIII
<u>Liquid Suction Heat Exchanger</u>	
Quantity	Two
Type	Vertical-Coil in Shell
Duty	Continuous
Heat Transferred	8.59×10^6 Btu/hr each
Tube Side Data:	
Fluid	NH ₃ Liquid
Flow Rate	728,500 lbs/hr each
Inlet Temperature	109.5°F
Outlet Temperature	99.5°F
Design Pressure	300 psig
Design Temperature	(-) 20°F
Shell Side Data:	
Fluid	NH ₃ Liquid and Vapor
Flow Rate	--
Inlet Temperature	32.6°F
Outlet Temperature-Liquid	32.6°F
-Vapor	49.8°F
Design Pressure	300 psig
Design Temperature	(-) 20°F
Construction:	
Tube Material	Carbon Steel
Shell Material	Carbon Steel

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
Applicable Code:	
Tube Side	ASME Section VIII
Shell Side	ASME Section VIII

Pressure Vessels and Tanks

Liquid Ammonia Receivers

Quantity	Two-50 percent
Volume	3800 Gals. each
Type	Horizontal
Supports	Saddles
Design Pressure	300 psig
Design Temperature	450°F
Material	Carbon Steel

Liquid Ammonia Storage Tank

Quantity	One
Size	30 ft. Diameter
Type	Spherical Pressure Vessel
Supports	Pipe Columns
Design Pressure	300 psig
Design Temperature	450°F
Material	Carbon Steel

Weak Liquor Storage Tank

Quantity	One
Volume	125,000 Gallons
Type	Horizontal
Supports	Saddles

TABLE 5.1.7-4 (Cont.)

<u>Item</u>	<u>Data</u>
Design Pressure	150 psig
Design Temperature	450°F
Material	Carbon Steel
<u>Pumps</u>	
<u>Strong Liquor Pump</u>	
Quantity	Two-50 percent
Type	Centrifugal
Duty	Continuous
Pump Data:	
Fluid	NH ₃ -H ₂ O Liquid
Flow Rate	1385 gpm, each
Temperature	126°F
Head	475 feet
Design Pressure	300 psig
Design Temperature	450°F
Motor Data:	
Type	TEFC-Explosion Proof
Horse Power	250 HP, each
Construction:	
Materials in Contact with Fluid	Carbon Steel
Applicable Code	Commercial Standards

5.1.7.11 D₂O Supply System

Functions

The D₂O Supply System is designed to receive and store D₂O from drums and tank trucks and to pump it to the moderator or reactor coolant system during initial filling. In addition, the four storage tanks are capable of containing the inventory of either the moderator system or the heat transport system in the event that draining for maintenance is required. The tanks can also store high isotopic D₂O during normal reactor operation.

Design Requirements

The D₂O Supply System is designed to seismic Category I requirements due to the economic value of the D₂O contained in the system. Pressure containing components of the system are constructed to ASME B&PV Code, Section VIII.

Storage of the D₂O is provided with four horizontal tanks, each having a capacity of approximately 30,000 gallons. Pumps and piping are provided for transferring the D₂O to either the moderator or heat transport system.

System Description

The D₂O Supply System equipment and valves are located in the Reactor Service and Fuel Handling Building.

Four horizontal storage tanks, each having a capacity of 30,000 gallons, and two pumps are provided in the system. Either one or both of the pumps may be used to transfer D₂O to either the moderator system or to the heat transport system. One pump is used to establish recirculation flow before samples are taken.

through the hypodermic sample station and to supply D_2O to the moderator deuteration-de-deuteration-system. The pumps are centrifugal type with integral canned motors.

Fresh D_2O is added to the storage tanks through the D_2O addition station from drums or from tank trucks by connecting a flexible hose to the tank inlet header. D_2O from the calandria can be transferred to the storage tanks by gravity or by using the moderator pumps. Similarly, the transfer of D_2O from the heat transport storage tank is made by gravity. Upgraded D_2O from the D_2O upgrading tower may be transferred to the storage tanks as well. It is important to prevent mixing of the moderator and heat transport heavy water because of the difference in tritium concentration.

TABLE 5.1.7-5
D₂O SUPPLY SYSTEM DESIGN DATA

<u>Item</u>	<u>Data</u>
D ₂ O Supply System	
Storage Capacity	120,000 gallons
D ₂ O Supply Pumps	
Quantity	2
Type	Canned Centrifugal
Duty	Intermittent
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	125 gpm
Temperature	Ambient
Head	100 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	3 Phase, squirrel cage induction motor
Construction:	
Materials in Contact with D ₂ O	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
D ₂ O Storage Tanks	
Quantity	4
Type	Vertical cylinder with dished ends
Capacity, each	30,000 gallons
Design Conditions:	
Operating Pressure	Atmospheric
Operating Temperature	Ambient
Design Pressure	15 psig
Design Temperature	150°F

TABLE 5.1.7-5 (Cont.)

Item

Data

Construction:

Material

Applicable Code

Austenitic Stainless Steel

ASME Section VIII

5.1.7.12 D₂O Vapor Recovery System

Functions

Three separate vapor recovery systems are provided in the reactor building to maintain a dry atmosphere in areas subject to D₂O leakage during operation or shutdown maintenance.

Design Requirements

The basic requirements of these three systems are:

- ° The recovery of D₂O leakage within the dried areas.
- ° The reduction of tritium activity within the dried areas to minimize the contamination spread and internal dose rates to station personnel in adjacent accessible areas.
- ° The maintenance of D₂O areas requiring frequent personnel entry as non-suited areas under normal operating circumstances to reduce maintenance times and minimize exposure to gamma fields.
- ° Minimize downgrading caused by air or light-water leakage into D₂O areas.
- ° The reduction of D₂O losses and activity release rates due to purging of D₂O areas during shutdown or routine maintenance.

System Description

The three separate vapor recovery systems which consist of dryers, duct-work, D₂O collection tanks, transfer pumps and piping are

located in the Reactor Service and Fuel Handling Building at grade elevation.

Recovery of absorbed water vapor occurs in the reactivation condensers located above the dryer vessels. The resultant water is pumped to one of a series of collection tanks located in the D_2O management - upgrading area. Recovered D_2O is segregated according to degree of downgrading and tritium activity.

Reactor building areas are segregated with three separate D_2O vapor recovery systems as follows:

1. Major areas subject to Reactor Coolant System D_2O leakage which are accessible only during reactor shutdown. Included in this system are:
 - Steam generator area, including shutdown cooler area.
 - Fueling machine operating areas.
 - Moderator room (exclusive of enclosure around equipment).

Design conditions for this system are $0^\circ F$ dew-point at a total leakage rate for all areas of 31 lb/hr ($D_2O + H_2O$). Three drier units are provided for this system each rated for 4000 scfm of an absorption air flow, with a separate 500 scfm rated pressure balance exhaust drier to maintain conditions as outlined above. A continuous negative pressure balance of the dried area must be maintained with respect to the adjacent accessible areas to prevent tritium from spreading to the latter areas.

2. Areas requiring frequent access including:

- Fueling machine maintenance locks.
- Fueling machine auxiliary equipment rooms.
- RCS and moderator instrumentation rooms.
- Monitoring rooms.

One 4000 scfm rated drier unit is provided for these areas. It is designed to maintain a room dew-point of -30°F at a total recovery rate of 2.2 lb/hr of 10% isotopic D_2O .

3. Area inside the enclosure around moderator equipment.

One 2000 scfm rated drier unit is provided for this area with a capability to recover 5.5 lb/hr of heavy water at 0°F dew-point. The moderator equipment enclosure is maintained at a negative pressure balance with respect to the areas included in system (1) or (2) as D_2O leakage from the moderator system equipment has a higher tritium activity than D_2O collected in the other two areas.

The driers, heat exchanger housing, and air-handling equipment are all constructed to full containment pressure according to the requirements for ASME B&PV Code, Section III, Class MC. Ductwork and dampers located outside containment are constructed to ASME B&PV Code, Section III, Class 2 for the same design conditions.

Dew point and temperature measurements are provided at the inlet and outlet of all driers. The driers are operated from local field control panels supplied with each drier. Remote status indicators and annunciation are provided in the control room for each drier. Drier malfunction annunciation is provided both locally and in the control room.

Indication of the D_2O Collection Tank Level and high level alarm are also provided in the control room.

Since the three D_2O vapor recovery systems are extensions of the containment envelope, two pneumatically-operated dampers in series are used to isolate each system on high pressure or high activity in the containment. The drain valves of the D_2O collection

tanks form the boundary of the extended containment system. The two pneumatically-operated drain valves in series are normally closed. During draining of the tanks, a high pressure or high activity signal automatically closes the valves.

The negative pressure in the Reactor Building is maintained by extracting air during the vapor recovery cycle. The air is double-dried before being discharged to the action exhaust system. Control of the net exhaust is made with a damper controlled from a manual control station in the control room.

TABLE 5.1.7-6
D₂O VAPOR RECOVERY SYSTEM DESIGN DATA

<u>Item</u>	<u>Data</u>
<u>D₂O Vapor Recovery System</u>	
Type	Regenerating Air Dryers
D ₂ O Recovery Capacity	55 lb/hr at 0°F Dew Point
Applicable Code	ASME Section III, Class 2
<u>D₂O Vapor Recovery System Condensers</u>	
Quantity	5
Type	Enclosed fin-tube (air to water)
Heat Transferred	90 KW (th)
Tube Side Data:	
Fluid	Chilled H ₂ O
Flow Rate	57.5 gpm
Inlet Temperature	50°F
Outlet Temperature	69°F
Design Pressure	200 psig
Design Temperature	220°F
Shell Side Data:	
Fluid	Air
Flow Rate	1400 ft ³ /hr
Inlet Temperature	300°F
Outlet Temperature	85°F
Design Pressure	20 psig
Design Temperature	350°F
Construction:	
Tube Material	Cu Ni
Tube Outside Diameter	0.625 inch
Shell Material	Austenitic Stainless Steel
Applicable Codes:	
Tube Side	
Shell Side	ASME Section III, Class 2

TABLE 5.1.7-6 (Cont.)

<u>Item</u>	<u>Data</u>
<u>D₂O Vapor Recovery Collection Pumps</u>	
Quantity	2
Type	Canned Centrifugal
Pump Data:	
Fluid	Downgraded D ₂ O
Flow Rate	25 gpm
Temperature	100°F
Head	50 ft
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Materials in Contact with Fluid	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
Motor Data:	
Type	3 phase, squirrel cage induction motor
<u>D₂O Vapor Recovery Fans</u>	
Quantity	12
Type	Centrifugal
Fan Data:	
Fluid	Air
Flow Rate	4 at 3500 ft ³ /min 4 at 1100 ft ³ /min 4 at 175 ft ³ /min 1 at 440 ft ³ /min 1 at 550 ft ³ /min
Temperature	Ambient
Design Pressure	20 psig
Design Temperature	150°F

TABLE 5.1.7-6 (Cont.)

<u>Item</u>	<u>Data</u>
Applicable Code:	Seismic Category I Equivalent
Motor Data:	
Type	3 phase, squirrel cage induction motor
<u>Vapor Recovery System D₂O Collection Tanks</u>	
Quantity	3
Type	Vertical cylinder with ASME Flanged & Dished Heads
Design Conditions:	
Operating Press	Atmospheric
Operating Temperature	100°F
Design Pressure	20 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 2
<u>D₂O Vapor Recovery Driers</u>	
Quantity	6
Type	Molecular sieve dessicant
D ₂ O Recovery Capacity	14 lb/hr at 0°F Dew Point
Design Pressure	20 psig
Design Temperature	300°F
Applicable Code	ASME Section III, Class 2

5.1.7.13 D₂O Cleanup System

Functions

The D₂O cleanup system removes dissolved, particulate, and organic impurities from recovered D₂O and produces a product suitable for processing in the D₂O Upgrading System.

Design Requirements

The feed to the D₂O cleanup system consists of a mixture of D₂O and H₂O. The isotopic varies from more than 2 percent D₂O to less than 99.8 percent, with an average of 62 percent D₂O.

The system consists of downgraded D₂O storage tanks, charcoal filter, ion exchange columns, product tank, and necessary pumps. The system is designed to handle approximately 1,000,000 pounds of D₂O per year.

System Description

The water to be cleaned is pumped from the downgraded D₂O storage tanks through the D₂O cleanup system to the product tanks. The system is provided with one carbon filter which removes particulates and organic material and three ion exchange columns in series which remove dissolved ionic impurities.

The charcoal and ion exchange resin are added to the vessels manually. Removal of spent resin is by slurring with high pressure demineralized water. Shielding is provided around the charcoal filter and ion exchange columns. The cleanup system is fabricated of stainless steel and constructed to ASME B&PV Code, Section VIII.

TABLE 5.1.7-7
D₂O CLEANUP SYSTEM DESIGN DATA

<u>Item</u>	<u>Data</u>
<u>D₂O Cleanup System</u>	
Feed Fluid	Recovered Downgraded D ₂ O
Feed Storage Capacity	15,000 gallons
Product Storage Capacity	1500 gallons
Cleanup Process	
Duty	Intermittent
Type	Ion Exchange
Flow Rate	10 gpm
Applicable Code	ASME Section VIII
<u>D₂O Cleanup Pumps</u>	
Quantity	3 (1 of which is spare)
Type	Canned Centrifugal
Duty	Intermittent
Capacity	100%
Pump Data:	
Fluid	D ₂ O
Flow Rate	10 gpm
Temperature	Ambient
Head	30 ft.
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Materials in Contact with Fluid	Austenitic Stainless Steel
Applicable Code	ASME Section VIII
Motor Data:	
Type	3 Phase, squirrel cage induction

TABLE 5.1.7-7 (Cont.)

<u>Item</u>	<u>Data</u>
<u>D₂O Cleanup Storage Tanks</u>	
Quantity	6
Type	Vertical cylinder with dished heads
Capacity	
Design Conditions:	
Operating Pressure	0 psig
Operating Temperature	Ambient
Design Pressure	15 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section VIII
<u>D₂O Cleanup Charcoal Filter</u>	
Quantity	1
Type	Vertical cylinder with dished top and conical bottom
Volume	75 gallons
Charcoal Capacity	10 cubic feet
Design Conditions:	
Flow Rate	10 gpm
Direction of Flow	Downward
Operating Pressure	15 psig
Operating Temperature	Ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

TABLE 5.1.7-7 (Cont.)

<u>Item</u>	<u>Data</u>
<u>D₂O Cleanup Ion Exchange Columns</u>	
Quantity	4
Type	Vertical cylinder with dished top and conical bottom
Volume	75 gallons
Resin Capacity	10 cubic feet
Design Conditions:	
Flow Rate	10 gpm
Direction of Flow	Downward
Operating Pressure	15 psig
Operating Temperature	Ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

5.1.7.14 D₂O Upgrading System

Functions

The D₂O Upgrading System separates solutions of D₂O and H₂O by distillation. The head product, or discard stream, is limited to contain no more than 0.1 percent D₂O, while the bottom product stream contains 99.8 percent D₂O.

Design Requirements

The feed to the upgrading system consists of a mixture of D₂O and H₂O. The isotopic varies from more than 2 percent D₂O to less than 99.8 percent, with an average of 62 percent D₂O.

The system consists of a steam heated feed evaporator, a process column, reboiler, reboiler recirculating pump, head condenser, and a cold trap. System process conditions are maintained at 120 mm Hg corresponding to a saturation temperature of D₂O of approximately 133°F. The system is designed for 1,000,000 pounds D₂O upgrading per year.

System Description

The primary component of the D₂O upgrading system is a rectification tower. This tower consists of a number of flanged sections and a flanged sump section mounted on a support skirt. Each of the flanged tower sections contains a metallic gauze packing and support which promotes intimate contact between the falling liquid and rising vapor cascade. The sections are provided with a reflux distributor, reflux collector, two sight glasses, manhole, and sample valve.

Within the column a steady and continuous reflux of liquid and vapor passes through the packed sections of the column. The liquid falls slowly downward through the packing, concurrent to the upward flow of vapor. When the falling liquid reaches the bottom of the column, it is collected in a sump, then re-evaporated and returned to the base of the column as a vapor. The vapor reaching the top of the column is condensed and returns to the bottom as a liquid.

The basic physical mechanism causing separation is the relative volatility of H_2O and D_2O . If the relative volatility of the mixture is increased, the desired separation can be obtained using a short column. The relative volatility of H_2O and D_2O increases with decreasing temperature. Considering these factors the tower is operated at a pressure of 120 mm Hg and a corresponding saturation temperature of approximately 133°F.

The product held in the bottom product tank is drained by gravity to the D_2O supply and inventory system or to drums.

The head product, as liquid, flows by gravity from the head condenser to the control panel, where its isotopic is measured. If the isotopic is above the setpoint, an alarm is actuated and the column will be automatically switched to total reflux operation.

The head product tank operates at atmospheric pressure and is vented to an exhaust cooler. The product held in the head product tank flows to the liquid waste handling system or to D_2O drums.

The control panel provides an automatic in-line isotopic analyzing function for the bottom and head product. This unit controls the concentration of both products in that it prevents product takeoff at the normal rate if the bottom product D_2O concentration is below or if the head product D_2O concentration is above the

setpoint. An infrared isotopic analyzing device performs the analyzing functions within the control panel.

A vacuum system is connected to the upgrading tower and associated equipment, controlling the system operating pressure to approximately 120 mm Hg. The remainder of the system operates at higher pressures and the head takeoff system operates at atmospheric pressure.

A cold trap is provided at the head of the column to recover D_2O which might be lost through the vacuum unit. To minimize the mixing of low tritium D_2O from the heat transport system and high tritium D_2O from the moderator system, it is necessary to drain the tower periodically and to separately upgrade D_2O from these two sources.

5.1.7.15 Compressed Air System

Functions

The compressed air system is designed to provide a continuous supply of oil-free air for pneumatic instruments and controls and general plant service. Dry air is supplied for instrumentation and controls, and undried air is supplied for various plant functions.

Design Requirements

The system contains three air compressors each rated at 350 CFM and a 100 psig discharge pressure. Air receivers are constructed in accordance with the ASME B&PV Code, Section VIII, for 125 psig design pressure and are equipped with safety valves set at 125 psig.

Instrument and service air piping is designed in accordance with ANSI B31.1 Code for Power Piping.

System Description

The compressed air system includes three electric motor-driven air compressors rated at 350 CFM and 100 psig discharge pressure. The compressors are connected in parallel, each with an intake filter-silencer, after-cooler, and moisture separator. Each compressor unit is capable of supplying air at 100 psig at a flow rate sufficient to meet normal plant requirements. The compressors discharge to two vertical air receivers.

Both receivers supply air to the service air and instrument air systems. The service air header contains a back-pressure regulating valve which closes when the upstream header pressure is reduced to 75 psig. This prevents the service air system from bleeding down the instrument air supply.

The compressed air for the instrument air system is supplied through two 100-percent capacity parallel headers, each containing:

- ° Twin tower air dryer for minus 40 F dew point, with regeneration cycle for reactivating the wet dessiccant.
- ° Pre- and after-filters for removal of moisture droplets, particles of dirt, rust, and scale.

Each compressor has an automatic dual control. Normal control is with one air compressor operating at constant speed under load-unload control. The second compressor is on standby and starts automatically on heavy demand or low instrument system pressure. The third compressor is used as a spare.

Compressed air for instrumentation and controls within the containment is supplied by two-nonlubricated compressors nominally rated at 100 CFM, discharging through filters and air driers to provide clean dry air. Service air to the containment air stations is supplied from the unit service air system for containment air leakage monitoring and other services. Instrument air from outside the containment can be connected to the containment instrument air system by hose connection in an emergency.

The station air compressors are powered from 480 volt emergency buses. Two compressors are powered from one bus and the third from a second bus. Upon loss of off-site power, the compressors may be manually started as emergency diesel generator capacity becomes available.

Air-operated valves and controls throughout the plant are arranged for safe failure with loss of air supply. Conventional instrumentation is provided for the compressed air system, including pressure indication in the main control room.

5.1.7.16 Air Conditioning, Heating, Cooling, and Ventilating Systems

Functions

The heating, ventilation, and air-conditioning systems are provided for all buildings of the plant for both normal and abnormal plant operations.

Design Requirements

The outside ambient air conditions are:

Summer design temperature: 88°F, DB
 75°F, WB
 Winter design temperature: 0°F
 Concident wind velocity: Moderate to 12 MPH

the inside design air conditions are:

<u>Area</u>	<u>Summer, Max°F</u>	<u>Winter, Min°F</u>
Control room complex	75	70
Cable spreading	104	55
4 kV switchgear	104	55
Battery rooms	95	65
Service building (occupied areas)	75	70
Service building (equipment area)	104	50
Turbine Building	104	55
Fuel storage room	104	50
Containment	120	50
Circulating water pump hose	104	50
Service water pump house	104	50
Penetrations building	104	50
Non-essential switchgear area	104	50
Diesel generator room	104	50

The HVAC systems are not safety related, except for the Control Room Habitability Systems, but the supports of all components in safety-class structures are designed in accordance with seismic Category I requirements. Separate system design descriptions contain the requirements for the control room habitability systems.

System Description

Heating for the buildings is supplied by steam from the auxiliary boiler located in the administration building.

Containment

During normal reactor operation sufficient heat is lost from the nuclear reactor system to offset building heat-loss during the winter months. Heat is supplied to the containment when the reactor is shutdown through steam-air heat exchange coils and through the air circulation system. In the summer months, cooling is supplied to the air handling system from the chilled water system.

Normally, the air in the steam generator and fueling machine room areas are not ventilated. The air within these areas is normally recirculated in a separate system to recover heavy water vapor and limit the tritium level in the containment air. The containment is maintained at a slight negative pressure, with respect to the outside environs, and the general direction of air flow is from the areas of low-to-high levels of probable contamination.

Air intake and exhaust is described in the system design description for the containment air purification and cleanup system.

Reactor Service Building

A one-way air conditioning system is installed in the reactor service building without any air recycling to provide a suitable temperature and humidity in the various rooms by continuous treatment of fresh air.

The fresh treated air follows a forced irreversible flow from zones with a low radioactive contamination to areas having a higher radioactive contamination, without the possibility of any air recycling.

Air is drawn into the building, filtered, and heated or cooled as required. Before being discharged from the building, the air is monitored for activity, then discharged through the plant stack filtration system.

The hot water heating system with heating coils in the ventilation inlet ducts is designed to maintain an indoor temperature of 70 F in winter when the ventilation system is running.

An air conditioning system for the Control Room is provided to ensure suitable temperatures and humidity in the various rooms as well as an adequate renewal of ventilated fresh air.

Turbine Building

A cooling system is provided in the Turbine Building which, in conjunction with heaters, will maintain the building temperature between 80°F and 90°F with a maximum temperature of 104°F in the areas where considerable heat rises occur. These include the condenser area, the feedwater heater area and the deaerator area. Cooling is provided from coils and fans with cooling water supplied from the recirculated water system.

In the winter time, ventilation air is drawn from outside the building, mixed with recirculated air inside the building and circulated as required. Ventilators are provided at the hot pieces of equipment to expel the hot air, and special provisions are provided to ventilate the areas where noxious gases may accumulate, including the chemical storage areas.

5.1.7.17 Fire Protection System

Functions

The fire protection system is provided to minimize the probability and effect of a fire. It is provided so the potential consequence of such an event, should one occur, does not result in undue risk to the health and safety of the public or of plant personnel.

Design Requirements

Where any appreciable quantities of combustible material exist, fire detection devices and/or fire protection systems are provided. The fire protection systems are evaluated and designed to assure that their operation or failure does not produce any unsafe condition nor damage any safety related equipment.

Physical separation or fire barriers are provided between associated redundant systems or redundant equipment where any combustible material exists. Units such as the emergency diesel-generators are placed in separate three-hour NFPA fire rated enclosures.

The occurrence of fire is minimized by the use of non-combustible materials in the construction of the plant. Construction materials are specified in accordance with NEPIA - Basic Fire Protection for Nuclear Power Plants. Electrical cables are specified with either non-flammable insulation or with self-extinguishing and non-propagating fire characteristics in addition to dielectric strength and other electrical-mechanical properties.

Fire barrier walls are provided between outdoor adjacent transformers; hydrogen manifolds are located out-of-doors; and auxiliary boilers are provided with combustion safety features. Automatic smoke and heat power venting is provided for the turbine building and other supporting building.

The fire protection system for the control room complex and alternate control room is a carbon-dioxide system with remote carbon dioxide bottles and piping to hose reel stations that provide coverage of the control room.

The cable spreading rooms is provided with a fixed carbon dioxide system with remote carbon dioxide bottles and piping that will flood the area in the unlikely event of a fire. A water deluge system is provided as a backup in the event it becomes necessary to resort to its use.

The battery rooms are provided with portable carbon dioxide fire extinguishers that are located outside of the battery rooms.

The fire protection system is designed in accordance with the applicable portions and requirements of the following:

- Nuclear Energy Property Insurance Association
- National Fire Protection Association Codes
- Criteria 3, Appendix A, Title 10 of Code of Federal Regulations, Part 50.
- USNRC Regulatory Guide 1.120 - Fire Protection Guidelines for Nuclear Power Plants.
- USNRC Branch Technical Position ASB 9.5-1

The fire protection system is not designed to seismic Category I requirements, but is designed so that failure of the system will not induce failure of a safety related system or safety related equipment.

System Description

Water Supply

The water supply for the fire protection system is stored in two 500,000 gallon storage tanks of which 300,000 gallons in each tank is reserved for fire protection. Three horizontal centrifugal pumps, one 1500 gpm electric motor-driven and two 1500 gpm diesel engine-driven, are provided and capable of taking suction from either tank. The third pump is redundant and serves as a standby unit. Two 25 gpm jockey pumps, complete with automatic local controls, are provided to maintain system pressure.

The fire pumps supply water to a 12-inch fire main ring header. The fire main ring header is designed as a closed-loop system which surrounds the plant site and permits water flow in either direction. Post indicator isolation valves are installed in the yard piping system to permit partial pipe line isolation without interruption of service to the remainder of the system.

The 500,000 gallon storage tanks at grade are heated and protected from freezing. All piping and valves are buried, electrically traced or installed in a heated area.

Yard Protection System

Fire protection is provided to exterior plant areas by fire hydrants located along the loop at approximately 250 feet intervals. Hose houses, complete with associated accessories, are provided to serve the yard hydrant system to provide coverage for each building.

Fixed Water Spray Systems

Fixed water spray deluge protection is provided for the following types of equipment:

- Generator step-up transformers
- Unit auxiliary transformers
- Reserve auxiliary transformers
- Standby reserve auxiliary transformers
- Station auxiliary transformers
- Hydrogen seal oil units
- Turbine oil reservoirs
- Lube oil conditioner
- Cable tunnels, electrical penetration areas, and cable spreading rooms
- Diesel generator fuel oil piping in floor trenches and day oil tanks

Standpipe Systems

The turbine building, office facility, and service building are provided with a standpipe system in accordance with NFPA Standards.

Fixed Carbon Dioxide System

Manually initiated fixed carbon dioxide fire extinguishing systems are provided in the following areas:

- Cable spreading rooms above and below the control room.

Automatic Sprinklers

The turbine generator areas below the operating floor containing oil lines are protected with an automatically actuated sprinkler system.

Portable Fire Extinguishers

Portable fire extinguishers are located throughout the site for immediate use against small fires.

Fire Detection Devices

The fire detection system consists of the following, and are used as applicable for the service intended:

- ° Ionization type detectors capable of sensing the products of combustion before visible smoke, heat or flame are initiated.
- ° Combination temperature rise and fixed temperature type thermostats.

5.1.7.18 Auxiliary Steam System

Functions & Design Requirements

The auxiliary steam system provides low pressure saturated steam for the nuclear steam supply system and the power conversion system during normal plant startup and shutdown. The auxiliary steam system also satisfies building heating and miscellaneous plant requirements. During plant startup, steam is supplied to the turbine gland sealing system, the steam generator blowdown evaporator, the boron recycle, and waste processing evaporators and various nuclear tanks. Shutdown steam requirements are similar, except that gland sealing does not consume any auxiliary steam.

System Description

The auxiliary steam system consists of two package boilers each rated at 80,000 lbs/hr of saturated steam at 150 psig and complete

with three boiler feed pumps (one spare), a triplex fuel oil pumping set (one spare pump), one common deaerator and storage tank, one common blowdown tank, two chemical feed units, one main fuel oil storage tank and controls for local automatic operation.

The boilers are fired by No. 2 fuel oil supplied directly from the main oil storage tank to the fuel oil pumping set. Steam atomization and air atomization are used for continuous operation and startup respectively.

Feedwater from the deaerator is pumped to the boilers, evaporated, and piped to building heating units and operating equipment. Heating system is returned to the deaerator and the equipment steam is added to the main cycle.

The auxiliary steam system and the main steam system interface at the reheater discharge.

5.1.7.19 Standby Power Supply and Auxiliaries (Diesel-Generator Units)

Functions

The diesel-generators distribute a-c power to the plant safety-related auxiliaries during emergency operating conditions.

Design Requirements

The emergency standby power supply is provided with two redundant diesel engine generator systems of identical design and characteristics to supply onsite power of sufficient capacity and capability to reliably shutdown the reactor during emergency plant operation. The load rating of each diesel generator is 6083 kW at 8760 hours per year (continuous).

The generator is rated at 6700 kW for continuous duty.

System Description

Redundant standby diesel-generator units are provided to supply onsite power to the plant safety-related auxiliaries during emergency operating conditions. The capacity of each diesel-generator is adequate to support operation of engineered safeguards systems loads within the continuous load rating. The loads are determined on the basis that the sum of the estimated loads needed to be powered at any one time is equal to or less than the continuous rating of each unit.

Each diesel-generator system comprises the auxiliaries necessary for quick-start operation, connection to the 4,160 volt emergency bus, and connections to the required services. No auxiliaries are shared between the diesel-generator systems. External power sources, other than d-c control power from the plant's batteries, are not required for engine starting or subsequent system operations.

To ensure rapid starting, each diesel generator set has its own independent starting air supply fed from its own set of redundant accumulators. Each accumulator is sized to give a diesel engine 30 seconds of steady engine cranking or five successive starting cranks. Each accumulator is supplied by its own air compressor. Each diesel generator set is supplied with two air starting systems. Elapsed time from receipt of "start diesel" signal to "ready-to-load" condition is 10 seconds.

The diesel engines are water cooled and obtain the water from the nuclear service water system: Each diesel requires approximately 1050 gallons per minute of water having a maximum inlet temperature of 105°F. To prevent freezing of the water during standby duty, each diesel is provided with a thermostatically controlled electric

immersion heater for the jacket water. The diesel is capable of operating without jacket cooling water for a period of time in excess of that required to start the cooling water pump from the emergency bus.

The redundant diesel-generators are provided with independent fuel oil storage tanks. The tanks are buried and located outside of the control and diesel building and have a capacity to operate each of the diesels for a period of seven days under full-load continuous duty. The fuel oil systems each have a fuel oil pump that supply the engine day storage supply tanks with fuel oil, as required.

The day tank is skid mounted and is sized to provide 3-hours supply of fuel oil for the diesel engine when operating at continuous full load. The day tank is equipped with level switches to provide automatic actuation of the associated fuel oil storage transfer pump, low-low level alarm and high-high level tripping of the pump with alarm. Overflow recirculating piping is provided from the system day tank to the fuel oil storage tank. Fuel oil is transferred from each fuel oil storage tank to its respective day tank by a pump powered from the associated emergency bus. Valving is provided to permit either transfer pump to supply one or both day tanks from either storage tank.

Each diesel-generator unit is located at the lowest level in the control and diesel building. An enclosure wall is designed to withstand limited explosions and provides a barrier against postulated missiles such as crank-case doors or rupture of an air receiver tank. The diesel-generator and its associated auxiliaries are designed to the requirements of seismic Category I.

An independent lubrication system is provided to assure adequate lubrication for the diesel engine moving parts and for lubrication

of the generator. The lube oil system is provided with a strainer, pump, oil cooler, and a strainer before the oil enters the engine or generator. When the engine is on standby duty the lube oil is maintained at the recommended temperature with electric heat. Also, a pump is provided to circulate oil through the engine during standby duty to enable the engine to start with a minimum of friction engine wear.

TABLE 5.1.7-8
EMERGENCY ELECTRIC POWER LOAD DATA

<u>Service</u>	<u>Diesel-Generator "A"</u>	<u>Diesel-Generator "B"</u>
Nuclear Service Water Pumps	800 HP	800 HP
Feed & Bleed (HPI)	500 HP	500 HP
Primary Component Cooling Pump	800 HP	800 HP
Ultimate Heat Sink Fans	250 HP	250 HP
Emergency Feedwater Pumps	900 HP	900 HP
Shutdown Cooling Pumps	400 HP	400 HP

5.1.7.20 Reactor Coolant System D₂O Sampling System

Functions

The D₂O Sampling System is provided to obtain samples of heavy water from various points in the Reactor Coolant System (RCS).

Design Requirements

Components of the system are designed to the pressure and temperature conditions of the RCS.

System Description

One sample cabinet is used for this system. Samples of liquid may be obtained by hypodermic sampling, from the reactor coolant purification system, the gland seal cooling system, the D₂O collection tank, the discharge lines of reactor coolant pumps No. 2 and 4, and reactor coolant inlet lines to Steam Generators No. 2 and 4.

Crud filters with flow integrators are provided for measuring the solid impurities in the RCS. Hypodermic and canister samples can be withdrawn for further analysis. Sample coolers and pressure reducing coils are provided in the lines connecting the sample cabinet with the reactor coolant system. The heavy water exiting the crud filters is returned to the D₂O clean-up system. The sample cabinet is connected to the reactor coolant system feed pump suction lines to close the circuit and return the unused heavy water. Any spills collect in the cabinet which is fitted with a moisture detector.

TABLE 5.1.7-9
RCS D₂O SAMPLING SYSTEM
DESIGN DATA

<u>Item</u>	<u>Value</u>
D ₂ O Sampling Circuit	
Types of Sample	Hypodermic 300 ml cannister crud
Points Sampled	- reactor inlet and outlet, both circuits - purification inlet to filters outlet of each filter, outlet of each ion exchange column - D ₂ O collection system
Design Conditions	
Pressure	2485 psig
Temperature	700°F
Normal Conditions (After cooler)	
Pressure	Up to 2000 psig
Temperature	Less than 100°F
Applicable Code	ASME Section III, Class 3
<u>Sample Pots</u>	
Quantity	6
Type	Hypodermic
Material	
Body	Austenitic Stainless Steel
Septum	Neoprene, Shore hardness 40
<u>Sample Cannisters</u>	
Quantity	2
Type	Removable pressure vessels
Volume	300 ml
Material	Austenitic Stainless Steel

TABLE 5.1.7-9 (Cont.)

<u>Item</u>	<u>Value</u>
<u>Crud Collectors</u>	
Quantity	6
Type	Filtration of suspended solids onto 0.45 micron filter paper
Material	Austenitic Stainless Steel
<u>Pressure Reducing Coils</u>	
Quantity	8
Normal Flow	0.31 gpm
Pressure reduced to	150 psig
<u>Sample Coolers</u>	
Quantity	4
Type	Shell and Tube
Material	Inconel 600 tubes
Heat exchange area	1 ft ² each

5.1.7.21 RCS D₂O Collection System

Functions

The D₂O collection system is provided to collect leakage from mechanical components, and to receive D₂O drained from equipment prior to maintenance. The collected D₂O is transferred to the D₂O storage tank or, if the D₂O is downgraded, to drums for transport to the upgrading column.

System Description

The collection tank is located at a low elevation relative to the reactor coolant equipment, and pumps are provided to transfer D₂O from the collection tank to the storage tank.

The D₂O collection tank collects any water that has leaked from the components of the RCS. Two pumps and a common discharge valve are associated with the tank. Computer logic is arranged to start the pumps and open the valve on high level in the collection tank. The D₂O collected is then pumped back to the storage tank. At low level, the pumps are automatically turned off and the isolating valve closes. Samples of the contents of the tank can be delivered to the Reactor Coolant D₂O Sampling System for isotopic analysis. If the isotopic purity is low then the collection tank contents are pumped to drums for upgrading. Level and temperature of the tank contents can be indicated and alarmed by the unit computer.

The cover gas in the collection tank is helium. The helium pressure is maintained at the atmospheric level by means of a connection to the reactor coolant D₂O storage tank which is at atmospheric pressure.

The D₂O collection system components are constructed from austenitic stainless steel to avoid corrosion.

5.1.7.22 RCS Deuteration and De-Deuteration System

Functions

The Reactor Coolant Deuteration and De-deuteration System is provided to replace light water contained in fresh ion exchange resin with heavy water (deuteration). The system is also used to replace and recover the heavy water from the "spent" ion exchange resin with light water before discarding the resin.

Design Requirements

Deuteration is the replacement of H_2O in the resin by D_2O by a slow upward flow of D_2O through the resin bed. De-deuteration is the replacement of D_2O in the resin by H_2O by a slow downward flow of H_2O through the resin bed in order to recover the maximum quantity of D_2O before discarding the spent resin. Some heavy water is downgraded during both the deuteration and the de-deuteration process. This water will be collected and upgraded for further use.

System Description

The system consists of a deuteration tank, a de-deuteration tank, a canned motor driven pump, a D_2O tank, an H_2O tank, piping, valves, strainers, fittings, and instrumentation.

The resin deuteration and de-deuteration tanks are about 5 feet in diameter and about 6 feet tall and hold approximately 70 ft³ of ion exchange resin. Both tanks have a conical bottom to facilitate resin removal by resin slurry. They are provided with removable 'wedge wire' screens capable of withstanding high differential pressure.

Both the D_2O tank and the H_2O tank have a capacity of about 150 ft^3 each and are provided with multiple position level switches and alarms. The level switches (4 for the D_2O tank and 5 for the H_2O tank) operate the automatic ON-OFF control valves located in the deuteration tank outlets. The pump is used for D_2O service only and has a capacity of about 75 gpm at 100°F. All the strainers in the system as well as the wedge wire screens retain any particles larger than 0.2 mm. The resin bead size is 0.38-0.60 mm.

The valves in the system are manual valves, except for two located in the main lines to the storage tanks in the D_2O cleanup system. These two valves will be ON-OFF automatically operated control valves. Diaphragm valves are used for D_2O service, and ball valves where resin transfer is involved.

The deuteration and de-deuteration processes can be carried on at the same time, independently of each other. When the deuteration and the de-deuteration processes are completed, the deuteration tank will be full of heavy water and the de-deuteration tank full of light water. The H_2O in the de-deuteration tank will be transferred to the resin disposal vault and then the D_2O from the deuteration tank will be transferred to the de-deuteration tank. These two transfers are made using instrument air.

To eliminate inadvertent downgrading of D_2O , flexible connections are provided wherever it is possible to mix D_2O with H_2O , that is, an operator will have to make the connection before D_2O and H_2O are mixed.

The deuteration and de-deuteration system is located in the Reactor Service Building. Only the de-deuteration tank is shielded with five foot thick light concrete. One shielding panel can be removed to allow access to the vessel. All resin slurry lines are shielded with one foot thick concrete.

The make and break connections of the system are located under exhaust hoods to prevent tritium from escaping into the room. The sampling and D_2O collection parts of the system are semi-automatic and physically located in the D_2O management area.

The deuteration and de-deuteration system is provided with the following instrumentation:

Deuteration - local and control room indications of the D_2O level in the D_2O tank and local indications of the specific gravity of downgraded D_2O and of the flow of downgraded D_2O to D_2O clean-up system tanks.

De-deuteration - local and control room indications of H_2O level in the H_2O tank and local indications of the specific gravity of downgraded D_2O and of the flow of downgraded D_2O to D_2O clean-up system tanks.

High levels in the D_2O tank and H_2O tank are both annunciated. Level measurement is interlocked with automatic valve to automatically close it at preset tank levels. These tank levels are related to the isotopic of the downgraded D_2O leaving the fresh resin tank.

5.1.7.23 Containment Air Purification and Cleanup Systems

Functions

The Containment Air Purification and Cleanup System is used to maintain the levels of activity within safe working limits in the containment during normal operation.

Design Requirements

During normal reactor operation the reactor coolant leakage inside the containment can result in a significant buildup of fission products. The level of activity must be maintained within safe working limits for personnel access for inspection and maintenance. This safe level is achieved by periodic filtering and purging of the containment atmosphere to reduce radioactive concentrations and remove the iodine contained therein.

The containment purge system, together with the recirculation filter system, is designed to reduce the containment airborne activity levels below the permissible levels within 16 hours following a reactor shutdown.

These systems are not safety related but are designed to seismic Category I structural requirements.

System Description

The containment system has different systems for air cleaning, each designed to meet a specific mode of operation.

- Cleanup filter system
- Containment purge system
- Containment recirculating filter system
- Containment spray system

The first two systems are designed for normal plant operation. Another containment cleanup system, the containment spray system is used for post accident conditions and its cooling functions are described in Section 5.1.4.1. The air handling portion of the recirculation filter system is required to function for post accident conditions to prevent localized hydrogen accumulation. The filter portion of the system is required for normal operation only.

Cleanup Filter System

This system is designed to ventilate and to filter air from selected potentially radioactive areas within the containment. HEPA and treated charcoal are used for the filter media.

The system is designed to reduce airborne concentrations of particulates and iodine to less than one-percent of inlet concentrations. Redundant exhaust fans and dampers are provided to ensure that at least one fan is available during cleanup operations.

Containment Purge System

This system is provided for periodic purging with filtering for iodine removal. This system in conjunction with the containment recirculation filter system is designed to reduce the containment airborne activity level to below the allowable permitted by Appendix B, Table 1, Column 1, of 10 CFR Part 20 within 16 hours following reactor shutdown. This assumes that no purging or recirculation filtering of containment air is initiated prior to reactor shutdown.

The purge system consists of a purge supply and a purge exhaust system. The purge supply system passes outside air through roughing filters, steam heating coils to temper the air in the winter season, supply air fan and a ducted distribution system.

The purge exhaust system has pre-filters, HEPA filters, and impregnated charcoal filters for removal of iodine. The filtered air is then exhausted to the primary vent stack.

Containment Recirculating Filter System

The containment recirculating filter system is designed to remove iodine from the containment atmosphere during normal plant operations to limit the iodine concentrations and exposure to plant operating personnel during limited access to the containment. The system is also designed to rapidly reduce the iodine concentrations after reactor shutdown to permit personnel access for inspection and maintenance operations. The system also functions to provide mixing of the containment atmosphere (without filtration) following a LOCA to prevent localized hydrogen accumulation.

The components of the air filtration portion of the containment recirculating air filter system are required only during normal plant operations. The remainder of the system is required for air mixing of the containment atmosphere during post-LOCA conditions. All components of the system are designed to seismic Category I requirements and the air handling components, with the exception of the filters, are qualified to perform under LOCA conditions.

TABLE 5.1.7-11
RCS DEUTERATION AND DE-DEUTERATION SYSTEM
DESIGN DATA

<u>Item</u>	<u>Value</u>
Deuteration and De-deuteration System	
Deuteration and De-deuteration Flow Rate	0.625 - 1.25 gpm
D ₂ O required to Deuterate (70 ft ³) resin	103 ft ³
D ₂ O/H ₂ O interface velocity	0.01 to 0.02 ft/min
Applicable Code	ASME Section III, Class 3
<u>Resin Transfer Pump</u>	
Quantity	1
Type	Centrifugal
Duty	Intermittent
Pump Data:	
Fluid	D ₂ O
Flow Rate	75 gpm
Temperature	Ambient
Head	210 ft
Design Pressure	150 psig
Design Temperature	150°F
Motor Data:	
Type	Squirrel cage induction motor
Horsepower	3/4
Construction:	
Materials in contact with fluid	Austenitic Stainless Steel
Applicable Code	ASME, Section III, Class 3
<u>D₂O and H₂O Head Tanks</u>	
Quantity	1 of each
Type	Vertical cylinder, dished ends
Capacity	150 ft ³

TABLE 5.1.7-11 (Cont.)

<u>Item</u>	<u>Value</u>
Design Conditions:	
Operating Pressure	Atmospheric
Operating Temperature	Ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3
<u>Resin Deuteration and De-Deuteration Tanks</u>	
Quantity	1 of each
Type	Vertical cylinder, dished top, conical bottom
Capacity	90 ft ³
Design Conditions:	
Operating Pressure	0-150 psig
Operating Temperature	Ambient
Design Pressure	150 psig
Design Temperature	150°F
Construction:	
Material	Austenitic Stainless Steel
Applicable Code	ASME Section III, Class 3

FIGURE 5.1.7-1
FUEL HANDLING SEQUENCE

5.1-329

FIGURE 5.1.7-2
MODERATOR D₂O COLLECTION
SYSTEM

Vent to Vapor
Recovery

Downgraded D₂O to Upgrading

FIGURE 5.1.7-3
MODERATOR DEUTERATION AND
DE-DEUTERATION SYSTEM

5.1-331

FIGURE 5.1.7-4
MODERATOR D₂O SAMPLING
SYSTEM

5.1-332

FIGURE 5.1.7-5
FLOW DIAGRAM
MODERATOR REFRIGERATION
SYSTEM

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FIGURE 5.1.7-6
D₂O SUPPLY SYSTEM

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5.1-334.

FIGURE 5.1.7-7
D₂O VAPOR RECOVERY SYSTEM

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5.1-335

FIGURE 5.1.7-8
D₂O CLEANUP SYSTEM

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FIGURE 5.1.7-9
D₂O UPGRADING AND
CLEANUP SYSTEMS

5.1-336

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5.1-337

FIGURE 5.1.7-10
D₂O UPGRADING SYSTEM

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5.1-338

FIGURE 5.1.7-11
RCS D₂O
SAMPLING SYSTEM

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FIGURE 5.1.7-12
RCS DEUTERATION &
DE-DEUTERATION SYSTEM

5.1.8 Turbine Plant Systems

The turbine plant equipment includes the steam and power conversion system components of the steam cycle. The thermal energy from the reactor is utilized to produce saturated steam within the steam generators and deliver it to the turbine stop valve at 1085 psia.

A description of the turbine plant equipment is detailed in the following sections:

- Turbine Steam System and Turbine Generator (5.1.8.1)
- Main Steam System (5.1.8.2)
- Condensate and Feedwater System (5.1.8.3)
- Condensing System (5.1.8.4)
- Emergency Feedwater System (5.1.8.5)

Other systems furnished are:

Turbine Building Closed Cooling Water System

A closed cooling water system is provided with three 50% capacity pumps (4600 gpm each) motor driven pumps, air tank and heat exchangers which dissipates heat to the main cooling towers.

Chemical Treatment System

The chemical treatment system is used to maintain the water chemistry of the feedwater and consists of two hydrazine feed pumps, two ammonia feed pumps, one hydrogen storage tank and one ammonia storage tank. The hydrazine chemically removes the dissolved oxygen from the feedwater and the ammonia controls the pH.

Neutralization System

The neutralization system consists of two pumps, a blower, and a tank. The tank is used to chemically neutralize the spent reagent from the demineralization system to acceptable levels prior to discharge.

5.1.8.1 Turbine Steam System And Turbine Generator

Functions

The function of the turbine-generator (TG) and the turbine steam system is to receive steam from the main steam system and to transform the thermal energy in the steam to electrical energy.

Design Requirements

- The turbine generator is designed for load following operation. Load changes are initiated by adjustment of the turbine control system load reference set point which positions the turbine admission valve. The turbine first stage pressure is sensed to control average coolant temperature. Changes in steam flow and feed flow are sensed and SG water level is controlled.
- The turbine is designed for normal operation based on steam conditions at the throttle valve, namely; 1085 psia, 554.6°F with a moisture content of 0.45% maximum. The steam flow rate to the high pressure turbine is 16.259×10^6 lbs/hr.
- The TG is designed for safe continuous operation at the maximum capability (valves wide open - VWO).

- ° The TG and the associated steam systems are capable of 50 percent load reduction without causing a reactor trip by dumping steam into the condenser through the turbine bypass system.

System Description

This system description includes the following: the turbine and the turbine steam system, the reheaters, extraction steam system, the turbine control system, turbine bypass system, the generator and the subsystems associated with the operation and control of the turbine generator.

The heat balance for the turbine cycle is shown on Figure 5.1.8-1. The turbine steam system piping/flow diagram is given on Figure 5.1.8-2. Figures 5.1.8-3 and 5.1.8-4 shows the plan and elevations of the turbine building and the heater bay and the equipment arrangement in these buildings.

The turbine steam system and the TG are ANS non-nuclear safety, with associated piping designed in accordance with the Power Piping Code, ANSI B31.1. Pressure containing vessels are constructed in accordance with the ASME Boiler & Pressure Vessel Code, Section VIII, Division 1.

Generator rating, temperature rise, and insulation class are in accordance with the latest ANSI standards, C50.10 & C50.13.

Turbine & Turbine Steam System

The turbine is an 1,800 rpm, tandem compound, six-flow, single stage steam reheat machine with 43-inch last-stage blades. It consists of four cylinders: one double flow high pressure cylinder and three double flow low pressure cylinders.

Steam from the main steam header is admitted to the high pressure turbine through the turbine stop and control valves. The main stop valves are welded directly to the inlet connections of the control valves. To improve turbine efficiency, to reduce low pressure turbine exhaust moisture, and to reduce maintenance on the low pressure blading, the steam is passed through combination moisture separator-reheaters. Two moisture separators are arranged for operation in parallel. Moisture is removed from the wet exhaust steam and the dry steam is reheated by condensing high pressure steam. The reheated steam flows to three low pressure double flow machines.

The total energy of the entrained steam and water in the moisture separator reheater system would, on loss of load, act to accelerate the turbine rotor to overspeed in excess of design. To prevent this occurrence, combined stop and intercept valves are provided in the crossover lines between the moisture separator-reheaters and the low pressure turbine. Relief valves which discharge into the condenser are provided in this piping to prevent overpressure of these reheaters and the associated piping.

The moisture separator/reheaters are located on the operating floor on both sides of the turbine. Steam from the main steam header heats the exhaust steam from the H.P. turbine, while chevron type baffles remove moisture from the exhaust steam.

The exhaust steam from the three LP turbines flows into individual sections, or shells, of the main condenser, where the steam is condensed by the circulating water.

Turbine extraction steam, used for feedwater heating, is taken from six extraction points: one on the high pressure turbine casing, one from the exhaust of the high pressure turbine, and four from the low pressure turbine casing. Six stages of closed

feedwater heaters are provided. Motor operated block valves and power-assisted non-return valves in the extraction piping protect against the possibility of turbine water induction or over-speed due to energy stored in the extraction steam system.

Turbine generator bearings are lubricated by a conventional pressurized oil system; the main lubricating oil pump being driven from the turbine during normal operation. During startup or shutdown, a-c motor drive pumps supply bearing oil to the TG.

The design conditions for the turbine are:

a. Steam pressure at stop valve	1085 psia
b. Steam quality at stop valve	99.55 percent
c. Turbine back pressure (design)	3.75 in. Hg Abs.
d. System makeup	zero percent
e. Final feedwater temperature	450°F
f. Rated steam design flow	16.259×10^6 lb/hr
g. 105 percent of rated flow	17.072×10^6 lb/hr

Turbine Control System

The turbine control system is capable of remote manual or automatic starting and loading of the unit at a preset rate, and holding speed and load at a preset level. The system contains valve positioning, operating and tripping devices with provisions for testing valve operation.

The turbine control system is an electrohydraulic control (EHC) and consists of three major units to minimize interactions. A speed control unit compares actual turbine speed with the speed reference, or actual acceleration with the acceleration reference, and provides one speed error signal for the load control unit. The load control unit combines the speed error signal with the load reference signal, and provides limits and biases to determine desired steam flow signals for the turbine stop valves, turbine control valves, and intercept valves. Finally, the valve flow control units accurately position the appropriate valves to obtain the desired steam flows to the turbine.

A high pressure fluid system is provided to convert the low power level signals from the EHC circuits to high powered level mechanical outputs for positioning steam valves. This system consists of a fluid reservoir, two independent pumping systems, fluid coolers, accumulators, a fluid transfer and filter unit.

Lubricating Oil System

The functions of the lubricating oil system are to receive, store, purify, cool, and provide lube oil to the thrust and journal bearings of the main turbine-generator and the feedwater pump turbines, to provide oil to the generator hydrogen seal oil system, and to provide high pressure oil to the turbine control system.

The lubricating oil system consists of two integrated sections (1) the turbine-generator lubricating oil section which includes the bearing oil pumps, turbine lubricating oil reservoir, and the oil coolers; and (2) the lubricating oil conditioning and storage circuit consisting of clear and used oil storage tanks, an oil purifier, and a motor driven transfer pump. The pump is a positive displacement type, capable of two-speed operation to accomplish both the transfer and circulation requirements.

The clean and used oil storage tanks are located inside a fireproof room equipped with a trap drain, water sprays, and vent fans. The pumps and piping are arranged so that oil can be processed from the turbine lubricating oil reservoir or either of the storage tanks. The processed oil can be returned to either of the storage tanks or to the turbine lubricating oil reservoir as required. Vapor extractors purge oil fumes from the reservoir and exhaust them to the atmosphere outside the turbine building.

Lubricating oil to the turbine-generator is normally supplied from a turbine shaft driven pump. An a-c motor drive turning-gear oil pump supplies bearing lubrication for startup, shutdown, and standby operation when a-c power is available. A d-c motor driven bearing oil pump, operated from the normal station battery, ensures bearing lubrication in the event a-c power fails.

Turbine lubricating oil coolers, located in the turbine lubricating oil reservoir, remove heat generated in the turbine-generator bearings and transfer it to the turbine plant component cooling system.

Turbine Gland Sealing System

The sealing system prevents the leakage of steam from the turbine packing glands into the turbine building and also prevents the leakage of air into the main condenser.

The annulus space between the turbine shafts and the casings is sealed by steam supplied to labyrinth packings. The labyrinth packing seals against vacuum on one side by allowing the sealing steam to leak outward to a vent annulus that is maintained at a slight vacuum, and seals against atmospheric pressure on the

other side by allowing a small amount of leakage inward. The vent annulus also receives air leakage from the outside. This air-steam mixture is then conducted to the gland condenser. When the packing seals against positive pressure, the sealing steam connection acts as a leakoff.

The steam seal header is automatically regulated at 4 psig. At startup and low loads auxiliary steam or main steam is used for the seals. At higher load when the amount of leakoff from the pressure packings is more than that required by the vacuum packings the excess is discharged to the gland condenser. The gland condenser returns seal leakoff to the condenser as a condensate.

The cooling medium for the gland condenser is the condensate system. During normal plant operation noncondensable gases are removed from the gland condenser by a motor-driven blower, and discharged to the atmosphere.

Turbine Bypass System

On a large external electrical load decrease (up to 50 percent), the turbine bypass system relieves main steam directly to the condenser, thus preventing a reactor or turbine trip and lifting of the main steam safety valves.

The turbine bypass system includes a turbine bypass header, branching from the main steam manifold in the main steam system and individual bypass lines connecting the bypass header to the condenser. An isolation valve, a turbine bypass control valve, and a second isolation valve are mounted in series on each individual turbine bypass line.

An uncontrolled plant cooldown caused by a single valve sticking open is mitigated by the use of a group of smaller valves installed in parallel instead of a single larger valve.

The full capacity of the turbine bypass sytem is 55 percent of maximum calculated steam flow at full load steam pressure which, together with the NSSS transient capability, meets the above design requirements. This steam flow is equally distributed among the three condenser shells by the present opening sequence of the turbine bypass control valves.

After a normal shutdown of the turbine-generator leading to plant cooldown, the turbine bypass control valves are opened to release steam generated from reactor coolant system sensible heat. Reactor cooldown, programmed to minimize thermal transients and based on sensible heat release, is accomplished by gradually decreasing the setpoint of a steam pressure controller. This closes the turbine bypass control valves, thus transferring the cooldown process to the residual heat removal system.

Generator

The generator is sized to accept the output of the turbine at rate steam conditions. It is a direct-coupled, 1800 rpm, 25,000 Volt, 3 phase, 60 Hz unit rated at 1650 MVA with 0.90 power factor. It is a totally enclosed hydrogen cooled type with deionized water as the liquid coolant. The generator is equipped with an excitation system, hydrogen control system and a seal oil system. The generator terminals are connected to the main step-up transformer and unit station service transformers with generator isolated phase bus leads.

The generator rotor is furnished with an internal cooling system including: hydrogen coolers, terminal bushings, instruments, grounding pads, seal housing insulation, foundation plates, shims, and special tools.

The generator stator is furnished with the following external equipment: deionized water circulating and cooling unit assembled on a skid and including storage tank, pumps, coolers, deionizer, flow meter, conductivity cells, gauges, piping, valves, filters, instruments, and regulating equipment; stator winding control cabinet assembled and combined with the hydrogen control cabinet including annunciator, generator automatic runback logic and all necessary control devices.

The generator hydrogen system includes: two hydrogen coolers, one skid mounted seal oil unit, hydrogen manifold with one bottle pressure regulator with high and low pressure gauges, pressure switch for hydrogen supply pressure "low" alarm, shutoff valves and bottle connectors, generator hydrogen pressure regulator hydrogen storage bottles, control cabinet, temperature detectors, and special tools.

The alternator bearings, the silicon diode rectifier assemblies, the main generator collector and the brush rigging are all totally enclosed within the alternator exciter housing with suitable heat exchangers and means for circulating air. The closed ventilation circuit is equipped with water to air coolers located in the exciter frame. The excitation switchgear is an integrated unit of standard low voltage, located indoors, and metal enclosed. The function of the excitation switchgear is to connect, rectify and control excitation to the a-c alternator exciter from the alternator stator, and to provide voltage regulation by adjustment of the generator field voltage (d-c regulator) or the generator terminal voltage (ac regulator). The excitation switchgear houses the exciter field breaker, the thyristor regulator bridge and the a-c and d-c regulator logic.

Exciter

The exciter is a direct driven alternator and stationary silicon diode rectifier type.

Gas Systems

The carbon dioxide system consists of a four ton liquid carbon dioxide storage unit with refrigeration system, vaporizer, relief valves and two pressure reducing valves. Carbon dioxide is used for purging hydrogen from the generator housing during shutdown, and for purging air from the housing before being filled with hydrogen during startup. Sufficient capacity is provided to twice purge the system of air and once of hydrogen.

Hydrogen gas is used to cool the rotor of the generator and is circulated within the generator housing under pressure. Two shell and tube type coolers at the ends of the generator are supplied with cooling water to dissipate the rotor heat and wind losses.

The hydrogen is supplied from a series of bottled containers which are individually connected to a manifold. The manifold is equipped with a relief valve and two pressure regulators with isolation valves.

Enough hydrogen supply is provided to fill and pressurize the generator once and supply the required makeup for a nominal period.

Other Turbine Plant Equipment

Turbine Building Closed Cooling Water System

A closed cooling water system is provided with three 50 percent capacity (4,600 gpm each) motor driven water pumps, air tank and heat exchangers, which dissipates heat to the main cooling towers. The heat exchangers are two 50 percent capacity shell and tube type, designed for a flow rate of 4,600 gpm on both the shell and tube sides. The tubes are 90-10 CuNi material, and supply 95°F water to the system based on a supply water temperature of 92°F from the plant service water system.

Chemical Treatment System

The chemical treatment system is used to maintain the water chemistry of the feedwater and consists of two hydrazine feed pumps, two ammonia feed pumps, one hydrazine storage tank and one ammonia storage tank. The hydrazine chemically removes the dissolved oxygen from the feedwater and the ammonia controls the pH.

Neutralization System

The neutralization system consists of two pumps, one blower and one tank. The neutralization tank is used to chemically neutralize the spent regenerant from the demineralization system and condensate polishing system to acceptable levels prior to discharge.

5.1.8.2 Main Steam System

Functions

The function of the Main Steam System (MS) is to transport saturated steam generated in the evaporator sections of the steam generators to the high pressure (HP) section of the turbine generator (TG).

The Steam Dump and Bypass System provides a flow path for the steam generator effluent to be bypassed to the condenser, when the TG is not operating, during low load conditions (less than 25% of the rated TG load), or at any other time when the SG's are producing more steam than needed by the turbine.

The main steam system also supplies steam to the reheaters, SG feed pump turbines, steam seal system, and steam jet air ejectors.

Design Requirements

- ° The MS system handles 17.24×10^6 lbs/hr of steam from 4 steam generators. The conditions of nuclear steam at SG nozzles are 1120 psia, 558.5°F, 0.25% moisture. The steam required at TG throttle valve is at 1085 psia, 554.6°F and 0.45% moisture.
- ° Safety relief valves (SRV) are provided on each MS line upstream of the MSIV's for protection of SG and piping from overpressurization. These valves are located in the penetration building.
- ° A power operated main steam isolation valve (MSIV) capable of establishing shutoff under conditions of design pressure, design temperature, and flow conditions resulting from a break just upstream or downstream is provided in each main steam line outside of containment. The full open to close

stroke time of the MSIV's in 5 seconds or less upon receipt of an MSIS (Isolation signal).

- ° The main steam from all 4 SG is mixed in a mixing header upstream of the TG throttle, and downstream of the MSIV's.
- ° Steam dump and/or turbine bypass system is sized to pass a minimum of 55 percent of the SG full power steam flow and is designed for the associated pressure.
- ° Two atmospheric dump valves capable of dissipating reactor decay heat and reactor coolant pump heat when the MSIV's are closed (resulting in a controlled plant cooldown) are provided on each MS line upstream of the MSIV's. Each of these valves will be sized for 350,000 lbs/hr of 350°F saturated steam. No single atmospheric dump valve will have a capacity greater than 1.9×10^6 lbs/hr at 1220 psia.
- ° The main steam piping valves and associated components from the steam generators up to and including the containment isolation valves are designed to seismic Category I Safety Class 2 and constructed in accordance with the ASME B&PV Code, Section III, Class 2. The MS piping downstream of MSIV's is designed in accordance with the Power Piping Code ANSI B31.1.

System Description

Steam from four steam generators flows through four 28-inch carbon steel pipes to a 48-inch main steam header. Each MS line includes a power-operated atmospheric relief valve, safety valves and isolation valve (MSIV). Turbine supply lines carry steam from the main steam header, through the high-pressure turbine stop valve assembly, to the HP turbine.

Principal design and performance characteristics of the main steam system and its principal components are summarized in Table 5.1.8-1. The turbine steam system includes the turbine bypass system. The heat balance is shown on Figure 5.1.8-1.

The containment penetrations for the MS piping are located in the penetration building. All four main steam pipes pass through this area located between the containment and the TGB. The safety valves and the power-operated atmospheric relief (dump) valves are provided in each main steam line immediately outside the containment upstream of the isolation valves and are located in the penetrations building.

There are five ASME B&PV Code, Section III safety valves (6" x 10") on each main steam line for protection of the steam generators and steam lines against over-pressurization. The spring loaded code safety valves are sized to limit the steam generator maximum pressure to less than 110 percent of the steam generator design pressure following the worst case increasing pressure transient.

The power operated atmospheric relief (dump) valves are capable of dissipating reactor decay heat and reactor coolant pump heat following reactor shutdown when the main heat sinks are not available or when the MSIV's are closed. These valves are each sized to pass 350,000 lbs/hr of 350°F saturated steam to atmosphere and discharge through an exhaust silencer. Operation of the power operated relief valves is automatic based on steam line pressure, and remote adjustment of the set point is provided. Local manual operators are provided in case of complete loss of automatic control.

There is one isolation valve in each 28 inch main steam line that provides the combined functions of containment isolation, non-return and excess flow trip. The excess flow trip function is required to stop flow in the event of a main steam line break downstream of the isolation valve. The non-return function protects against reverse flow from other steam generators through interconnecting piping, in

the event of a main steamline break upstream of the isolation valve.

The main steam lines are combined into a header downstream of the isolation valves. Pipes from this header supply steam to the high pressure turbine, the single stage reheat of the moisture separator-reheaters, the steam sealing system, and the turbine bypass system SG feed pump turbines.

Steam piping connects the emergency feed water pump steam supply header to two main steam pipes upstream of the main steam isolation valves. These pipes and the header provide steam to drive the auxiliary feedwater pump turbine.

The main steam system is capable of removing heat from the reactor coolant system following sudden load rejection or trip of the TG by automatically bypassing main steam to the condenser through the turbine bypass system or by relieving to the atmosphere through the main steam safety valves or main steam atmospheric dump valves, if the turbine bypass system is unavailable.

TABLE 5.1.8-1

Main Steam System

Principal Design and Performance Characteristics

<u>Item</u>	<u>Design & Performance Characteristics</u>
<u>Main Steam System</u>	
No load pressure, psia	1220
Normal operating pressure, psia	1120
Normal operating temp., °F	558.5
Normal operating flow, lbs/hr x 10 ⁶	17.24
Design pressure, psig	1320
Design temperature, °F	580
Maximum out-of-balance pressure between an SG and any other, psi	10
<u>MSIV's</u>	
Number	4 (one on each main steam line)
Closing time, see (max.)	5 (after receipt of closure signal)
<u>MS Safety Valves</u>	
Number	20 (five on each MS line)
Relieving capacity, total	110% of rated steam flow at 110% design pressure.
<u>MS Atmospheric Dump Valves</u>	
Number	8 (two on each ms line)
Relieving capacity, total, lbs/hr	8 x 350,000 (At least 10% max. oper. flows at no load press.).

5.1.8.3 Condensate and Feedwater System

Functions

The function of the condensate and feedwater systems is to return condensed steam from the condenser and the drains of the regenerative feedwater heating cycle to the steam generators, while maintaining the water inventories throughout the systems constant. The systems automatically control the water levels in the steam generators and condenser hotwell during steady state and transient conditions.

Design Requirements

- ° The portion of the feedwater system extending from and including the feedwater flow control valves to the steam generator inlets is safety related and seismic Category I.
- ° The entire condensate system and the portions of the feedwater system upstream of the feedwater flow control valves are neither safety related nor seismic Category I.
- ° The feedwater isolation valves are categorized as containment isolation valves and, as such, shall be located as close as practical to the outside of the containment structure.
- ° The safety related portion of the feedwater system is constructed in accordance with ASME B&PV Code, Section III, Class 2 and is designated Safety Class 2.
- ° The condensate system and the nonsafety related portions of the feedwater system are constructed in accordance with ASME B&PV Code, Section VIII and ANSI-B31.1 and are designated non-nuclear safety class.

- ° The condensate feedwater systems are isolated from the steam generators following a main steam isolation signal (MSIS).

System Description

The condensate and feedwater system takes suction from the main condenser hotwells and from the heater drain tank and delivers feedwater to the steam generators at increased temperature and pressure. Under normal operating conditions two-thirds of steam generator flow is obtained from the condenser hotwell and one-third is obtained from the heater drains.

Each steam generator is provided with a feedwater control system. This system is designed to provide 18.96×10^6 lb/hr of 450°F water to the steam generators under turbine control valves wide-open conditions. Included in this design flow is 1% steam generator normal blowdown. The feedwater heaters are designed to provide a final feedwater temperature of 450°F with six stages of closed feedwater heating.

The condensate is drawn from the condenser hotwell by three half-capacity motor driven, vertical condensate pumps, each operating at two-thirds capacity during normal operation. The pumps discharge into a common header.

After leaving the steam packing exhausters, the condensate flow path is split into two parallel strings each with six stages of feedwater heating. Each string is comprised of five low pressure heaters and one high pressure heater. Bypass piping and block valves are provided around strings of feedwater heaters to take units out of service for maintenance.

Condensate pumped from the condenser hotwell through the five low pressure heaters reaches the suction connection of the steam generator feed pumps at the required suction head. The flow from the two 55 percent capacity, centrifugal, turbine-driven steam generator feedwater pumps will then pass through the high pressure feedwater heater to the steam generators.

The condensate system discharges water to, or takes water from, the condensate storage tank as required to maintain the water inventory in the steam generator-turbine cycle.

Equipment Description

Condensate Pumps

The condensate pumps provide about 67 percent of the total feedwater flow. Three vertical centrifugal motor-driven pumps take suction from the condenser hotwells. Each pump is designed to provide at least fifty percent of the full load condensate flow; therefore, during normal operation only two pumps will be operating and one will be on standby.

Design Parameters

° Quantity	3 @ 50%
° Capacity per pump	11,615 GPM
° Total developed head	550 ft.
° Temperature of condensate	108.7°F
° Electric motor rating	2,000 HP

Feedwater Heaters

Two parallel strings of feedwater heaters each consisting of five low pressure feedwater heaters and one high pressure feedwater

heater are provided. The heaters in each string are designed for one-half the heat balance requirement at the turbine valves wide open (VWO) condition. The heaters are designed to provide a final feedwater temperature of 450°F.

Extraction steam from the main turbine and crossover piping is the heat source for the heaters. Automatic non-return valves are provided in the extraction lines for turbine overspeed protection. The drains from the highest pressure heaters are cascaded into the second highest pressure heaters. The drains from the moisture separators are also piped to these heaters. The drains from the second highest pressure heaters are piped to the heater drain tank and pumped to the suction piping of the steam generator feed pumps by the heater drain pumps. Heater drains from the four lowest pressure heaters are cascaded and eventually discharged into the condenser.

The two sets of lowest pressure heaters are located horizontally in the condenser necks. The set of lowest pressure heaters have separate drain coolers which are located on the turbine room basement floor. The remaining strings of feedwater heaters are located in the turbine room auxiliary bay. All heaters are of the two pass, U-tube type. The separate drain coolers are of the single pass, straight tube type.

Heater Drain Pumps

The heater drain pumps provide about one-third of the total feedwater flow. Two vertical centrifugal motor driven pumps take suction from the heater drains tank and deliver the flow to the suction of the steam generator feed pumps. Each pump has 50 percent capacity.

Design Parameters

- ° Quantity 2 @ 50%
- ° Capacity per pump 2500 gpm
- ° Total developed head 200 ft.
- ° Temperature of drains 370.7°F
- ° Electric motor rating 125 HP

Steam Generator Feed Pumps

Two horizontal, turbine-driven steam generator feed pumps are provided in the feedwater system. Recirculation control valves are provided in piping off of the pump discharge lines to permit direct recirculation of feedwater to the condenser hotwell. The control valves in each line will open on a low-flow signal from a flow measuring device located upstream of each pump and will thereby protect the pumps from damage due to insufficient flow.

Design Parameters

- ° Capacity per pump 17,225 GPM
- ° Total developed head 2800 ft.
- ° Temperature of feedwater 370°F (@ Feed Pump)
- ° Turbine drive rating 15,000 BHP

Steam Generator Feed Pump Turbines

Individual steam turbines drive the steam generator feed pumps. The turbine drives are of the dual admission type, and each is equipped with two sets of main stop and control valves. One set regulates high pressure steam from the steam generators, and the other set regulates low pressure steam extracted from the crossover piping.

Steam Generator Startup Feed Pump

One steam generator startup feed pump is provided to each unit to provide normal requirements for startup, cooldown and for no load operation. The pump is motor driven from a normal plant electrical bus.

Design Parameters

- Pump Capacity 1,000 GPM
- Total developed head 3,000 ft
- Temperature of feedwater 70°F (max)
- Electric motor rating 875 HP

Circulating Water System

The circulating water system is designed to provide a continuous flow of cooling water to the main condensing system to remove heat rejected from the power cycle. The system also supplies water to the service water system.

A vacuum priming tank, connected to mechanical vacuum pumps and to the condenser waterboxes is provided to aid in the establishment of the syphon in the circulating water system at startup, and to continually remove released air from the condenser waterboxes.

- Number pumps 5 @ 25%
- Capacity of each pump 16,150 gpm
- Total developed head 100 ft.
- Motor Horsepower 5500 HP

5.1.8.4 Condensing System

Main Condenser Functions

The objective of the main condenser system is to provide a heat sink for the turbine exhaust steam, turbine bypass steam, and other steam and water flows discharged to the condenser.

Design Requirements

The condenser is capable of accepting 8.49×10^9 BTU/hr heat load at the rated turbine load and design operating conditions. The condenser can also accept 8.59×10^6 lbs/hr of bypass steam flow based on 50 percent of full load steam flow without exceeding the turbine low vacuum trip point.

The condenser has been designed to deaerate the condensate and limit the oxygen content to 0.005 cc/liter at any load during normal operation.

Description

The main condenser system consists of three deaerating, double-pass, single pressure, radial flow type surface condensers. Each one-third capacity condenser is located beneath one of the three low-pressure cylinders with the tubes oriented transverse to the turbine-generator axis and is rigidly supported on a foundation. The steam exhausts directly down into the condenser shells through exhaust openings in the bottom of each low pressure turbine casing. The condensers also receive steam from the steam-generator feedpump turbines.

A stainless steel expansion joint is installed between the upper and lower steam inlet sections to permit movement resulting from

the temperature changes of the equipment. Cross connections are provided for equalization of pressure between condenser shells. Provisions have been made for the mounting of two one-third capacity low pressure extraction feedwater heaters in the neck of each condenser. The hotwells are sized for a minimum of 3-minute storage.

The main condenser system will produce a back pressure of 2.50 inches of mercury absolute, when operating at rated turbine output with 77°F cooling water and 85 percent clean tubes. Valving is provided so that the condenser flow may be reversed for backwashing the circulating water piping to control marine growth and fouling.

Copper/nickel tubes are used throughout the condenser. The condenser tube sheets are copper/nickel. The condenser shells, tube support plates, and waterboxes are constructed of carbon steel.

The condenser maintains a negligible oxygen concentration in the effluent condensate by limiting subcooling and thereby limiting gas solubility. Mechanical vacuum pumps evacuate the noncondensable gases during normal operation and during startup.

The main condenser system is designed as a non-nuclear system. The extraction steam and turbine drain piping located internal to the condensers are designed in accordance with the Power Piping Code, ANSI B31.1.

Condenser Design Parameters

Number of shells per unit	Three - 1/3 size
Total condensing surface	994,497 sq. ft.
Number of passes per shell	Two
Tube size	1-1/8" O.D. X 0.035" thick
Tube length	51'0"
No. of tubes per shell	22,074
Tube material	90/10 copper-nickel
Circulating water flow	644,600 GPM
Tube velocity	7.25 fps
Temperature rise	26°F
Condenser pressure - design	3.75 Hg absolute
Inlet water temperature-design	92°F

Equipment is provided to regenerate the demineralizer resins. Wastes from this regeneration are discharged to the radioactive liquid waste system.

5.1.8.5 Emergency Feedwater System

Functions

The function of the emergency feedwater system is to ensure a sufficient supply of cooling water to the steam generators so they can act as heat sinks for sensible and decay heat removal from the reactor core under the following events:

- Loss of normal feedwater supply with or without a coincident loss of off-site power.
- Major or minor secondary system pipe breaks with or without a loss of offsite power.

- Steam generator tube rupture with or without a loss of offsite power.
- A single failure of any component is assumed in addition to the initiating event.

Design Requirements

- The system starts automatically on a loss of station power, a safeguard sequence signal, a trip of both steam generator feed pumps, or a 2/3 low-low level signal in any steam generator.
- The system is safety related and seismic Category I.
- The system, from and including the containment isolation valves and up to the connections with the feedwater system, is constructed in accordance with ASME B&PV Code, Section III, Class 2 and Safety Class 2 criteria.
- The balance of the system is constructed in accordance with ASME B&PV Code, Section III, Class 3 and Safety Class 3 criteria.
- The system delivers sufficient emergency feedwater to the steam generators following loss of normal feedwater to prevent lifting of the pressurizer relief valves caused by temperature and pressure buildup in the reactor coolant system.
- The steam delivers emergency feedwater against a steam generator pressure corresponding to the lowest main steam safety valve set pressure plus accumulation.

System Design Description

Upon loss of feedwater flow, the reactor is made subcritical by control rod insertion. Decay and sensible heat will be transferred to the steam generators by the reactor coolant system. When power is not available for the reactor coolant pumps, natural circulation will suffice. Heat is removed from the steam generator via the safety and relief valves, and the construction of this process is maintained by water makeup from the emergency feedwater system. The emergency feedwater system is designed to operate until the reactor coolant system pressure is reduced to a value below which the Shutdown Cooling system can be operated.

The emergency feedwater system pumps unheated water from the condensate storage tank to the steam generators, and is comprised of one motor-driven pump, one turbine-driven pump and a system of piping, valves and orifices. Either pump has sufficient capacity to satisfy the above design requirements.

An adequate reserve of water is always maintained in the condensate storage tank (approximately 400,000 gallons), and makeup to the tank is available from the demineralized water makeup system.

The motor-driven pump, pump controls and actuating signal are powered from an emergency bus. Steam for the steam turbine-driven pump is supplied from the main steam system upstream of the MSIV's via air-operated valves. The solenoid operators controlling the air-operated valves are supplied power from separate battery sources.

Both pumps feed a header which in turn supplies four air-operated flow limiting valves. Each valve is normally in the open position when the system is not operating. The valve fails to the open position on loss of air. During system operation, the valve

closes to a position which regulates the minimum flow. The valve controller receives operating air from its own air regulator. The air regulator operates to maintain a differential pressure across the flow orifice which corresponds to the desired flow rate. Should a valve fail to the open position, flow is limited by the flow orifice, assuming a downstream pipe rupture and both auxiliary feed pumps operating. Air is furnished from an air receiver located in the emergency pump room. The receiver is supplied by the station instrument air system through two check valves arranged in series. The receiver is alarmed in the control room should its pressure fall below the minimum requirement. Each flow limiting valve is fitted with a handwheel for manual operation. The flow limiting function is accomplished without electrical input.

Downstream of each air-operated flow limiting valve, the emergency feed line joins one of the steam generator feed lines outside the containment. The combined feed line enters the containment through a single penetration and feeds a single steam generator.

In the automatic mode, emergency feed regulation utilizes a flow control valve using the pressure drop across the orifice to limit the flow to each steam generator. In the manual mode, the valve will be controlled by an operator to provide the flow requested.

The combination of one turbine-driven and one motor-driven emergency feedwater pump provides a diversity of power sources to assure delivery of condensate under an emergency condition.

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FIGURE 5.1.8-2
MAIN STEAM SYSTEM

0 6 12 18 24
SCALE FEET

FIGURE 5.1.8-3
TURBINE BUILDING
SECTION B-B

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FIGURE 5.1.8-4
TURBINE BUILDING
GENERAL ARRANGEMENT
PLAN EL. 55'-0"

5.1.9 Radioactive Waste Management Systems

The radioactive waste systems remove radioactive contaminants from the plant system before discharge of effluents to the environment during normal operation and expected operational occurrences.

Design Requirements

The radioactive waste systems are designed to prevent an uncontrolled release of radioactive contaminants into the environment and to limit offsite doses from normal plant releases to very small fractions of Title 10, Code of Federal Regulations, Part 100 (10 CFR 100) dose limits. The design criteria of 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion As Low as is Reasonable Achievable for radioactive material in Light Water Cooled Nuclear Power Reactor Effluents", are applied to the design of the Heavy Water Reactor radwaste systems. The radwaste systems operate in conjunction with heavy water recovery systems to minimize the loss of heavy water.

The radwaste systems are composed of the Liquid Radwaste System, the Gaseous Radwaste System and the Solid Radwaste System, and are discussed below:

5.1.9.1 Liquid Radwaste System

The Liquid Radwaste System is designed to collect on a continuous basis, and process on a batch basis, all radioactive liquid waste generated in the operation of the plant. Inputs are segregated by chemical and radioactivity concentrations to facilitate chemical adjustments and selective treatment as necessary.

As such, equipment drains in the containment, the D₂O Recovery System product, and any source that may contain recoverable traces of D₂O are routed to the D₂O Cleanup section of the Liquid Radwaste System. Waste, such as floor drains in the Service Building, are routed to the Miscellaneous Waste section. Radioactive laundry and hot shower drains are routed to the Detergent Waste section, and regenerant waste from the condensate polishing system are routed to the Regenerant Waste section. Radioactive laboratory liquid waste are handled by the Hot Lab waste section. The process flow diagram for the liquid waste system is shown on Figure 5.1.9-1.

After treatment in the liquid radwaste system, the effluent is either routed to the D₂O Upgrading System or is channelled for reuse in the plant.

5.1.9.2 Gaseous Radwaste System

The radioactive gaseous waste system is designed to collect, process and hold all radioactive gases generated during reactor operation. The process flow diagram is shown on Figure 5.1.9-2. The Gaseous Radwaste System is located in the radwaste area of the Service Building.

The major sources of radioactive gases are the reactor coolant degasser-condenser, and the moderator cover gas system. Additional inputs occur on a batch basis from the moderator cover gas system, the nitrogen blanketing system, the various D₂O storage tanks located throughout the plant, and the feed evaporator in D₂O Upgrading System. The gases are sent to the suction side of the compressor, where they are mixed with a recirculating nitrogen stream in order to maintain a hydrogen concentration below the explosive point. The gases are then compressed to approximately 85 psig, pass through a recombiner to remove the hydrogen, and

then enter a gas decay tank. The contents of the full gas decay tank are held a minimum of 60 days to allow for the decay of short-lived isotopes. After the decay period has elapsed, the contents of the gas decay tank are sampled, filtered, and released through the building exhaust ventilation system.

5.1.9.3 Solid Radwaste System

The Solid Radwaste System provides for the handling of all solid waste generated within the plant and includes facilities for moving, concentrating for storage, storage, retrieval, and preparation for shipping. The objective of the system is to provide these services for the solid waste generated during normal operation without subjecting plant personnel to excessive radiation exposure.

The system process flow diagram is shown on Figure 5.1.9-3.

Remote control of solidifying and drumming of the high level waste, the use of shielded overpacks for the containers, and a minimum of 2 ft thick concrete shield walls insure the ability to accommodate the expected radiation levels.

The solid radwaste system has several input sources. Dry materials that become contaminated in the course of plant operation are compacted into containers and disposed of as a solid radwaste. Typical materials which are disposed of in this way are:

- ° Paper, tape, polyethylene, etc. that has been contaminated;
- ° Anti-contamination clothing, gloves, and shoe covers that are contaminated beyond the capability of the laundry system;
- ° Small handtools which cannot be economically decontaminated;
- ° Miscellaneous materials from contaminated areas; and
- ° All disposable HVAC equipment, such as filters, from a possibly contaminated system.

Other solid wastes, in the form of slurries, come from the plants purification systems. Some of the sources are as follows:

- Moderator purification demineralizers;
- Primary heat transport purification demineralizers;
- Spent fuel pool demineralizer;
- Radwaste demineralizers;
- Ion exchange columns;
- Evaporator bottoms; and
- Condensate polishing system powdered resins.

In addition, spent filter cartridges and filter backwash are processed by the Solid Radwaste System.

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FIGURE 5.1.9-1
RADIOACTIVE LIQUID
WASTE SYSTEM

FIGURE 5.1.9-2
RADIOACTIVE GASEOUS
WASTE SYSTEM

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FIGURE 5.1.9-3
RADIOACTIVE SOLID
WASTE SYSTEM