With the BIT functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

Reactivity Control System Malfunction

Criterion 31: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout of a control rod, buy limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals. Details of the effects of continuous withdrawal of a control rod and continuous deboration are described in Section 14.1 and Section 9.2, respectively.

Maximum Reactivity Worth of Control Rods

Criterion 32: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs control rod clusters. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rods comprise the control groups which are used to control load and reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed yielding reactivity insertion rates no greater than 66 pcm/sec which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

The rod drive design permits only two groups to be withdrawn at the same time. Should more than two groups try to move, insufficient power is available to activate the movable holding latches and the affected control rods would disengage and fall.

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Specifically, a bank contains either one or two groups. Within a bank the group may be moved only sequentially one step at a time. Two banks may be moved simultaneously, e.g. banks C and D. However, group 1 in bank D may be moved together (one step), then group 2 in each bank simultaneously (one step). Therefore, no more than two groups can be moved together and this forms the basis of the assumption.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 80 steps per minute (~50 inches per minute).

3.1.3 Safety Limits

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient conditions without violating the integrity of the fuel elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Design parameters which are pertinent to safety limits are specified below for the nuclear, control, thermal and hydraulic, and mechanical aspects of the design.

Nuclear Limits

At full power (license application power) the nuclear heat flux hot channel factor F_Q^N, is not exceeded. For any condition of power level, coolant temperature and pressure which is permitted by the control and protection system during normal operation and anticipated transients the hot channel power distribution is such that the minimum DNB ratio is greater than the applicable limit and the linear heat rate is less than 22.7 kW/ft. For any normal steady state operating condition, the maximum linear heat rate does not exceed 6.64 x F_Q kW ft, where F_Q is the maximum value dictated by the Core Operating Limits Report (COLR).

Potential axial xenon oscillations are controlled with the control rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

Fuel Enrichment Limits

Detailed nuclear analysis (refer to Reference 65 of Section 3.2) has been completed to demonstrate that the existing spent fuel storage racks can safely store fuel with initial enrichments up to 5.0 w/o U-235 provided they are done so as specified in Figures 9.5-2A, 9.5-2B, and 9.5-2C.

Control Bank Insertion Limits

The control bank insertion limits for D, C and B control banks were originally revised for Cycle 6 operation. The associated change in control bank insertion limits (refer to Reference 57 of Section 3.2) results in increased flexibility in core design, and a reduction in the calculated core peaking factor Fq at the bank insertion limit. The revised insertion limit curve for the current cycle is provided in the cycle-specific Core Operating Limits Report.

Reactivity Control Limits

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The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

1) A minimum hot shutdown margin as shown in the Technical Specifications is available assuming a 10% uncertainty in the control rod calculation (see Table 3.2-3).

2) This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.

3) The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

The reactor core was designed to meet the following limiting thermal and hydraulic criteria:

1) The minimum allowable DNBR during normal operation, including anticipated transients for the WRB-1 correlation is documented in Table 14.1-0. The minimum allowable DNBR for the W-3 correlation is 1.3 from 1000 to 2400 psia and 1.45 from 500 to 1000 psia.

2) Fuel temperature will not exceed 4700° F during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB) which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The ratio of the heat flux causing DNB at a particular core location, as predicted by the W-3 or WRB-1 correlation, to the existing heat flux at the same core location is the DNB ratio. The limiting DNB ratio corresponds to a 95% probability at a 95% confidence level that DNB does not occur and is chosen to maintain an appropriate margin to DNB for all operating conditions.

In connection with the functional elimination of the BIT, two hypothetical and one credible steamline break cases were either analyzed or evaluated:

Hypothetical steamline breaks:

1) Steampipe severance, downstream of the flow restrictor, with offsite power available;

2) Steampipe severance, downstream of the flow restrictor, without offsite power available;

Credible streamline break:

1) A failed secondary safety or relief valve, with offsite power available.

The DNB analyses show that the DNB design basis is met for the hypothetical steamline break with offsite power, and that no consequential fuel failures are anticipated. The hypothetical break without offsite power and the credible steamline break were evaluated and determined to be bounded by the results of the hypothetical steamline break with offsite power.

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Mechanical Limits

Reactor Internals

The reactor internal components were designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with Rod Cluster Control Assemblies. Core drop in the event of failure of the normal supports is limited so that the Rod Cluster Control Assemblies do not disengage from the fuel assembly guide thimbles.

The internals were further designed to maintain their functional integrity in the event of a major Loss-of-Coolant Accident. The dynamic loading resulting from the pressure oscillations because of a Loss-of-Coolant Accident does not cause sufficient deformation to prevent Rod Cluster Control assembly insertion.

Fuel Assemblies

The fuel assemblies were designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain and differential expansion during both steady state and transient reactor operating conditions have been considered in the design of the fuel rods and fuel assembly. The assembly was structurally designed to withstand handling and shipping loads prior to irradiation and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core, subsequent handling during cooldown shipment and fuel reprocessing.

The fuel rods are supported at seven locations along their length within the fuel assemblies by brazed grid assemblies which were designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods and without imposing restraints of sufficient magnitude to result in buckling or distortion of the rods.

The fuel rod cladding was designed to withstand operating pressure loads wtihout rupture and to maintain fuel integrity throughout design life.

Cycle 7 was the first cycle where Vantage 5 fuel was utilized. Features of this fuel include: debris filter bottom nozzle, reconstitutable top nozzle, six inch natural uranium axial blankets and the optimal use of integral fuel burnable absorber (IFBA) which incorporates a thin coating of ZrB2 on the outside cylindrical surface area of the fuel pellet.

Cycle 10 was the first cycle where Vantage + fuel was utilized. Features of this fuel included three intermediate flow mixers, enriched annular axial blankets, and low pressure drop midgrids, ZIRLO guide thimbles and midgrids. ZIRLO as a clad material was introduced in Cycle 9. Vantage + fuel also contains various high burnup features.

Beginning with Cycle 14, "15x15 upgrade" fuel was introduced to the core. See section 3.2.5.5 for details

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Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies (RCCA) are similar to those used for the fuel rod cladding. The cladding was designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included for the RCCA cladding thickness. Cladding of RCCA's is stainless steel.

Adequate clearance is provided between the absorber rods and the guide thimbles which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated without overheating of the absorber cladding. The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

Control Rod Drive Assembly

Each control rod drive assembly was designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure containing components were designed to meet the requirements of the ASME Code Section III. Nuclear Vessels, for Class A vessels.

The control rod drive assemblies for the full length rods provide rod cluster control assembly insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which are established to limit power-peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction. Also, the control rod drive assemblies for the full length rods provide a fast insertion rate during a "trip" of the RCCA which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system.

The part length control rod drive mechanisms (Reference 46) were used to position the part length rod control cluster assemblies within the reactor core. Subsequent to initial plant operation (during the Cycle 1 to Cycle 2 refueling outage), the part length RCCAs were removed from the reactor. Their drive mechanisms and position indication system were retired in place. Thimble plug devices were placed in the fuel assemblies where the part length rods had been.

3.2 REACTOR DESIGN

3.2.1 Nuclear Design and Evaluation

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters which are significant to design objectives. Throughout this section, Cycle 1 and Current Cycle parameters have been utilized in as much as they represent examples and/or somewhat typical values for initial and subsequent cycles. The capability of the reactor to achieve these objectives while performing safely under normal operational modes, including both transient and steady state, is demonstrated.

3.2.1.1 Nuclear Characteristics of the Design

A summary of the reactor nuclear design characteristics for Cycle 1 and the current cycle is presented in Table 3.2-1. Figures 3.2-2 through 3.2-12 provide core configuration information

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for Cycle 1 and are included for historical purposes. For current cycle information refer to the cycle specific Nuclear Parameters and Operations Package and the cycle specific Core Operating Limits Report.

Reactivity Control Aspects

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. The concentration of boric acid is varied as necessary during the life of the core to compensate for: (1) changes in reactivity which occur with change in temperature of the reactor coolant from cold shutdown to the hot operating, zero power condition; (2) changes in reactivity associated with changes in the fission product poisons xenon and samarium; (3) reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium); and (4) changes in reactivity due to burnable poison burnup.

The control rods provide reactivity control for: (1) fast shutdown; (2) reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level); (3) reactivity associated with any void formation; (4) reactivity changes associated with the power coefficient of reactivity.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling is the more restrictive of 2050 ppm or that required to be 5% subcritical. The boron concentration and the control rods together are required to provide at least 5 percent shutdown margin for these operations. Boron concentration requirements for refueling and Plant Startup (BOL) are found in Reference 72.

These boron concentrations are well within solubility limits at ambient temperature. Refueling concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial Cycle 1, full power boron concentration without equilibrium xenon and samarium was 1228 ppm. As fission product poisons were built up, the boron concentration was reduced to 899 ppm.

This initial boron concentration was that which permitted the withdrawal of the control banks to their operational limits. The Cycle 1 xenon-free hot, zero power shutdown (k=0.99) with all but the highest worth rod inserted, could be maintained with the boron concentration of 669 ppm. This concentration is less than the full power operating value with equilibrium xenon.

Control Rod Requirements

Neutron-absorbing control rods provide reactivity control to compensate for more rapid variations in reactivity. The rods are divided into two categories according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control group of rods. The remaining rods, which provide shutdown reactivity, are termed shutdown rods. The total shutdown worth of all the rods was also specified to provide adequate shutdown with the most reactive rod stuck out of the core.

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Control rod reactivity requirements at beginning- and end-of-life for Cycle 1 and the current cycle are summarized in Table 3.2-2. The installed worth of the control rods is shown in Table 3.2-3

The difference between required and installed worth is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change in incurred with a change in power level due to the Doppler and moderator temperature effects.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end of life when the moderator temperature coefficient has its most negative value. The moderator temperature coefficient range is given in Table 3.2-1, line 42, while the cumulative reactivity change is shown in the first line of Table 3.2-2. By the end of the fuel cycle, the nonuniform axial depletion causes a severe power peak at low power. The reactivity associated with this peak is part of the power defect.

Operational Maneuvering Band

The control group is operated at full power within a prescribed band of travel in the core to compensate for periodic changes in boron concentration, temperature, or xenon. The band has been defined as the operational maneuvering band. When the rods reach either limit of the band, a change in boron concentration must be made to compensate for any additional change in reactivity, thus keeping the control group within the maneuvering band.

The fully withdrawn bank position can vary within a few steps from the reference fully withdrawn condition from cycle to cycle.

The greatest depth in the core to which the control banks may be inserted is known as the insertion limit. The maximum insertion limit on which the FSAR transient analyses are based is 23.5 percent D-Bank insertion, which corresponds to a D-Bank position of 176 steps at 100 percent power. The actual limit may be administratively decreased via a COLR change, to facilitate core design without having to reanalyze the FSAR transients.

Control Rod Bite

If sufficient boron is present in a chemically-shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shut down safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods must be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of five percent per minute, or by a step load change of ten percent. An insertion rate of 4×10^{-5} Δp per second is

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determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate, one control bank of rods had to remain inserted at least 13 percent into the core at the Cycle 1 beginning-of-life. The reactivity associated with this bite was 0.03 percent.

Indian Point 3 is analyzed for continuous operation at the bite position, at no bite position (i.e., ARO) and for any rod position in between (85).

Xenon Stability Control

The control rods are capable of suppressing xenon induced power oscillations in the axial direction, should they occur. Out-of-core instrumentation was provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced power oscillations. Extensive analyses, with confirmation of methods by spatial transient experiments at Haddam Neck, have shown that any induced radial or diametral xenon transients would die away naturally.[2] A full discussion of axial xenon stability control can be found in Reference 3.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing 1.3 percent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam line break accident is considered. The excess control available at the Cycle 1 end-of-cycle, hot zero power condition with the highest worth rod stuck out, allowing a 10% margin for uncertainty in control rod worth is shown in Table 3.2-3.

Calculated Rod Worths

The complement of 53 full length control rods, arranged in the pattern shown in Figure 3.2-1 meets the shutdown requirements. Table 3.2-3 lists the calculated worths of this rod configuration for beginning-of-life, and end-of life, for Cycles 1 and the current cycle.

In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed were decreased in the design by 10 percent to account for any errors or uncertainties in the calculation. This worth was established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in operating reactors shows the calculation to be well within the allowed uncertainty of 10%.

Reactor Core Power Distribution

The accuracy of power distribution calculations was initially confirmed through approximately one thousand flux maps during some twenty years of reactor operation under conditions very similar to those which were expected for Indian Point 3. Details of this confirmation are given in Reference 5.

Definitions

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Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel, and are expressed in terms of quantities related to the nuclear or thermal design, namely:

Power density is the thermal power produced per unit volume of the core (kW/liter).

Linear power density is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes it differs from (kW/liter) by a constant factor which includes geometry and the fraction of the total thermal power which is generated in the fuel rod.

Average linear power density is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

Local heat flux is the heat flux at the surface of the cladding (Btu/ft^2/hr). For nominal rod parameters this differs from linear power density by a constant factor.

Rod power or rod integral power is the length integrated linear power density in one rod (kW).

Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming all rods have equal length).

The hot channel factors used in the discussion of power distributions in this section are defined as follows:

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod parameters.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad.

Combined statistically, the net effect of the hot channel factors has a value of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Manufacturing tolerances, hot channel power distribution and surrounding channel power distributions are treated explicitly in the calculation of the Departure from Nucleate Boiling Ratio.

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It is convenient for the purposes of discussion to define subfactors of F_Q. However, design limits are set in terms of the total peaking factor.

F_Q = Total peaking factor or heat flux hot-channel factor
= (Maximum kW/ft) / (Average kW/ft)

without densification effects

$$F_Q = F_Q^N \times F_Q^E$$
$$= F_{xy}^N \times F_Z^N \times F_U^N \times F_Q^E$$

where:

F_Q^N and F_Q^E are defined above.

F_U^N = factor for conservatism, assumed to be 1.05.

F_{XY}^N = ratio of peak power density to average power density in the horizontal plans of peak local power.

F_Z^N = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height, then is the core average axial peaking factor.

To include the allowances made for densification effects, which are height dependent, the following quantities are defined:

S(Z) = the allowance made for densification effects, at height Z in the core.

P(Z) = ratio of the power per unit core height in the horizontal plane at height Z to the average value of power per unit core

Then:

F_Q = Total peaking factor

= (Maximum kW/ft) / (Average kW/ft)

Including densification allowance,

$$F_Q = \max_{on\ z}\left\{\left[F_{XY}^N(Z)\right]\left[P(Z)\right]\left[S(Z)\right]\right\}\left(F_U^N\right)\left(F_Q^E\right)$$

Results reported in Reference 83 show that current fuel designs manufactured by Westinghouse are highly stable with respect to fuel densification and that the formation of small gaps in the axial pellet columns are very infrequent events. Reference 83 concludes that a densification power spike factor (S(Z)) of 1.0 is appropriate for current Westinghouse fuel, effectively eliminating this penalty from the safety analysis.

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Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel and burnable poison loading patterns, and the presence or absence of a single bank of control rods. Thus, at any time in the cycle a horizontal section of the core can be characterized as: 1) unrodded, or 2) with group D control rods. These two situations, combined with burnup effects, determine the radial power shapes which can exist in the core at full power. The effect on radial power shapes of power level, xenon, samarium and moderator density are considered also, but they are quite small. The effect of non-uniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, the core radial enthalpy rise distribution, as determined by the integral of power up each channel, is of greater interest. Figure 3.2-2 through 3.2-4 show representative Cycle 1 radial power distributions for one eighth of the core for representative operating conditions. The conditions are:

1) Hot Full Power (HFP) – Beginning-of-Life (BOL) – unrodded – no xenon.

2) HFP – End-of-Life (EOL) – unrodded – equilibrium xenon

3) HFP – BOL – Bank D in – equilibrium xenon

Since the position of the hot channel varies from time to time, a single reference radial design power distribution is selected for DNB calculations. This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power.

Axial Power Distributions

The shape of the power profile in the axial or vertical directions is largely under the control of the operator through either the manual operation of the control rods or automatic motion of the rods, responding to manual operation of the soluble boron system. Nuclear effects which cause variations in the axial power shape include: moderator density, Doppler effect on resonance absorption, spatial xenon and burnup. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time. Signals are available to the operator from the excore ion chambers (which are long ion chambers outside the reactor vessel running parallel to the axis of the core). Separate signals are taken from the top and bottom halves of the chambers. The difference between top and bottom signals from each of four pairs of detectors is displayed on the control panel and called the flux difference, ΔI. Calculations of the core average peaking factor for many plants, and measurements from operating plants under many operating situations, are associated with either I or axial offset in such a way that an upper bound can be placed on the peaking factor. For these correlations, axial offset is defined as:

Axial Offset = $(\Phi_t - \Phi_b)/(\Phi_t + \Phi_b)$

where: Φ_t and Φ_b are the top and bottom detector readings.

Local Power Peaking

Fuel densification, which has been observed to occur under irradiation in several operating reactors, causes the fuel pellets to shrink both axially and radially. The pellet shrinkage

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combined with random hang-up of fuel pellets results in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z) where Z is the axial location in the core. The method used to compute the power spike factor is described in Reference 6.

Results reported in Reference 83 show that current fuel designs manufactured by Westinghouse are highly stable with respect to fuel densification and that the formation of small gaps in the axial pellet columns are very infrequent events. Reference 83 concludes that a densification power spike factor (S(Z)) of 1.0 is appropriate for current Westinghouse fuel, effectively eliminating this penalty from the safety analysis.

Limiting Power Distributions

According to the ANS classification of plant conditions, Condition I occurrences are those which are expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action, Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation.

Implicit in the definition of normal operation is proper and timely action by the reactor operator. That is, the operator follows recommended procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation. Thus, as stated above, the worst or limiting power distribution which can occur during normal operation is to be considered as the starting point for analysis of Condition II, III, and IV events.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (Condition II). Therefore, the limiting power shapes which result from such Condition II events are those power shapes which deviate from the normal operating condition at the recommended axial offset band, e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power shapes which fall in this category are used for determination of the reactor protection system setpoints so as to maintain margin to overpower or DNB limits.

The means for maintaining power distributions within the required hot channel factor limits are described in the Technical Specifications. A complete discussion of power distribution control in Westinghouse PWR's is included in Reference 8. Detailed background information on the design constraints on local power density in a Westinghouse PWR, on the defined operating procedures, and on the measures taken to preclude exceeding design limits is presented in the Westinghouse topical report on power distribution control and load following procedures.[9] The following paragraphs summarize these reports [8,9] and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F^N_{\Delta H}$, include all of the nuclear effects which influence the radial and/or axial power distributions throughout

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core life for various modes of operation, including load follow, reduced power operation, and axial xenon transients.

Radial power distributions are calculated for the full power condition, and fuel and moderator temperature feedback effects are included in these calculations. The steady state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on radial power distribution is small but is included as part of the normal design process. Radial power distributions are relatively fixed and easily bounded with upper limits.

The core average axial profile, however, can experience significant changes which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the plant parameters which are readily observed. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

1) Core power level
2) Core height
3) Coolant temperature and flow
4) Coolant temperature program as a function of reactor power
5) Fuel cycle lifetimes
6) Rod bank worths
7) Rod bank overlaps

Normal operation of the plant assumes compliance with the following conditions:

1) Control rods in a single bank move together with no individual rod insertion differing by more than the rod group alignment limits specified in the Technical Specifications.

2) Control banks are sequenced with overlapping banks

3) The control bank insertion limits are not violated

4) Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures which are followed in normal operation. Briefly, they require control of the axial offset (flux difference divided by fractional power) at all power levels within a permissible operating band of a target value corresponding to the equilibrium full power value. In the first cycle, the target value changed from about +10 to –3 percent linearly through the life of the cycle. This minimized xenon transient effects on the axial power distribution, since the procedures essentially kept the xenon distribution in phase with the power distribution.

Calculations are performed for normal operation of the reactor including load following maneuvers. Beginning, middle and end of cycle conditions are included in the calculations. Different histories of operation are assumed prior to calculating the effect of load follow transients on the axial power distribution. These different histories assume base loaded

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operation and extensive load following. For a given plant and fuel cycle, a finite number of maneuvers are studied to determine the general behavior of the local power density as a function of core elevation.

These cases represent many possible reactor states in the life of one fuel cycle, and they have been chosen as sufficiently definitive of the cycle by comparison with much more exhaustive studies performed on some 20 or 30 different, but typical, plant and fuel cycle combinations. The cases are described in detail in Reference 9 and they are considered to be necessary and sufficient to generate a local power density limit which, when increased by 5 percent for conservatism, will not be exceeded with a 95 percent confidence level. Many of the points do not approach the limiting envelope. However, they are part of the time histories which lead to the hundreds of shapes which do define the envelope. They also serve as a check that the reactor studied is typical of those more exhaustively studied.

Thus, it is not possible to single out any transient or steady-state condition which defines the most limiting case. It is not even possible to separate out a small number which form an adequate analysis. The process of generating a myriad of shapes is essential to the philosophy that leads to the required level of confidence. A maneuver which provides a limiting case for one reactor fuel cycle is not necessarily a limiting case for another reactor or fuel cycle with different control bank worths, enrichments, burnups, coefficients, etc. Each shape depends on the detailed history of operation up to that time and on the manner in which the operator conditioned xenon in the days immediately prior to the time at which the power distribution is calculated.

The calculated points are synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes in the first cycle. In these calculations, the effects on the unrodded radial peak of xenon redistribution that occurs following the withdrawal of a control bank (or banks) from a rodded region is obtained from two-dimensional X-Y calculations. A 1.03 factor to be applied on the unrodded radial peak was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of this effect may be found in Reference 9. The calculated values have been increased by a factor of 1.05 for conservatism, and by a factor of 1.03 for the engineering factor F_Q^E.

The envelope drawn over the calculated (max F_Q x Power) points in the COLR represents an upper bound envelope on local power density versus elevation in the core. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements. Additionally, s 3.2-5 are based on a radial power distribution invariant with core elevation. Finally, as previously discussed, this upper bound envelope is based on procedures of load follow which require operation within an allowed deviation from a target equilibrium value of axial flux difference. These procedures are detailed in the Technical Specifications and are followed by relying only upon excore surveillance supplemented by the normal full core map requirement at every effective full power month, and by computer based alarms or manual logging on deviation and time of deviation from the allowed flux difference band.

To determine reactor protection system setpoints with respect to power distributions, three categories of events are considered, namely rod control equipment malfunctions, operator errors of commission and operator errors of omission. In evaluating these three categories of events, the core is assumed to be operating within the four constraints described below:

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1) Control rods in a single bank move together with no individual rod insertion differing by more than the rod group alignment limits specified in the Technical Specifications.

2) Control banks are sequenced with overlapping banks

3) The control bank insertion limits are not violated

4) Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence). Also included are motions of the banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions are calculated throughout these occurrences assuming short-term corrective action, that is, no transient xenon effects are considered to result from the malfunction. The event is assumed to occur from typical normal operating situations which include normal xenon transients. It is further assumed in determining the power distributions, that total core power level will be limited by reactor trip to below 120 percent. Since the study is to determine protection limits with respect to power and axial offset, no credit is taken for trip setpoint reduction due to flux difference. The peak power density which can occur in such events, assuming reactor trip at or below 120 percent, is less than that required for centerline melt, including uncertainties and densification effects.

The second category assumes that the operator mispositions the rod bank in violation of the insertion limits and creates short term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a flux difference violation. The resulting F_Q is multiplied by an appropriate allowance for calorimetric error. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error.

Analyses of possible operating power shapes show that the appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values addressed in the COLR.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits. It has been determined that, provided the above conditions 1) through 4) are observed, the COLR limits are met.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by

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means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection Systems.

Moderator Temperature Coefficient

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to the coefficient, and in a chemical shim core the rods are only partially inserted. Also, the chemical poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to boron being removed from the core. This is directly proportional to the amount of reactivity controlled by the dissolved poison.

In order to reduce the dissolved poison requirement for control of excess reactivity, burnable poison rods have been incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of poison, and the moderator temperature coefficient will be reduced.

The burnable poison for Cycle 1 was in the form of borated Pyrex glass rods clad in stainless steel. There were 1434 of these borated Pyrex glass rods in the form of clusters distributed throughout the initial core in vacant rod cluster control guide tubes, as illustrated in Figures, 3.2-6 through 3.2-7. Information regarding research, development and nuclear evaluation of the burnable poison rods can be found in Reference 1. These rods initially controlled 10% $\Delta\rho$ of the installed excess reactivity and their insertion into the core resulted in a reduction of the initial hot zero power boron concentration in the coolant to 1330 ppm. The moderator temperature coefficient is negative at operating conditions with burnable poison rods installed. Subsequent cycles utilized B_4C in AL_20_3 in wet annular burnable absorbers (WABA) and ZrB_2 in Integral Fuel Burnable Absorbers (IFBA) and also the already mentioned Pyrex rods.

The effect of burnup on the moderator temperature coefficient is periodically calculated, and the coefficient becomes more negative with increasing burnup. This is due to the buildup of fission products with burnup, and dilution of the boric acid concentration with burnup. The reactivity loss due to equilibrium xenon is controlled by boron, and as xenon builds up boron is taken out. The calculated net effect and the predicted moderator temperature coefficient at equilibrium xenon for Cycle 1 and at full power BOL was – 0.84 x 10-4/°F. With core burnup, the coefficient became more negative as boron was removed because of a shift in the neutron energy spectrum due to the buildup of plutonium and fission products. At Cycle 1 EOL with no boron or rods in the core, the moderator coefficient was -3.5 x 10^{-4}/°F. Reference 72 provides the Cycle 14 moderator coefficient.

Moderator Pressure Coefficient

The moderator pressure coefficient has an opposite sign to the moderator temperature coefficient. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient, a change of 50 psi in pressure having no more effect on reactivity than half-degree change in moderator temperature. The calculated Cycle 1 BOL and EOL pressure coefficients are specified in Table 3.2-1, Line 43.

Moderator Density Coefficient

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A uniform moderator density coefficient is defined as a change in the neutron multiplication* per unit change in moderator density. The range of the moderator density coefficient for Cycle 1 from BOL and EOL is specified in Table 3.2-1, Line 44.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication* per degree change in fuel temperature. The coefficient was obtained in the past by calculating neutron multiplication as a function of effective fuel temperature by the code LEOPARD.[4] The results for Cycle 1 are shown in Figure 3.2-11. More recent calculations of Doppler and power coefficients are performed using the ANC(84) code.

> *NOTE: Neutron multiplication is defined at the ratio of the average number of neutrons produced by fission in each generation to the total number of corresponding neutrons absorbed.

In order to know the change in reactivity with power, it is necessary to know the change in the effective fuel temperature with power as well as the Doppler coefficient. It is very difficult to predict the effective temperature of the fuel using a conventional heat transfer model because of uncertainties in predicting the behavior of the fuel pellets. Therefore, an empirical approach is taken to calculate the power coefficient, based on operating experience of existing Westinghouse cores. Figure 3.2-12 shows the power coefficient as a function of power for Cycle 1 obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

3.2.1.2 Nuclear Evaluation

The basis for confidence in the procedures and design methods comes from the comparison of these methods with many experimental results. These experiments include criticals performed at the Westinghouse Reactor Evaluation Center (WREC) and other facilities, and also measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in other Safety Analysis Reports such as the FSAR for Indian Point 2, Docket No. 50-247.

3.2.1.3 Enrichment Error

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes that are more peaked than those calculated with the correct enrichments. There is an 8% uncertainty margin between the calculated worst value and the design value of power peaking assumed for the analysis of normal steady state operation and anticipated transients. The incore system of moveable flux detectors, used to verify power shapes at start of life, is capable of revealing any enrichment error or loading error which causes power shapes to be peaked in excess of the design value. These power shape measurements are taken at low power when extremely adverse power shapes can be tolerated.

An analysis of the effect of an inadvertent loading of an assembly with an enrichment increased by 20% over the nominal value, showed that the error was detectable at many of the detector locations in the core. In the case of a centrally placed assembly with this enrichment error, five flux detectors would show a signal more than 5% above the expected value. If the assembly bearing the enrichment error is placed off-center and as far from a flux detector as possible, the tilt caused by a 20% error in enrichment is detectable in more than half of the detector locations

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in the core, both as a flux increase over expected symmetric values and as a flux decrease on the opposite side of the core.

If the movable detector system fails to detect an enrichment error, then the power shapes are such that there is margin to the design conditions, and normal plant operation may be safely continued. It is incredible than any positive indication of power shape anomalies which are sufficiently large to cause a significant departure from design conditions, would be ignored.

These measurements are an integral part of the physics startup tests when considerable emphasis is placed on obtaining good power shape measurements.

These considerations, together with the fuel handling procedures described in Section 3.3, preclude power operation in the presence of any significant fuel enrichment error.

3.2.2 Thermal and Hydraulic Design and Evaluation

A large amount of material has been retained in this section as historical background. The thermal and hydraulic design parameters at the stretched power uprate conditions are provided in Table 3.2-4.

DNB Design Basis

There will be at least a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events), at a 95 percent confidence level. Historically, this has been conservatively met by adhering to the following thermal design basis: there must be at least a 95 percent probability that the minimum departure from nucleate boiling ration (DNBR) of the limiting power rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.

DNB Analysis Method

The Westinghouse version of the VIPRE-01 (VIPRE) code was used to perform the thermal/hydraulic calculations for both the mini-uprate and stretched power uprate programs. The VIPRE code is equivalent to the THINC-IV (THINC) code and has been approved by the NRC for licensing applications to replace the THINC code. The use of VIPRE is in full compliance with the conditions specified in the NRC Safety Evaluation Report (SRE) on WCAP-14565-P-A[78].

The design method employed for both fuel types to meet the DNB design basis is the Revised Thermal Design Procedure (RTDP)[79]. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2)

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To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow and potential transition core, the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs result in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operation and design flexibility. The Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety input values to give the lowest minimum DNBR. The limit for STDP is appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

For this design, the WRB-1 correlation is used for analysis of the VANTAGE+ fuel assemblies with a correlation limit of 1.17 (both typical and thimble cells). When the core condition is outside the range of the WRB-1 correlation, the W-3 correlation is applied with a correlation limit of 1.30 (both cell types) with pressure greater than 1000 psia.

DNB With Physical Burnout

Westinghouse [29] has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod. After this occurrence, the 25 rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operation were observed.

DNB With Return to Nucleate Boiling

Additional DNB tests have been considered by Westinghouse[30] in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundle for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to re-establish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

Hydrodynamic and Flow Power Coupled Instability

The interaction of hydrodynamic and spatial effects has been considered and it is concluded that a large margin exists between the design conditions and those for which an instability is possible.

Heated channels in parallel can lead to flow instability. If substantial boiling takes place, periodic flow instabilities have been observed and, as long ago as 1938, Ledinegg[24] proposed a stability criterion on the basis of which the concept of inlet orificing has been developed to stabilize flow. More recent work[25-27] has demonstrated that periodic instabilities are possible which violate the Ledinegg criterion.

In normal flow channels with little or no boiling, the type of instability proposed by Ledinegg is not possible since it results primarily from the large changes in water density along the channel due to boiling. Moreover, the periodic instabilities examined by Quandt[25-26] and Meyer[27] are not exhibited in non-boiling channels of the type found in PWR cores. [28]

3.2.2.1 Thermal and Hydraulic Characteristics of the Design

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Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The surface temperature of the pellet is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains maximum cladding surface temperature below about 657 °F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap [11, 12], and may be calculated by the following equation:

$$h = 0.6P + K/ f(14.4 \times 10^{-6})$$

where

h = conductance in Btu/hr-ft^2-°F
P = contact pressure in psi
k = the thermal conductivity of the gas mixture in the rod
f = the correction factor for the accommodation coefficient

The thermal conductivity of uranium dioxide was evaluated from published results of work at ORNL [13] Chalk River [14], and WAPD. [15] The design curve for thermal conductivity is given in Figure 3.2-13. The section of the curve at temperatures between 0 °F and 3000 °F is based on the data of Godfrey, et al. [13]

The section of the curve between 3000 °F and 5000 °F was based on two factors:

1) Inpile observations of fuel melting dictate a positive temperature coefficient for conductivity above approximately 3000 °F, The temperature dependence in this range should conform to an exponential curve, since this reflects the most credible physical interpretation of the high temperature conductivity increase.

2) The area under the recommended curve is such that the integral is equal to approximately 97 w/cm as given by Robertson, et at [14] and Duncan. [15] This value is based upon the interpretation of fuel melt radius as determined at Hanford[16] and Chalk River. [14]

Thermal conductivity can be represented best by the following equation:

$$k = (11.8 + 0.0238T)^{-1} + 8.775\times10^{-13}T^3$$

with k in w/cm- °C for 95 percent theoretical density and T in °C.

Based upon the above considerations, the maximum central temperature of the hot pellet at steady state nominal and overpower conditions are shown in Table 3.2-4. The temperature is well below the melting temperatures of the irradiated UO_2 (Refer to Criterion 6 in section 3.1.2), which is taken as the unirradiated fuel temperature of 5080°F[17] decreasing by 58°F per 10,000 MWd/metric ton uranium and covering the manufacturing/modeling uncertainties.

Fuel Thermal Performance

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Ability to predict fuel performance at high burnup is based on both published data and proprietary data. Effects of irradiation on UO_2 melting point, fuel swelling, fission gas release, clad creep, clad yield strength, etc., have been incorporated into a computer program to enable the prediction of fuel performance. The proof of performance was determined by the Saxton and Zorita programs. Saxton fuel test rods operated to about 42,000 MWD/MTM peak burnup, and Zorita fuel test rods operated to about 32,000 MWD/MTM, have been used in verifying performance models. In 1972, the Saxton lead rod accumulated additional burnup to about 53,000 MWD/MTM at the end of the Core III operation. Zorita test rods were exposed to about 45,000 MWD/MTM at the end of second cycle operation in mid 1972. In addition, special removable rod assemblies have been irradiated in the Zion, Surry Unit 1 and 2, and Trojan plants. These assemblies have provisions for removable rods which permit non-destructive examination of fuel rods during refueling shutdowns. Two Zion 15 x 15 high burnup test assemblies have been irradiated for five fuel cycles and have achieved assembly burnups of about 55,000 MWD/MTM. Profilometry measurements after cycle Nos. 1, 2, 3, and 4 have shown less fuel swelling and outward cladding strain than predicted by Westinghouse models. Examination of removable rod 17 x 17 assemblies irradiated in Surry Units 1 and 2 for up to four cycles have shown that cladding creep down is somewhat less than predicted by Westinghouse models.

Applying best estimate models to Indian Point 3 Region III peak burnup rod, the cladding strain damage limit was calculated to be reached at a power level greater than 21 kW/ft.

In the calculation of the steady-state performance of a nuclear fuel rod, the following interacting factors were considered:

1) Cladding creep and elastic deflection

2) Pellet swelling, thermal expansion, gas release, and thermal properties as a function of temperature

3) Internal pressure as a function of fission gas release, rod geometry, and temperature distribution.

These effects have been combined in a computer code which is considered to be Westinghouse proprietary information. With these interacting factors considered, the code determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and cladding temperatures, and cladding deflections are calculated. The fuel rod is divided lengthwise into several sections and radially into a number of annular zones. Fuel density changes, cladding stresses, strains and deformations, and fission gas releases are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The gap conductance between the pellet surface and the cladding inner diameter is calculated as a function of the composition and pressure of the gas mixture, and the contact pressure between clad and pellet. After computing the fuel temperature for each pellet's annular zone, the fractional fission gas release is assessed from the diffusion-trapping model described by Weisman, et al.[31] The total amount of gas released is based on the average fractional release within each axial increment and the gas generation rate which in turn is a function of burnup. Finally, the gas released is summed over all axial increments and the pressure is calculated.

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The code shows good agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures and clad deflections. Included in this spectrum are variations in power, time, fuel density, and geometry.

The worst strain conditions in the fuel rod occur at high burnup after the clad and fuel have reached mechanical equilibrium (cladding and fuel in intimate contact). At this time in life, the tolerances have negligible effect and therefore the linear heat rating is calculated using nominal dimensions.

The new PAD fuel rod performance code used to analyze VANTAGE + reloads contained an improved gap conductance model that calculates lower fuel temperatures. The new PAD code was approved by the NRC[80]. Fuel rod thermal evaluations are performed at rated power, maximum overpower, and during transients at various burnups for the stretched power uprate program. These analyses assure that design criteria for reactor core design (criterion 6) given in section 3.1.2 are met.

3.2.2.2 Westinghouse Experience With High Power Fuel Rods

A completed high power test program had the objective of defining failure limits for the combined effects of linear heat generation rate and burnup, providing increased assurance that plants have adequate performance and design margins to the fuel failure threshold, and verifying the adequacy of design methods and computer codes. Results from this program are given in Section 8 of Reference 50. Additional information on fuel rod experience is presented in Reference 51. A summary of the comprehensive experimental program to extend the operating experience to higher power and to higher exposures for fuel rods is provided in Figure 3.2-14.

The figure shows that thirty Saxton Plutonium Project non-pressurized fuel rods have operated at a design peak power level of up to 18.5 kW/ft to a peak exposure of approximately 30,000 MWD per MTM [Megawatt days per metric ton of metal (U + PU)]. No failures have occurred with this fuel. In the Saxton overpower test, two selected fuel rods from the Saxton Plutonium Project assemblies were removed after peak exposure of 18,000 MWD/MTM and inserted in a subassembly for short time irradiation at a design rating of 25 kW/ft. Results of this program indicate satisfactory performance of the fuel in every respect. The Saxton Plutonium Project was extended by irradiating approximately 250 rods to peak burnups of about 50,000 MWD/MTM at design linear power levels ranging form 9.5 to 23.6 kW/ft.

In the above tests (performed on non-pressurized rods), the strain fatigue experienced by the cladding was more severe than expected to occur for pressurized rods which would be placed under identical operating conditions.

Internally pressurized fuel rods have been under investigation (19) at Westinghouse for several years. These investigations include out-of-pile and in-pile experimental programs and analytical studies. Fuel rods internally pressurized with various gases have been irradiated in the Saxton reactor. Tests results show that initial pressurization is effective in substantially reducing the rate of cladding-creep on to the UO_2 fuel. The Saxton test results confirm the results of analyses which predict fuel-cladding mechanical interaction early in life for non-pressurized fuel rods and delayed interaction for initially pressurized fuel rods.

To verify the substantial design margin which exists in the fuel rods with regard to excessive internal pressures in a fuel rod, several highly pressurized Zircaloy-clad fuel rods were irradiated for several months in the Saxton reactor, then removed for examination. At an internal pressure

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of approximately 3500 psia, the fuel operated satisfactorily for the period of the test without any indication of failure. Two fuel rods, deliberately tested at unrealistically high internal pressures, experienced clad cracking but operated satisfactorily for the period of the test. Thus, even with excessive internal pressures that result in clad failures, the test results are favorable.

3.2.2.3 Sizing of Fuel Rod Plenum

The criterion for sizing the fuel rod plenum length was that the internal gas pressure is limited to a value below which could cause: 1) the diametral gap to increase due to outward cladding creep during steady state operation, and 2) extensive DNB propagation to occur. During operational transients, fuel rod clad rupture due to high internal gas pressure is precluded by meeting the above design basis. The end of life internal gas pressure depends on the initial gas pressure, void volumes (plenum, gap, dish, open porosity, etc.), the amount of fission gases released, and the amount of helium released from IFBA fuel. The estimated fraction of fission gases present in the gap and the plenum was about 20% for the maximum burnup rod (limiting case evaluation) at the end of three cycles of reactor operation.[31]

For the lead rod, the calculated internal pressure is less than the limit value; the clad stress is less than the yield strength of Zircaloy; the clad strain is less than 1% at end of life at normal operating conditions. The ability of fuel to withstand expected transients at the end of life has been evaluated. Such phenomena as high internal gas pressure have been studied under transient conditions and do not present any particular problem or significantly influence fuel behavior during transients.

The limiting criterion for evaluating mechanical performance for power transients is presently conservatively assumed to be yield stress. Both slow and rapid transients have been investigated. Slow transients, e.g., those caused by Xenon oscillations or local power shifts due to depletion, are generally of low magnitude and are of no concern since the resulting stresses will be small and will relax due to creep characteristics of the UO_2 and Zircaloy.

Rather rapid transients and transients of sufficient magnitude to cause high clad stresses could arise from some accidents such as major steam line break accident. Calculations indicate that the yield strength of the cladding could be exceeded in some fuel rods which would experience large power increase during this accident. The clad yield stress will also be exceeded during a loss of coolant accident because the yield strength of Zircaloy decreases to a very small value at high temperatures expected with this accident. It is not possible to preclude some rods from failure during severe transients. However, this is consistent with the plant design basis.

3.2.2.4 Heat Flux Ratio and DNB Correlation

WRB-1 Correlation

The DNB heat flux ration (DNBR) as applied to typical cells (flow cells with walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}}$$

Where:

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$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

Where:

$q''_{DNB,EU}$ = the uniform heat flux as predicted by the WRB-1 DNB correlation (Ref. 58)

F = the flux shape factor to account for nonuniform axial heat flux distributions (Reference 18) with the "C" term modified as in Reference 32

q''_{loc} = the actual local heat flux.

The WRB-1 (Reference 58) correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data which has better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of variables is:

Pressure: $1440 \le P \le 2490$ psia

Local Mass Velocity: $0.9 \le \frac{G_{loc}}{10^6} \le \frac{3.7 lb}{ft^2 - hr}$

Local Quality: $-0.2 \le X_{loc} \le 0.3$

Heated Length, inlet to CHF Location: $L_h \le 14$ feet

Grid Spacing: $13 \le g_{sp} \le 32$ inches

Equivalent Hydraulic Diameter: $0.37 < d_e < 0.60$ inches

Equivalent Heated Hydraulic Diameter: $0.46 < d_h < 0.58$ inches

Figure 3.2-15A shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

As documented in References 81 and 82, a 95/95 limit DNBR of 1.17 is appropriate for 15x15 VANTAGE+ fuel assemblies.

<u>The W-3 DNB Correlation</u>

The W-3 DNB correlation [18 and 32] is used where the primary DNB correlation is not applicable. The WRB-1 correlation is developed based on mixing vane data and therefore is only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident

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conditions where the system pressure is below the range of the primary correlation. For system pressure in the range of 500 to 1000 psia, the W-3 correlation is 1.45 [59]. For system pressure greater than 1000 psia, the W-3 correlation is limited to 1.30. A cold wall factor [33] is applied to the W-3 DNB correlation to account for the pressure of the unheated thermal surfaces.

Historical Information of W-3 Correlation

Departure from Nucleate Boiling (DNB) is predicted upon a combination of hydrodynamic and heat transfer phenomena and is affected by the local and upstream conditions, including the flux distribution. In reactor design, the heat flux associated with DNB and the location of DNB are both important. The magnitude of the local fuel rod temperature after DNB depends upon the axial location where DNB occurs.

The W-3 DNB correlation[18] was developed to predict the DNB flux and the location of DNB equally well for a uniform and an axially non-uniform heat flux distribution. This correlation replaced the preceding WAPD q" and H DNB correlations published in Nucleonics(20), May 1963, in order to eliminate the discontinuity of the latter at the saturation temperature and to provide a single unambiguous criterion for the design margin.

The W-3 correlation, and several modifications of it, have been used in Westinghouse critical heat flux (CHF) calculations. The W-3 correlation was originally developed form single tube data[32], but was subsequently modified to apply to the "L" –grid[33] rod bundle data. These modifications to the W-3 correlation have been demonstrated to be adequate for reactor rod bundle design. The W-3 DNB correlation[18] incorporates both local and system parameters in predicting the local DNB heat flux. This correlation includes the nonuniform flux effect and the upstream effect which includes inlet enthalpy or length. The local DNB heat flux ratio (defined as the ratio of the DNB heat flux to the local heat flux) is indicative of the contingency available in the local heat flux without reaching DNB. The sources of the data used in developing this correlation included:

WAPD-188	1958	CU-TR-NO. 1 (NW-208)	1964
ASME PAPER 62-WA-297	1962	CISE-R-90	1964
CISE-R-63	1962	DP-895	1964
ANL-6675	1962	AEEW-R-356	1964
GEAP-3766	1962	BAW-3238-7	1965
AEEW-R213 AND 309	1963	AE-RTL-778	1965
CISE-R-74	1963	AEEW-355	1965
CU-MPR-XIII	1963	EUR-2490.e	1965

The comparison of the measured to predicted DNB flux of this correlation is given in Figure 3.2-15. The local flux DNB ratio versus the probability of not reaching DNB is plotted in Figure 3.2-16. This plot indicates that with a DNBR of 1.3 the probability of not reaching DNB is 95% at a 95% confidence level.

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Rod bundle data without mixing vanes agreed very well with the predicted DNB flux as shown in Figure 3.2-17 and rod bundle data with mixing vanes (Figure 3.2-18 show, on the average, an 8% higher value of DNB heat flux than predicted by the W-3 DNB correlation.

It should be emphasized that the inlet subcooling effect of the W-3 correlation was obtained from both uniform and non-uniform data. The existence of an inlet subcooling effect has been demonstrated to be real and hence the actual subcooling should be used in the calculations. The W-3 correlation was developed from tests with flow in tubes and rectangular channels. Good agreement is obtained when the correlation is applied to test data for rod bundles.

Departure from Nucleate Boiling Ratio

The DNB heat flux ratio (DNBR) as applied to the design when all flow cell walls are heated is:

$$\mathrm{DNBR} = \left(q''_{\mathrm{DNB,N}}\right)\left(F'_{s}\right)(0.986)\Big/q''_{\mathrm{loc}}$$

where:

$$q''_{\mathrm{DNB,N}} = \left(q''_{\mathrm{DNB,EU}}\right)\Big/F$$

and:

$q''_{\mathrm{DNB,EU}}$ is the uniform DNB heat flux as predicted by the W-3 DNB correlation (18) when all flow cell walls are heated.

F is the flux shape factor to account for non-uniform axial heat flux distributions [18] with the "C" term modified as in Reference 32.

F'_{s} is the modified spacer factor which uses an axial grid spacing coefficient, KS = 0.046, and a thermal diffusion coefficient, TDC = 0.019, based on the 26-inch grid spacing data.

q''_{loc} is the actual local heat flux.

The DNBR as applied to this design when a cold wall is present is:

$$\mathrm{DNBR} = \left(q''_{\mathrm{DBN,N,CW}}\right)\left(F'_{S}\right)(0.986)\Big/q_{\mathrm{loc}}$$

where:

$$q''_{\mathrm{DNB,N,CW}} = \left(q''_{\mathrm{DBN,EU,Dh}}\right)(\mathrm{CWF})\Big/F$$

$q''_{\mathrm{DBN,EU,Dh}}$ is the uniform heat flux as predicted by the W-3 cold wall DNB correlation[33] when not all flow cell walls are heated (thimble cold wall cell).

CWF = Cold Wall Factor

Local Non-Uniform DNB Flux

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The local non-uniform $q''_{DNB,N}$ is calculated as follows

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

Where:

$$F = \frac{C}{q''_{local\_at\_\ell_{DNB}} * (1 - e^{-C\ell_{DNB}})} \int_0^{\ell_{DNB}} q''(z) e^{-C(\ell_{DNB} - z)} dz \qquad (5)$$

ℓ_{DNB} = distance from the inception of local boiling to the point of DNB, in inches.

Z = distance from the inception of local boiling measured in the direction of flow, in inches.

The empirical constant, C, as presented in Reference 18 has been updated through the use of more recent non-uniform DNB data. However, the revised expression does not significantly influence (less than once percent deviation from that of Reference 18) the value of the F-factor and the DNBR. It does provide a better prediction of the location of DNB. The new expression is:

$$C = 0.15 \exp[\frac{(1 - X_{DNB})^{4.31}}{(G/10^6)^{0.478}}] \text{ (in)}^{-1} \qquad (6)$$

Where

G = mass velocity, $\frac{lb}{hr - ft^2}$

X_{DNB} = quality of the coolant at the location where DNB flux is calculated.

In determining the F-factor, the value of q" local at DNB in equation (5) was measured as $Z = \ell_{DNB}$, the location where the DNB flux is calculated. For a uniform flux, F becomes unity so that $q''_{DNB,N}$ reduces to $q''_{DNB,EU}$ as expected. The comparisons of predictions by using W-3 correlations and the non-uniform DNB data obtained by B&W [21], Winfrith [22 & 23] and Fiat are given in Figure 3.2-19 and 3.2-20. The criterion for determining the predicted location of DNB is to evaluate the ratio of the predicted DNB flux to the local heat flux along the length of the channel. The location of the minimum DNB ratio is considered to be the location of DNB.\

Procedure for Using W-3 Correlation

In predicting the local DNB flux in a non-uniform heat flux channel, the following two steps are required:

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1) The uniform DNB heat flux, $q''_{DNB,EU}$, is computed with the W-3 correlations using the specified local reactor conditions.

2) This equivalent uniform heat flux is converted into corresponding non-uniform DNB heat flux, $q''_{DNB,N}$, for the non-uniform flux distribution in the reactor. This is accomplished by dividing the uniform DNB flux by the F-factor. [8] Since F is generally greater than unity $q''_{DNB,N}$ will be smaller than $q''_{DNB,EU}$.

To calculate the DNBR of a reactor channel, the values of $\left(q''_{DNB,N}/q''_{loc}\right)$ along the channel are evaluated and the minimum value is selected as the minimum DNBR incurred in that channel.

The W-3 correlation depends on both local and inlet enthalpies of the actual system fluid, and the upstream conditions are accommodated by the F-factor. Hence, the correlation provides a realistic evaluation of the safety margin of heat flux.

Application of the W-3 Correlation in Design

During steady state operation at the nominal design conditions, the DNB ratios are determined. Under other operating conditions, particularly overpower transients, more limiting conditions develop than those existing during steady state operation. The DNB correlations are sensitive to several parameters. In addition, thermal flux general under transient conditions is also sensitive to many parameters. Therefore, for each case studied, a conservative combination of the significant parameters is used as an initial condition. These parameters include:

a) Reactor coolant system pressure
b) Reactor coolant system temperature
c) Reactor power (determined from secondary plant calorimetrics)
d) Core power distribution (hot channel factors)

For transient accident conditions where the power level, system pressure and core temperature may increase, the DNBR is limited to a minimum value of 1.30. The Reactor Control and Protection System is designed to prevent any credible combination of conditions from occurring which would result in a lower DNB ratio.

For the W-3 correlation, the 95/95 limit DNBR is 1.30 at system pressure greater than or equal to 1000 psi. For lower pressure application (500-1000 psi), the 95/95 limit DNBR is 1.45 (Reference 59).

3.2.2.5 Film Boiling Heat Transfer Coefficient

Heat transfer after departure from nucleate boiling was conservatively assumed to be limited by film boiling immediately, and the period of transition boiling neglected.

The correlation used to evaluate these film boiling heat transfer coefficients was developed by Tong, Sandberg and Bishop [34] and is shown in Figure 3.2-21.

$$\left(hD/k\right)_f = 0.0193\left(DG/\mu\right)_f^{0.80}\left(C_p\mu/k\right)_f^{1.23}\left(p_g/p_b\right)^{0.68}\left(p_g/p_\ell\right)^{0.068-}$$

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where: $p_b = p_g a + p_\ell (1 - a)$

and

Cp = heat capacity at constant pressure, Btu/lb-F

D = equivalent diameter of flow channel, feet

H = heat transfer coefficient. Btu/hr-ft^2-F

G = mass flow rate, lb/hr-ft^2

k = thermal conductivity, Btu/hr-ft-F

a = void fraction

p = density, lbm/ft^3

μ = viscosity, lbm/ft-hr

Subscripts:

g = evaluation of the property at the saturated vapor condition

ℓ = evaluation of the property at the saturated liquid condition

f = evaluation of the property at the average film temperature

w = evaluation of the property at the wall temperature

b = evaluation of the property at the average bulk fluid condition.

The heat transfer correlation was developed for flow rates equal or greater than 0.8 x 10^6 lb/hr/sq ft over a pressure range of 580 to 3190 psia, for qualities as high as 100 percent, and heat flux from 0.1 to 0.65 x 10^6 Btu/hr/sq ft.

3.2.2.6 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux factors consider the local maximum at a point (the "hot spot" – maximum linear power densities), and the enthalpy rise factors involve the maximum integrated value along a channel (the "hot channel").

Definition of Engineering Hot Channel Factor

Each of the total hot channel factors is the product of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations from design conditions. The engineering hot channel factors account for the effects of flow conditions and fabrication tolerances and are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density and enrichment; inlet flow distribution; flow redistribution; and flow mixing.

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Heat Flux Engineering Subfactor, F_q^E

The heat flux engineering hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication variations for fuel pellet diameter, density and enrichment, and has a value of 1.03 at the 95 percent probability level with 95 percent confidence. No DNB penalty need be taken for the short relatively low intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.[35]

Enthalphy Rise Engineering Subfactor, $F_{\Delta H}^E$

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise in reload analysis is directly considered in the Westinghouse version of VIPRE-01 code (VIPRE)[77, 78] thermal subchannel analysis under any reactor operating condition. The items presently considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

Pellet diameter, density and enrichment.

Design values employed in the VIPRE analysis related to the above fabrication variations are based on applicable limiting tolerances such that these design values are met for 95 percent of the limiting channels at a 95 percent confidence level. Measured manufacturing data on Westinghouse fuel show that the tolerances used in this evaluation are conservative. In addition, each fuel assembly is checked to assure the channel spacing design criteria are met. The effect of variations in pellet diameter, enrichment and density is considered in establishing RTDP design limit.

Inlet Flow Maldistribution

Data have been considered from several 1/7 scale hydraulic reactor model tests [36, 37, 38] in arriving at the core inlet flow maldistribution criteria to be used in the subchannel analyses. THINC-1 analyses using these data have indicated that a conservative design basis is to consider a five percent reduction in the flow to the hot assembly.[39] The design basis of 5% flow reduction to the hot assembly is also used in the VIPRE analysis for the stretch power uprate.

Flow Redistribution

The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the flow redistribution is inherently considered in the VIPRE analysis

Flow Mixing

Mixing vanes have been incorporated into the spacer grid design. These vanes induce flow mixing between the various flow channels in a fuel assembly and also between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances. The subchannel mixing model now incorporated in the THINC or VIPRE code and used in reload reactor design is based on experimental data.[40]

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3.2.2.7 Core Pressure Drop and Hydraulic Loads

Core and vessel pressure losses are calculated by equations of the form:

$$\Delta P_L = (K + F\frac{L}{D_e})\frac{\rho V^2}{2g_c(144)}$$

where:

ΔP_L = unrecoverable pressure drop, lbf/in^2

ρ = fluid density, lbm/ft lbf/in^3

L = length, feet

D_E = equivalent diameter, feet

V = fluid velocity, ft/sec

g_c = 32.174 lbm-ft/lbf-sec^2

K = form loss coefficient, dimensionless

F = friction loss coefficient, dimensionless

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

The results of full scale tests of core components and fuel assemblies are utilized in developing the core pressure loss characteristic in reload reactor design. The pressure drop for the vessel has been obtained by combining the core loss with correlation of 1/7 scale model hydraulic test data on a number of vessels [(36)(37)] and form loss relationships. [(41)] Moody [(42)] curves have been used to obtain the single phase friction factors.

The fuel assembly holddown springs were designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events, with the exception of the turbine overspeed transient associated with a loss of external load. The holddown springs were designed to tolerate the possibility of an overdeflection associated with fuel assembly liftoff for this case and provide contract between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a Loss-of-Coolant Accident.

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Hydraulic loads at normal operating conditions are calculated considering the best estimate flow and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the best estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create flow rates 20 percent greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight. The hydraulic forces are not sufficient to lift a rod control cluster during normal operation even if the rod cluster is detached from its coupling.

3.2.3 Mechanical Design and Evaluation

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2-22 and in elevation in Figure 3.2-23. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and guide thimble plugging devices, provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, were designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. A listing of the core mechanical design parameters is given in Table 3.2-5.

The fuel assemblies are arranged in a roughly checkered circular and/or zoned pattern. The assemblies are all identical in configuration, but contain fuel of different enrichments depending on the location of the assembly within the core.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of 144 inches within Zircaloy-4 or ZIRLO™ tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are internally pressurized with helium during fabrication. The enrichments of the fuel for the first three regions and current cycle in the core are given in Table 3.2-5. Heat generated by the fuel is removed by demineralized light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The core is divided into fuel assembly regions of different enrichments. The loading arrangement for the initial cycle is indicated on Figure 3.2-24. Refueling originally took place generally in accordance with an inward loading schedule, but now a modified loading schedule for low - low leakage core design is used.

The control rods, designated as Rod Cluster Control Assemblies (RCCA), consist of groups of individual absorber rods which are held together by a spider at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes which form an integral part of the upper core support structure. Figure 3.2-25 shows a typical rod cluster control assembly.

As shown in Figure 3.2-23 the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the fuel assembly alignment which permits free movement of the control rods from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surfaces. During the refueling outrage for Cycle 8/9, the upper core plate was modified such that locations A5 and A11 were each missing one pin and location B13 was missing both pins. A special analysis was performed by Westinghouse showing that

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this configuration was acceptable for normal and design basis event operation. This analysis conservatively assumed absence of both alignment pins in these three locations, and also for location A6, which has two intact pins, one of which has been straightened. [71]

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a form fitting baffle surrounding the fuel assemblies which confines the upward flow of coolant in the core area to the fuel bearing region.

Reactor Internals

Design Description

The reactor internals were designed to support and orient the reactor core fuel assemblies and control rod assemblies, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support incore instrumentation. The reactor internals are shown in Figure 3.2-23.

The internals were designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, and earthquake acceleration. The internals were analyzed in a manner similar to that employed for Connecticut Yankee, San Onofre, Zorita, Saxton and Yankee. Under the loading conditions specified, which included conservative effects of design earthquake loading, the structure satisfied stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals were fabricated primarily from type 304 stainless steel.

The reactor internals are equipped with bottom-mounted incore instrumentation supports. These supports were designed to sustain the applicable loads outlined above.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure and the in-core instrumentation support structure.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.2-23. This support structure assembly consists of the core barrel, the core baffle, and lower core plate and support columns, the thermal shield, the intermediate diffuser plate and the bottom support plate which is welded to the core barrel. All the major material for this structure is type 304 stainless steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provided support and orientation for the fuel assemblies.

The lower core plate is a 2-inch thick member through which the necessary flow distributor holes for each fuel assembly were machined. Fuel assembly locating pins (two for each

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assembly) are also inserted into this plate. Columns were placed between this plate and the bottom support plate of the core barrel in order to provide stiffness and to transmit the core load to the bottom support plate. Intermediate between the support plate and lower core support plate was positioned a perforated plate to diffuse uniformly the coolant flowing into the core.

The one-piece thermal shield is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield/core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing in which material samples can be inserted and irradiated during reactor operation are welded to the thermal shield and extend to the top of the thermal shield. These samples are held in the rectangular tubing by a preloaded spring device at the top and bottom.

The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annualus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support plate, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core and coolant enters the area of the upper support structure and then flows generally radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum to provide cooling of the head. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell, and partially through the lower support columns to the lower core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be distributed to the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by a radial support type connection of the upper core plate to slab sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by “key” and “keyway” joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel ID. Another Inconel block is bolted to each of these blocks, and has a “keyway” geometry. Opposite each of these is a “key” which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits. This eliminates any possibility of failure of the core support.

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In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals.

The energy absorbers, which are cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design were determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

The free fall in the hot condition is on the order of ½ inch, and there is an additional strain displacement in the energy absorbing devices of approximately ¾ inch. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control rods are designed to provide assurance of control rod insertion capabilities under this assumed drop of internals condition. The drop distance of about 1 ¼ inch is not enough to cause the tips of the shutdown group of RCC assemblies to come out of the guide tubes in the fuel assemblies.

Upper Core Support Assembly

The upper core support assembly, shown in Figure 3.2-28, consists of the top support plate, deep beam sections, and upper core plate between which are contained 48 support columns and 61 guide tube assemblies. The support columns establish the spacing between the top support plate, deep beam sections, and the upper core plate and are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies, shown on Figure 3.2-29, sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the top support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 0°, 90°, 180°, and 270°. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly location pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and top support plate and then the reactor vessel head. Transverse loads form coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the

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top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported form the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper in-core instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation. Section 7.4 contains more information on the layout of the incore instrumentation system.

The incore instrumentation support structure was designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

Evaluation of Core Barrel and Thermal Shield

The internals design was based on analysis, test and operational information. Troubles in previous Westinghouse PWR's were evaluated and information derived was considered in this design. For example, the Westinghouse design uses a one-piece thermal shield which is attached rigidly to the core barrel at one end and flexured at the other. The early designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

Early core barrel designs that have malfunctioned in service, now abandoned, employed threaded connections such as tie rods, joining the bottom support to the bottom of the core barrel, and a bolted connection that tied the core barrel to the upper barrel. The malfunctioning of core barrel designs in earlier service was believed to have been caused by the thermal shield which was oscillating, thus creating forces on the core barrel. Other forces were induced by unbalanced flow in the lower plenum of the reactor. In the Indian Point 3 RCC design there are no fuel followers to necessitate a large bottom plenum in the reactor. The elimination of these fuel followers enabled Westinghouse to build a shorter core barrel.

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The Connecticut Yankee reactor and the Zorita reactor core barrels are of the same construction as the Indian Point reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot-functional test, and deflection and strain gages employed in the Zorita reactor during the hot-functional test provided important information that was used in the design of the present day internals, including that for Indian Point. When the Connecticut Yankee thermal shield was modified to the same design as for Southern California Edison, it, too, operated satisfactorily as was evidenced by the examination after the hot-functional test. After hot-functional tests on all of these reactors, a careful inspection of the internals was provided. All the main structural welds were examined, nozzle interfaces were examined for any differential movement, upper core plate inside supports were examined, the thermal shield attachments to the core barrel including all lockwelds on the devices used to lock the bolt were checked, no malfunctions were found.

Substantial scale model testing was performed at WAPD. This included tests which involved a complete full scale fuel assembly which was operated at reactor flow, temperature and pressure conditions. Tests were run on a 1/7 scale model of the Indian Point 3 reactor. Measurements taken from those tests indicated very little shield movement, on the order of a few mils when scaled up to Indian Point. Strain gauge measurements taken on the core barrel also indicated very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances was included. Information gathered from these tests was used in the design of the thermal shield and core barrel. It was concluded from the testing program and the analyses with the experience gained that the design as employed on the Indian Point Plant is adequate. Further confirmation of the internals design was made on Indian Point 2. Deflection gauges were mounted on the thermal shield top and bottom for the hot-functional test. Six such gauges were mounted in the top of the thermal shield equidistant between the fixed supports and eight located at the bottom, equidistant between the six flexures, and two next to flexure supports. The internals inspection, just before the hot-functional test, included looking at mating bearing surfaces, main welds, and welds that are used on bolt locking devices. At the conclusion of the hot-functional test, measurement readings were taken from the deflectometers on the shield and the internals were re-examined at all key areas for any evidence of malfunction.

The final report of the Indian Pont 2 vibrational test, Reference 52, was transmitted to the Deputy Director for Reactor Projects in August, 1972. This report supports the use of Indian Point 2 internals as the prototype for Indian Point 3 internals.

Core Components

Design Description

Fuel Assembly

The original fuel design for Indian Point 3 was the Westinghouse Low Parasitic (LOPAR) fuel assembly. For the Cycle 5 reload, a new design fuel, the Westinghouse Optimized Fuel Assembly (OFA) was introduced (and LOPAR was phased out by Cycle 7). The major design difference between the two designs in the use of the five middle Zircaloy grids for the new design versus five middle Inconel grids for the old design. (Reference 54)

For the Cycle 7 reload, another new design, VANTAGE-5, was introduced (and by Cycle 11, the only OFA assembly is the central assembly).

For the cycle 9 reload, ZIRLO™ clad was introduced and continues to be used.

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For the Cycle 10 reload, a third design, VANTAGE+ was introduced.

For the Cycle 11 reload, the VANTAGE+ design was enhanced to include "PERFORMANCE+" features. This enhanced design has all the features of the VANTAGE+ design and also includes the Protective Bottom Grid (Reference 76).

For the Cycle 14 core, the design known as 15x15 Upgrade was introduced. This design includes an enhanced grid and IFMs to reduce grid-rod fretting.

The overall configuration of the fuel assemblies is shown in Figures 3.2-30 and 3.2-31. The assemblies are square in cross-section, nominally 8.426 inches on a side and have an overall height of 13 feet 4 inches.

VANTAGE 5 Fuel

The Vantage 5 Fuel assembly has been designed to be compatible with the OFAs, reactor internals interfaces, the fuel handling equipment, and refueling equipment. The VANTAGE 5 design dimensions are essentially equivalent to the IP3 OFA assembly design from an exterior assembly envelope and reactor internals interface standpoint. (Reference 73)

The significant new mechanical features of the VANTGAGE 5 design relative to the OFA design in operation include the following:

- Integral Fuel Burnable Absorber (IFBA)
- Reconstitutable Top Nozzle
- Slightly longer fuel rod and assembly for extended burnup capability
- Axial Blankets
- Redesigned fuel rod bottom end plug to faciliate reconstitution capability

Other different mechanical features are the use of a standardized chamfer pellet design and the Debris Filter Bottom Nozzle (DFBN).

VANTAGE + Fuel

Vantage + uses the following V5 features

- Reconstitutable Top Nozzle (RTN)
- Extended Burnup Fuel Assembly Design
- Extreme Low Leakage Loading Pattern
- Enriched Integral Fuel Burnable Absorbers (IFBAs)
- Debris Filter Bottom Nozzle DFBN
- Axial Blankets

In addition V+ incorporates the following features as described in Reference 74

- ZIRLO™ Fuel Cladding
- Low Pressure Drop (LPD) Mid-grids
- Integral Flow Mixer grids (IFMs)
- ZIRLO™ guide thimbles and instrumentation tubes
- Variable Pitch Fuel Rod Plenum Spring

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- Mid-Enriched Annular Fuel Pellets in Axial Blanket
- Fuel Assembly and Fuel Rod Dimensional Modifications

The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563 inch between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the RCCA rods and one for in-core instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, seven grid assemblies, twenty guide thimbles, and one instrumentation thimble. Occasionally, stainless steel, zirconium alloy filler rods, or slightly enriched uranium fuel rods will replace failed fuel rods in reconstituted fuel assemblies. These rods are identical in shape and size to fuel rods. Special analyses are performed prior to any new fuel cycle utilizing reconstituted fuel.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are secured to the top and bottom nozzles respectively. The grid assemblies, in turn, are fastened to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

Bottom Nozzle

The bottom nozzle is a square box-like structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, was fabricated from 304 stainless steel parts consisting of a perforated plate, four angle legs, and four pads or feet. The angle legs are welded to the plate forming a plenum space for coolant inlet to the fuel assembly. The perforated plate serves as the bottom limit for radiation and thermally induced growth for the fuel rods. The bottom support surface for the fuel assembly is formed under the plenum space by the four pads which are welded to the corner angles. The bottom nozzle now has more but smaller flow holes to mitigate possible fuel damage from debris.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly and to the channel between assemblies through the holes in the plate. The flow holes in the plate were sized and positioned beneath the fuel rods so that the rods cannot pass through the plate.

The RCC guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle. These loads as well as the weight of the assembly are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins. For Vantage 5 fuel the bottom nozzle has been made thinner to allow for increased fuel rod growth and higher burnup. This feature was maintained for V+ fuel.

Top Nozzle

The top nozzle is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is mixed and directed toward the flow holes in the upper core plate. The nozzle is comprised of an

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adapter plate enclosure, top plate, two clamps, four 2-leaf holdown springs and assorted hardware. All parts with the exception of the springs and their hold down bolts were constructed of type 304 stainless steel. The springs were made from age hardenable Inconel 718 and the bolts from Inconel 600. The assemblies of the OFA design have about 4.5% increase in hydraulic resistance to flow compared to the LOPAR design. This results in an increased lift force to the OFA and requires 3-leaf holdown springs in the top nozzle instead of 2-leaf springs used for LOPAR assemblies.

The adapter plate portion of the nozzle is square in cross-section, and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the control guide thimbles were fastened through individual bored holes in the plate. Thus, the adapter plate acts as the fuel assembly top end plate, and provides a means of distributing evenly among the guide thimbles any axial loads imposed on the fuel assemblies.

The nozzle enclosure is actually a square tubular structure which forms the plenum section of the top nozzle. The bottom end of the enclosure is welded to the periphery of the adapter plate, and top end is welded to the periphery of the top plate.

The top plate is square in cross-section with a square central hole. The hole allows clearance for the RCC absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to the upper internals area. Two pads containing axial through-holes, which are located on diametrically opposite corners of the top plate, provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate.

Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf springs which are mounted on the top plate. The springs are fastened to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Fastening of each pair of springs was accomplished with a clamp which fits over the ends of the springs and two bolts (one per spring) which pass through the clamp and springs, and thread into the top plate. At assembly, the spring mounting bolts were torqued sufficiently to reload against the maximum spring load and then lockwelded to the clamp which is counter-bored to receive the bold head.

The spring load is obtained through deflection of the spring pack by the upper core plate. The spring pack form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring pack is bent downward and captured in a key slot in the top plate to guard against loose parts in the reactor in the event (however remote) of spring fracture. In addition, the fit between the spring and key slot and between the spring and its mating slot in the clamp are sized to prevent rotation.

In addition to its plenum and structural functions, the nozzle provides a protective housing for components which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the fuel assembly adapter plate and the end of the guide tube in the upper internals package. Plugging devices, which fill the ends of the fuel assembly thimble tubes at unrodded core locations, and the spiders which support the source rods and burnable poison rods, are all contained within the fuel top nozzle.

The current top nozzle design is such as to allow easy reconstitution.

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Guide Thimbles

The control rod guide thimbles in the fuel assembly provide guided channels for the control rods during insertion and withdrawal. They were fabricated from a single piece of Zircaloy-4 tubing, which is drawn to two different diameters. Starting with Cycle 10, the guide thimble material is ZIRLO™. The larger inside diameter at the top provides a relatively large annular area for rapid insertion during a reactor trip and to accommodate a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is a reduced diameter to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

The OFA design guide thimbles are the same as the guide thimbles for LOPAR design except for a 13 mil inner diameter and water diameter reduction. The OFA guide thimble diameter provides adequate diametral clearance for control rods, source rods, burnable absorber rods, and thimble plugs.

Flow holes are provided just above the transition of the two diameters to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is closed at the bottom by means of a welded end plug. The end plug was fastened to the bottom nozzle during fuel assembly fabrication.

The top ends of the guide thimbles are expanded into stainless steel sleeves which are welded to the top grid. The sleeves are fitted through individual holes in the adaptor plate and welded around the circumference of the holes.

The 15x15 Upgrade incorporates the new tube-in-tube guide thimble design which utilizes a separate dashpot tube assembly that is inserted into the guide thimble assembly, pulled to a press fit over the thimble end plug and bulged into place. As the dashpot in this design can provide additional lateral support in that bottom thimble span, it is expected that there will be additional resistance to lateral deformation and incomplete rod insertions as a result of this design modification. The thimble screw for the tube-in-tube design is slightly longer than in the previous guide thimble tube design so that it can properly engage with the threads on the new guide thimble end plug and extend through the end plug of the dashpot tube assembly.

Grids

The spring clip grid assemblies consist of individual slotted straps which are assembled and interlocked in an "egg-crate" type arrangement and then furnace brazed to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes, and tabs were punched and formed in the individual straps prior to assembly.

Different types of grid assemblies are used in the fuel assembly. One type having mixing vanes which project from the edges of the straps into the coolant stream, is used in the high heat region of the fuel assemblies for mixing of the coolant. A grid of this type is shown in Figure 3.2-32. Grids of another type, located at the bottom and top ends of the assembly, are of the non-mixing type. They are similar to the mixing type with the exception that mixing vanes are not used on the internal straps. They are made of Inconel.

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The spacing between grids is shown on Figure 3.2-31. The variation in span lengths is the result of optimization of the thermal-hydraulic and structural parameters. The grids are fastened securely to each guide thimble.

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose.

Inconel 718 was chosen for the grid material for the top and bottom grid because of its good corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material was age hardened to obtain the material strength necessary to develop the required grid spring forces.

The OFA design features five middle Zircaloy-4 grids which have thicker and wider straps than the LOPAR Inconel grids to compensate for the difference in natural strength properties. Zircaloy grids maintain their integrity during the most severe load conditions of a combined seismic/LOCA event.

The Vantage+ fuel assemblies have five Low Pressure Drop (LPD) mid grids and three intermediate flow mixing (IFM) grids for increased DNB margin. These are all made of ZIRLO™.

The I-Spring design, introduced in Cycle 14 with the 15x15 Upgrade fuel, uses a revised spring, dimple and strap design to reduce the probability of grid-rod fretting.

Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in a slightly cold worked and partially annealed Zircaloy-4 or ZIRLO™ tubing which plugged and seal welded at the ends to encapsulated the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without over-stressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a stainless steel helical compression spring which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process. [47] A hold-down force of approximately six times the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack.

The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide power which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length.

A different fuel enrichment, as listed in Table 3.2-5, was used for each of the three regions in the first core loading. (Checker Board Pattern, see Figure 3.2-24.) Subsequent regions were uniquely designed for their enrichments. Current cycle is provided in Table 3.2-5 and 3.2-24A.

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Each fuel assembly is identified by means of a serial number engraved on the upper nozzle. The fuel pellets are fabricated by a batch process so that only one enrichment region is processed at any given time. The serial numbers of the assemblies and corresponding enrichment are documented by the manufacturer and verified prior to shipment.

Each assembly is assigned a specific core loading position prior to insertion. A record is then made of the core loading position, serial number, and enrichment.

During initial core loading and subsequent refueling operations, detailed written handling and checkoff procedures were utilized throughout the sequence. The initial core was loaded in accordance with the core loading diagram similar to Figure 3.2-24 which shows the location for each of the three enrichment types of fuel assemblies used in the loading together with the serial number of the assemblies in the region.
Extensive administrative controls (as discussed in Sections 3.2.3 and 3.3.3) render the possibility of loading fuel assemblies with incorrect enrichments or without their burnable poison rods extremely unlikely. Independent checks are made, prior to fuel loading, of each fuel assembly matching the contents of the assembly with its position in the core. Further checks are provided during core loading utilizing detailed written handling and checkoff procedures.

Achieving criticality curing core loading is prohibited in any case as the subcritical neutron flux is continuously monitored and the inverse count rate ratio is plotted to detect any unexpected rise in the subcritical neutron flux. Core loading is stopped should the subcritical count rate rise by more than a preset factor.

Any such loading error, not significant enough to be detected during initial core loading is of no consequence from a criticality standpoint and would be detected by the power distribution map. During subsequent refueling operations the flux profile is flatter and loading errors can be detected by the power distribution map. (See the Technical Specifications)

Rod Cluster Control Assemblies

The control rods or rod cluster control assemblies (RCCA) each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, one of which is shown in Figure 3.2-25 are provided to control the reactivity of the core under operating conditions.

The absorber material used in the control rods is silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCC assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes supporting cylindrical fingers from which the absorber rods are suspended. Handling detents, and detents for connection to the drive shaft are machined into the upper end of the hub. A spring pack is

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assembled into a skirt integral to the bottom of the hub to stop the RCC assembly and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub, and the fingers are joined to the vanes by furnace brazing. A centerpost which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from type 304 stainless steel except for the springs which are Inconel X-750 alloy and the retainer which is of 17-4 PH material.
The absorber rods are secured to the spider so as to assure trouble free service. The rods were first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins were welded in place. The end plug below the pin position was designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver-indium-cadmium rods were inserted into cold-worked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral and end clearance were provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs were made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions.

Neutron Source Assemblies

Neutron source assemblies are utilized in the core. In Cycle 1, these consisted of two assemblies with four secondary source rods each and two assemblies with one primary source rod each. In subsequent cycles, secondary sources only are utilized. The secondary source rods are fastened to a spider at the top end of the assembly. The initial core primary source rods were attached to a burnable poison assembly. Beginning with Cycle 14, secondary source assemblies using baseplate mounts began to be introduced into the core.

In the core, the neutron source assemblies are inserted into the RCC guide thimbles in fuel assemblies at unrodded locations. The location and orientation of the assemblies in the initial core is shown in Figure 3.2-33.

The primary and secondary source rods of the initial core utilized the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing) in which the sources are inserted. The source rods contain Sb-Be pellets. The primary source rods each contained capsules of Pu 238-Be source material at a neutron strength of approximately 2×10^8 neutrons/sec. Design criteria for the source rods are: cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and clad. For reload cores, the primary source rods have been removed and only the secondary source rods remain in core locations determined by the reload design.

Plugging Devices

In order to limit bypass flow through the RCC guide thimbles in fuel assemblies which do not contain either control rods, source assemblies, or burnable poison rods, the fuel assemblies at

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those locations were fitted with plugging devices. The plugging devices consist of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly and mixing device attached to the top surface. At installation in the core, the plugging devices fitted with the fuel assembly top nozzles and rested on the adapter plate. The short rods project the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack was compressed by the upper core plate when the upper internals package was lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles and on burnable poison rod assemblies in these guide thimble positions where there are no burnable poison rodlets.

All components in the plugging device, except for the springs, were constructed from type 304 stainless steel. The springs (one per plugging device) were wound from an age hardenable nickel base alloy to obtain higher strength.

Burnable Poison Rods

The burnable poison rods are statically suspended and positioned in vacant assembly guide thimble tubes within the fuel assemblies at nonrodded RCC core locations. The poison rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat base plate which fits with the fuel assembly top nozzle and rests on the top adapter plate.

The base plate and the poison rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the poison rods cannot be lifted out of the core by flow forces.

The old poison rods consist of borosilicate glass tubes contained within type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall type 304 stainless steel tubular inner liner. A typical old burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-34. These rods can still be used for fluence reduction on the reactor vessel, as well as for burnable poison. The new poison rod design is referred to as Wet Annular Burnable Absorber (WABA). The WABA design has an annular aluminum oxide-boron carbide (Al_2OB_4C) absorber pellets contained within two concentric zircalogy tubings with water flowing through the center tubes as well as around the outer tubes. Eight demonstration poison rods of a new design, consisting of annular pellets of Al_2OB_4C contained within two concentric zircaloy tubings were also employed in the Cycle 3 and Cycle 4 cores. Cycles 5 and beyond have used large complements of WABA. Cycle 7 also utilized the Integral Fuel Burnable Absorber (IFBA). This consists of a thin coating of zirconium diboride on the cyclindrical surface of the pellet. Cycles 9 and beyond have used IFBA and WBA.

The rods were designed in accordance with the standard fuel rod design criteria; i.e.., the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass or annular pellets as a result of the $B_{10}(n,\gamma)$ reaction. A more detailed discussion of the old burnable poison rod design is found in WCAP-9000. [45] A more detailed discussion of the new burnable poison rods if found in WCAP-10021 (Revision 1) [55]. The new burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-34.

Based on available data on properties of borosilicate glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during

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operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner was sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should possibly occur.

The top end of the inner liner is open to receive the helium which diffuses out of the glass.

To ensure the integrity of the burnable poison rods, the tubular cladding and end plugs were procured to the same specifications and standard of quality as was used for stainless steel fuel rod cladding and end plugs in other Westinghouse plants. In addition, the end plug seal welds were checked for integrity by visual inspection and X-ray. The finished rods were helium leak checked.

Starting in Cycle 11, Hafnium Flux Suppressors have been placed in the eight core locations closest to the reactor vessel to suppress neutron fluence on the reactor vessel, thereby prolonging vessel life. These hafnium flux suppressors replace eight borosilicate-glass (Pyrex) burnable absorbers that were used up to Cycle 10. The hafnium flux suppressors are constructed with the same upper fixture as the Pyrex and WABA burnable absorbers currently available for use, and are handled in the same manner during refueling. The flux suppressors use unclad, hafnium rods and may be used in future operating cycles. (Reference 76)

Protective Bottom Grid

Starting with Cycle 11, the fuel assemblies (known as PERFORMANCE+) have a Protective Bottom Grid, which is an extra grid strap located at the bottom of the fuel pins, between the Bottom Nozzle and the bottom grid. The purpose of this protective bottom grid is to capture debris entering the fuel assembly and trap it at an elevation below the top of the fuel pin end plug. Therefore, any debris caught by the protective bottom grid cannot fret through or otherwise damage the fuel pin cladding in such a manner that the fuel pellets or rod plenum is exposed, thus making the fuel assembly more resistant to debris-related fuel failures.

The PERFORMANCE+ design also has the fuel pins mounted lower in the grid cage than fuel assembly designs before Cycle 11, almost touching the bottom nozzle. During operation, it is expected that pin growth and grid relaxation will allow the pin to rest on the bottom nozzle. The resultant transfer of weight from the grid cage to the bottom nozzle is expected, and may result in less bowing of the irradiated fuel assembly. (Reference 76)

Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods were calculated by an overall fuel rod design model [48] which incorporates the time-dependent fuel densification. [49] The increase of internal pressure in the fuel rod due to this phenomena was included in the determination of the maximum cladding stresses at the end of core life when fission product gap inventory is a maximum. IFBA fuel has an additional contribution to rod internal pressure in the form of helium which is assumed to release at a rate of 100 percent.

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The maximum allowable strain in the cladding, considering the combined effects of internal fission gap pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than one percent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods have met a high standard of excellence from the standpoint of functional requirements, many inspections and tests were performed both on the raw material and the finished product. These tests and inspections included chemical analysis, elevated temperature, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing and helium leak tests.

In the event of cladding defect, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature or fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of high burnup uranium dioxide [43] fuel element behavior indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considered the effect of burnup, temperature distribution, and internal voids. It was incorporated in the overall fuel rod design model. [48]

Actual damage limits depend upon neutron exposure and normal variation of material properties and are greater than the design limits. For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and stain are as follows:

1) Internal gas pressure

   The maximum rod internal pressure under nominal conditions is substantially less than the calculated pressure at the design limits. The end-of-life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history and IFBA loading. However, it does not exceed the design limit defined in Section 3.1.2.

2) Cladding temperature

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions (715°F) is given in Table 3.2-4, along with many other thermal and hydraulic design parameters.

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3) Burnup

Fuel burnup results in fuel swelling which produces cladding strain. The strain damage limit is not expected to be reached until the peak burnup reaches a value in excess of 58,000 MWD/MTU.

4) Fuel temperature and kW/ft

The fuel is designed so that the maximum fuel temperature will not exceed 4700°F during normal operation or malfunction transients.

Evaluation of Burnable Poison Rods

The burnable poison rods are positively positioned in the core inside RCC assembly guide thimbles and held down in place by attachment to a baseplate assembly compressed beneath the upper core plate and hence cannot be ejected and cause a reactivity transient. Due to the low heat generation rate, and the conservative design of the poison rods, there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients, including loss of coolant.

Many nuclear plants are using or have used burnable poison rods of the old design. Reference 51 describes the test operational experience of the old poison rod design. Eight demonstration poison rods of the new design were used in Cycles 3 and 4. Post irradiation examination following Cycles 3 and 4 indicated rodlets performed as expected and these were no anomalies. Further details of examinations performed can be referenced in WCAP-10021 (Revision 1). [55] The design was retained for Cycles 5 and beyond.

Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCC control rods, and burnable poison rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for the nominal design lifetime of 15 Effective Full-Power Years

In the case of the fuel rod grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (approximately 0.001), and the stress associated with the motion is significantly small (less than 100 psi). Like wise, the reactions at the grid spring due to the motion is much less than the preload spring force and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and the grid support is not anticipated.

The effect of thermal cycling on the grid-clad support is merely a slight relative movement between the grid contact surfaces and the clad, which is gradual in nature during heat-up and cool-down. Since the number of cycles of the occurrence is small over the life of a fuel assembly, negligible wear of the mating parts is expected.

These conclusions have been verified by fuel operating experience in a number of nuclear plants as described in Reference 51.

The dynamic deflection of the full length control rods and the burnable poison rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot. With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed

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cyclic deflection through the available clearance gap results in an insignificantly low stress in either the clad tubing or in the flexture joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider, or the retainer plate in the case of the burnable poison rods.

A calculation, assuming the rods are supported by the surface of the dashpots and at the upper end by the spider or retainer, results in a similar conclusion.

Control Rod Drive Mechanism

Full Length Rods

Design Description

The control rod drive mechanisms are used for withdrawal and insertion of the rod cluster control assemblies into the reactor core and to provide sufficient holding power for stationary support.

Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity. Typical total insertion time is less than 1.8 seconds and always less than 2.7 seconds.

The complete drive mechanism, shown in Figure 3.2-36, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indictor coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each mechanism pressure hosing is threaded onto an adapter on top of the reactor pressure vessel and seal welded. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Main coolant fills the pressure containing parts of the control rod drive mechanism. All working components and the shaft are immersed in the main coolant and depend on it for component damping.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. They move into sets of latches which lift, lower, and hold the grooved drive shaft.

The three magnets are turned on and off in a fixed sequence by solid-state switches for the full length rod assemblies.

The sequencing of the magnets produces step motion over the 144 inches of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between

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the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The mechanisms were designed to operate in water 650°F at and 2485 psig. The temperature at the mechanism head adapter will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

A multi-conductor cable connects the mechanism operating coils to the 125 volt DC power supply. The power supply is described in Section 7.3.2.

Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets.

They actuate two sets of latches which engage the grooved section of the drive shaft.

The upper set of latches move up or down to raise or lower the drive rod by 5/8 inch. The lower set of latches have a maximum 1/16 inch axial movement to shift the weight of the control rod from the upper to the lower latches.

Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed and installed while the reactor is pressurized.

The three operator coils are made of round copper wire which is insulated with a double layer of filament type glass yarn.

The design operating temperature of the coils is 200 C. Average coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains a coil casing temperature of approximately 120 C or lower.

Drive Shaft Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144 inches of control rod travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45° angle sides.

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The drive shaft is attached to the control rod by the coupling. The coupling has two flexible arms which engage the grooves in the spider assembly.

A ¼ inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its lower end, there is a disconnect assembly. For remote disconnection of the drive shaft assembly from the control rod, a button at the top of the drive rod actuates the connect/disconnect assembly.

During plant operation, the drive shaft assembly remains connected to the control rod at all times. It can be attached and removed from the control rod only when the reactor vessel head is removed.

Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of a cylindrically wound differential transformer which spans the normal length of the rod travel (144 inches).

Drive Mechanism Materials

All parts exposed to reactor coolant, such as the pressure vessel, latch assembly and drive rod, are made of metals which resist the corrosive action of the water.

Three types of metals are used exclusively: stainless steel, Inconel X, and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used. Cobalt based alloys are used for the pins, latch tips, and bearing surfaces.

Inconel X is used for the springs of both latch assemblies and 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts (such as threads) during assembly.

Outside of the pressure vessel, where the metals are exposed only to the Reactor Containment environment and cannot contaminate the main coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc-plated to approximately 0.001 inch thickness to prevent corrosion.

Principles of Operation

The drive mechanisms, shown schematically in Figures 3.2-35 and 3.2-36 withdraw and insert their respective control rods as electrical pulses are received by the operator coils.

ON and OFF sequence, repeated by switches in the power programmer causes either withdrawal or insertion of the control rod. Position of the control rod is indicated by the differential transformer action of the position indicator coil stack surrounding the rod

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travel housing. The differential transformer output changes as the top of the ferromagnetic drive shaft assembly moves up the rod travel housing.

Generally, during plant operation the drive mechanisms hold the control rods withdrawn from the core in a static position, and only one coil, the stationary gripper coil, is energized on each mechanism.

Control Rod Withdrawal: The control rod is withdrawn by repeating the following sequence:

1) Movable Gripper Coil – ON

   The moveable gripper armature raises and swings the movable gripper latches into the drive shaft groove.

2) Stationary Gripper Coil – OFF

   Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches then swing out of the shaft groove.

3) Lift Coil – ON

   The 5/8 inch gap between the lift armature and the lift magnet pole closes and the drive rod raises one step length.

4) Stationary Gripper Coil – ON

   The stationary gripper armature raises and closes the gap below the stationary gripper magnetic pole, swings the stationary gripper latches into a drive shaft groove. The latches contact the shaft and lift it 1/16 inch. The load is so transferred from the movable to the stationary gripper latches.

5) Movable Gripper Coil –OFF

   The movable gripper armature separates from the lift armature under the force of the spring and gravity. Three links, pinned to the movable gripper armature, swing the three movable gripper latches out of the groove.

6) Lift coil – OFF

   The gap between the lift armature and the lift magnet pole opens. The moveable gripper latches drop 5/8 inch to a position adjacent to the next groove.

Control Rod Insertion:

The sequence for control rod insertion is similar to that for control rod withdrawal:

1) Lift Coil – ON

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The movable gripper latches are raised to a position adjacent to a shaft groove.

2) Movable Gripper Coil – ON

The movable gripper armature raises and swings the movable gripper latches into a groove.

3) Stationary Gripper Coil – OFF

The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

4) Lift Coil – OFF

Gravity and spring force separate the lift armature from the lift magnetic pole and the control rod drops down 5/8 inch.

5) Stationary Gripper Coil – ON

6) Movable Gripper Coil – OFF

The sequences described above are termed as one step or one cycle and the control rod moves 5/8 inch for each cycle. Each sequence can be repeated at a rate of up to 72 steps per minute and the control rods can therefore be withdrawn or inserted at a rate of up to 45 inches per minute.

Control Rod Tripping:

The holding or static mode is with the stationary gripper coil. If power to the stationary gripper coil is cut off, as for tripping, the combined weight of the drive shaft and the rod cluster control assembly is sufficient to move the latches out of the shaft groove. The control rod falls by gravity into the core. The tripping occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight acting upon the latches.

Fuel Assembly and RCCA mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and full length RCC assembly, functional test programs were conducted on a full scale Indian Point 2 prototype 12 ft can less fuel assembly and control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated 2,260,892 steps and 600 scrams. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the fuel assembly and drive line components did not reveal significant fretting. The wear of the absorber rods, fuel assembly guide thimbles, and upper guide tubes was minimal. The control rod free fall time against 125% of nominal flow was less than 1.5 seconds to the dashpot (10 ft of travel). Additional tests had previously been made on a full scale San Onofre mock up version of the fuel assembly and control rods. [44]

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Mockup Tests at 1/7 Scale for Indian Point 2

A 1/7 scale model of the Indian Point 2 internals was designed and built for hydraulic and mechanical testing. The tests provided information on stresses and displacements at selected locations on the structure due to static loads, flow induced loads, and electromagnetic shaker loads. Flow distribution and pressure drop information were obtained. Results of the static tests indicated that mean strains in the upper core support plate and upper support columns are below design limits. Strains and displacements measured in the model during flow tests verified that no damaging vibration levels were present. Additional information gained from the tests were the natural frequency and damping of the thermal shield and other components in air and water. Model response can be related to the full scale plant for most of the expected exciting phenomena, but across the board scaling is not possible. Specifically exciting phenomena which are strongly dependent on Reynolds number cannot be scaled. In areas where Reynolds number may be important, either: (1) the measured vibration amplitudes were many times lower than a level that would be damaging, or (2) full scale vibration data were obtained.

Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location were also successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

Axial and Lateral Bending Tests

In addition, axial and lateral bending tests were performed in order to simulate mechanical loading of the assembly during refueling operation. Although the maximum column load expected to be experienced in service is approximately 1000 lb, the fuel assembly was successfully loaded to 2200 lb axially with no damage resulting. This information was also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

3.2.4 ZIRLO Clad Fuel

ENTERGY is using fuel assemblies containing fuel rods fabricated with the advanced zirconium alloy cladding material ZIRLO™ into the Indian Point Unit 3 Cycle 9 core and beyond. These fuel assemblies will have fuel rods fabricated with ZIRLO™ cladding to obtain additional operational benefit from the cladding's improved corrosion resistance. The transition to VANTAGE 5 fuel was made at Indian Point 3 for Cycle 7 as described in the submittal to the NRC dated January 20, 1989 (IPN-89-007). Cycle 10 began the use of V+ fuel which utilizes ZIRLO™ for guide thimbles and intermediate spacers as well as clad.

This report shows, based on both evaluations and analyses, that no unreviewed safety questions exist as a result of inserting ZIRLO™ clad fuel rods into the Indian Point 3 reactor core. This report shows that the subsequent proposed changes to the Indian Point Unit 3 Technical Specifications will not involve significant hazard considerations.

3.2.4.1 Background

Westinghouse has developed a new zirconium based fuel rod clad alloy, known as ZIRLO™, to enhance fuel reliability and achieve extended burnup. This alloy provides significant improvement in fuel rod clad corrosion resistance and dimensional stability under irradiation. ZIRLO™ cladding corrosion resistance has been evaluated in long-term, out-of-pile tests over a

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wide range of temperatures (up to 600°F in water tests, up to 932°F in steam tests). Additional tests have also been conducted in lithiated water environments. The improved corrosion resistance of ZIRLO™ cladding has also been demonstrated to very high burnups in the BR-3 reactor.

A conditional licensing approval for the use of this advanced alloy cladding in two demonstration fuel assemblies for the North Anna Unit 1 reactor core was given in a USNRC letter dated May 13, 1987. The USNRC granted an exemption (Reference 66) from the provision of 10CFR50.46, 10 CFR50.44 and 10CFR51.52 with respect to the use of the North Anna demonstration fuel assemblies with the advanced cladding material, ZIRLO™. The Information required to support the licensing basis for the implementation of the ZIRLO™ clad fuel rods in Indian Point 3 is given in References 67 and 68. The fuel assemblies will be utilized in Indian Point 3, beginning with Cycle 9.

3.2.4.2 Areas Assessed

The following areas have been assessed during the safety evaluation process: chemical/mechanical properties, neutronic performance, thermal and hydraulic performance, cladding performance under non-LOCA conditions, and cladding performance under LOCA conditions.

Reference 69 addresses the VANTAGE 5 design and its application to a 17 x 17 fuel assembly. The VANTAGE 5 design may be applied to other fuel assembly arrays (14 x 14, 15 x 15) where such applications are evaluated on a plant specific basis and licensed in accordance with NRC requirements. Indian Point Unit 3 has been licensed for 15 x 15 VANTAGE 5 as already noted. Subsequently, the applicable models and methods employed to address the 15 x 15 VANTAGE 5 design have been licensed for Indian Point Unit 3. The principal difference between the Indian Point Unit 3 Region 11 fuel and the licensed 15 x 15 VANTAGE 5 fuel is the use of ZIRLO™ cladding. The use of ZIRLO™ cladding does not alter the previously licensed models and methods of Reference 69 with the exception of the LOCA model and methodology as noted in Chapter 14. The revised LOCA model and methodology were used as the basis to evaluate the effects of the change in cladding material. These evaluations have shown that the present LOCA related design bases and limits remain valid. Where the models and methods of Reference 69 are not affected by ZIRLO™ cladding, Indian Point 3 plant specific evaluations and analyses have also shown that the current design bases and limits remain valid.

3.2.4.3 Previous Irradiation Experience

Fuel rods fabricated with ZIRLO™ cladding have been previously irradiated in a foreign reactor (BR-3 reactor) at linear power levels up to 17 kw/ft, and burnups significantly greater than those planned for the Indian Point 3 fuel assemblies. Corrosion and hydriding data obtained on the ZIRLO™ were compared with the reference Zircaloy-4 cladding of fuel rods irradiated as controls in the same test assemblies. Based on the irradiation results of the test assemblies in the foreign reactor, the Indian Point Unit 3 ZIRLO™ cladding waterside corrosion and hydriding will be significantly less than that expected for the Zircaloy-4 clad fuel rods. The irradiation test results substantiate a lower clad irradiation growth $(\Delta L/L)$ and creepdown for the ZIRLO™ cladding compared to Zircaloy-4 cladding.

Two demonstration fuel assemblies, containing ZIRLO™ clad fuel rods, began irradiation in the North Anna Unit 1 reactor during June 1987. The ZIRLO™ clad fuel rods achieved over 21,000 MWD/MTU burnup in their first cycle (complete during February 1989). Visual inspection during refueling showed no abnormalities. One demonstration assembly with ZIRLO™ TM clad fuel

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rods underwent a second cycle of irradiation and achieved over 37,000 MWD/MTU burnup (completed January 1991). Visual inspection of the two cycle ZIRLO™ clad fuel rods during refueling showed no abnormalities. Cladding corrosion measurements showed that the reduced corrosion obtained with the ZIRLO™ clad rods was significantly better than that anticipated on the basis of licensing basis evaluations. The present and future irradiation results are and will be considered in the design of the fuel rods with ZIRLO™ cladding to assure that all fuel rod design bases are satisfied for the planned irradiation life of the Indian Point Unit 3 fuel assemblies.

3.2.4.4 Chemical/Mechanical Properties

This chemical composition (see Table 3.2-6) of the ZIRLO™ clad fuel rods in the Indian Point 3 fuel assemblies is similar to Zircaloy-4 except for slight reductions in the content of tin (Sn), iron (FE), and zirconium (Zr) and the elimination of chromium (Cr). ZIRLO™ cladding also contains a nominal amount of niobium (Nb). These small composition changes are responsible for the improved corrosion resistance compared to Zircaloy-4. The physical and mechanical properties are very similar to Zircaloy-4 while in the same metallurgical phase. However, the temperatures at which the metallurgical phase changes occur are different for Zircaloy-4 and ZIRLO™ cladding (Appendix A of Reference 67). These differences are considered in the evaluations discussed below for cladding behavior under non-LOCA and LOCA conditions. Further aspects of the ZIRLO™ cladding performance under LOCA conditions are given in Reference 67). Evaluations have been performed using the NRC approved fuel rod performance code (Reference 70) to verify that the fuel rod design bases and design criteria are met for assemblies containing ZIRLO™ clad fuel rods. The fuel rod design bases, criteria and models, which are affected by the use of ZIRLO™ cladding are described in Reference 67.

3.2.4.5 Neutronic Performance

The design and predicted nuclear characteristics of fuel rods with ZIRLO™ cladding are similar to those of VANTAGE 5 design (Reference 69). The evaluations have shown (Reference 67) that the nuclear design bases are satisfied for fuel rods with ZIRLO™ cladding and that the use of ZIRLO™ cladding will not affect the standard nuclear design analytical models and methods to accurately describe the neutronic behavior of fuel rods with ZIRLO™ cladding. The safety limit characteristics of the VANTAGE 5 fuel design (Reference 69) are not affected.

3.2.4.6 Thermal and Hydraulic Performance

The thermal and hydraulic design bases for fuel rods with ZIRLO™ cladding are identical to those of the VANTAGE 5 design (reference 69). Since the use of the ZIRLO™ clad fuel does not cause changes affecting the parameters which are major contributors in this area (i.e., DNB, core flow, and rod bow), the design bases of the VANTAGE 5 design (Reference 69) remain valid.

3.2.5 VANTAGE + Design Features

The design features of the V+ fuel design include the following

- Current Vantage 5 (w/o IFMs) Fuel Features
  - Reconstitutable Top Nozzle (RTN)
  - Extended Burnup Fuel Assembly Design
  - Extreme Low Leakage Loading Pattern

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  - Enriched Integral Fuel Burnable Absorbers (IFBAs)
  - Debris Filter Bottom Nozzle (DFBN)
  - Axial Blankets
- ZIRLO™ fuel cladding
- Low Pressure Drop (LPD) Mid grids
- Integral Flow Mixing grids (IFMs)
- ZIRLO™ guide thimble and instrument tubes
- Variable Pitch Fuel Rod Plenum Spring
- Mid-enriched Annual Fuel Pellets in Axial Blanket
- Fuel Assembly and Fuel Rod Dimensional Modifications
- Low Cobalt Top and Bottom Nozzles
- Coated Cladding (Pre-oxidized ZIRLO™ cladding)
- Gripable Top End Plug

3.2.5.1 VANTAGE 5 Fuel Features

The use of ZIRLO™ fuel cladding was implemented in the VANTAGE 5 fuel design beginning in Cycle 9 after issuance of the V5 SER.

With respect to the Extended Burnup Fuel Assembly Design, the V+ fuel design includes an increase in the region average discharge burnup from 40,000 + to 50,000 + MWD/MTU. The effects of the increase in extended burnup on the performance of the fuel are included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process.

3.2.5.2 Mid-enriched Annular Fuel Pellets in Axial Blankets and Enriched IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking toward the middle of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of IFBAs which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and the time in core life. The effects on the reload safety analysis parameters due to axial blankets, including annular fuel pellets in the axial blanket, and IFBAs, including enriched IFBAs, are taken into account in the reload design process. The axial and radial power distribution assumptions used in the safety analysis kinetics calculations have been determined to be applicable for evaluating the axial blankets and IFBAs in the IP3 V+ fuel design.

3.2.5.3 LPD Mid Grids, IFM ZIRLO Guide Thimbles and Instrument Tubes, Low Cobalt Top and Bottom Nozzles and Fuel Assembly and Fuel Rod Dimensional Modifications.

The IP3 V+ fuel design incorporates the use of ZIRLO™ LPD mid grids, IFMs, guide thimbles and instrument tubes, and includes minor fuel assembly and fuel rod dimensional modifications to accommodate the Extended Burnup Fuel Assembly design.

With respect to the non-LOCA accident analysis, the use of ZIRLO™ guide thimbles and instrument tubes, low cobalt top and bottom nozzles and the minor fuel assembly and fuel rod dimensional modifications have no direct effect on the analysis results since the characteristics of these features are not specifically modeled in the transient analysis. Any effects of these items on the performance of the fuel are included in the safety analysis via the reload safety analysis parameters (e.g., fuel temperatures, flow rates, pressure drops) which are taken into account in the reload design process.

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The use of the LPD mid grids in the V+ fuel design are primarily for the purpose of offsetting the increase in flow resistance associated with the introduction of the IFMs. Therefore, with respect to the parameters which affected the non LOCA accident analysis, the V+ fuel design is hydraulically compatible with the resident V5 fuel design and has no direct effect on the non LOCA safety analysis. Any localized assembly to assembly hydraulic differences which may affect the performance of the fuel are included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process and documented in the Reload Safety Evaluation.

With respect to the implementation of IFMs V+ fuel design, it should be noted that although the use of IFMs provides a DNB benefit caused by enhanced flow mixing, the benefit is not fully realized in the safety analysis since the resident V5 fuel design does not include IFMs and therefore is more limiting with respect to DNB. As a result, the Core Thermal Limits and resulting OTΔT and OPΔT reactor trip setpoints applicable for the transition to VANTAGE + Fuel are currently based on the DNB performance of the more limiting VANTAGE 5 (w/o IFMs) fuel. Once the transition to a full core V+ fuel is completed, the DNB benefit associated with the IFMs can be fully realized.

3.2.5.4 Variable Pitch Fuel Rod Plenum Spring

The optimized fuel rod plenum spring feature of the V+ fuel design is primarily to provide increased fuel rod plenum volume which benefits rod internal pressure concerns. Any effects of this spring design on the performance of the fuel is included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process.

3.2.5.5 15x15 Upgrade Design

Beginning with Cycle 14, a modified fuel design, designated "15x15 Upgrade", was introduced to the Indian Point Unit 3 Core (IP3).

Fuel cladding is Zirlo on all new fuel assemblies (Region 16A and 16B). ZIRLO is also used on intermediate grid straps, IFMs and thimble tubes. The Westinghouse model designation for the new Cycle 14 is 15x15 Upgrade Fuel with I-Spring (hereafter called 15x15 Upgrade). The remaining fuel in the Cycle 14 core is model VANTAGE+ with Performance Plus features. The IP3 core will be fully loaded with 15x15 Upgrade fuel in Cycle 15.

The 15x15 Upgrade fuel design includes a number of new features:

1) I-Spring mid-grid design to reduce grid-to-rod fretting
2) Enhanced Intermediate Flow Mixers (IFMs) to reduce grid-to-rod fretting
3) Balanced vane pattern on all grids to provide more balanced flow for vibration reduction.
4) Tube-in-tube guide thimble design, replacing the Vantage+ dashpot design. The new guide thimbles are designed to improve margin to Incomplete Rod Insertion (IRI).

In addition to these features, the 15x15 Upgrade Fuel includes fuel performance and reliability features that have been added to IP3 fuel in previous cycles. These include: debris resistant bottom nozzles, fuel rod oxide coatings, protective bottom grid and bead-blasted top nozzle spring screws. However, the grade of inconel used in the spring screws will be Type 718, which is stronger than the previously-used Type 600.

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TABLE 3.2-1

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 14

		Cycle 1	Cycle 14
STRUCTURAL CHARACTERISTICS			
1.	Fuel Weight (UO_2), MTU	88.60	88.6
2a.	Zircaloy Weight, lbs	44,450	N/A
2b.	ZIRLO™ Weight, lbs	N/A[2]	< 41,000 (per design basis assumption of hydrogen release)
3.	Core Diameter, inches	132.71	Same as Cycle 1
4.	Core Height, inches	144	Same as Cycle 1
	Reflector Thickness and Composition		
5.	Top – Water Plus Steel	~10 inches	Same as Cycle 1
6.	Bottom – Water Plus Steel	~10 inches	Same as Cycle 1
7.	Side – Water Plus Steel	~ 15 inches	Same as Cycle 1
8.	H_2O/U, (cold) Core (volume)	3.99	N/C[1]
9.	Number of Fuel Assemblies	193	Same as Cycle 1
10.	UO_2 Rods per Assembly	up to 204	204
PERFORMANCE CHARACTERISTICS			
11.	Heat Output, MWt (initial rating)	3025	3216
12	Heat Output, MWt (maximum calculated turbine rating)	3216	3216
13.	Fuel Burnup, MWD/MTU	17,346	25,000 (predicted)

[1] Not Calculated for Cycle 14
[2] Not Applicable to Cycle 1

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TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 14)

		Cycle 1	Cycle 14
	Enrichments, w/o[3]		
14.	Region 1	2.28	(14B) 4.950
15.	Region 2	2.80	(15A) 4.950
16.	Region 3	3.30	(16A) 4.600 (16B) 4.950
17.	Equilibrium Enrichment	3.20	4.41
18.	Nuclear Heat Flux Hot Channel Factor, F_Q^N	2.56[4]	2.50
19.	Nuclear Enthalpy Rise Hot Channel Factor, $F^N)_H$	1.55	1.70
CONTROL CHARACTERISTICS			
	Effective Multiplication (Beginning of Life) With Rods in; No Boron		
20.	Cold, No Power, Clean	1.197	N/C[5]
21.	Hot, No Power, Clean	1.144	N/C[5]
22.	Hot, Full Power, Clean	1.131	N/C[5]
23.	Hot, Full Power, Xe & Sm Equilibrium	1.091	N/C[5]
24.	Material	5% Cd; 15% In; 80% Ag	Same as Cycle 1
25.	Full Length	53	Same as Cycle 1

[3]Cycle 14 Blankets: Region 14, 15, and 16 are 3.2 w/o.
[4] Nuclear peaking factor limits were revised in response to ACRS concerns by a generic peaking factor envelope for Cycle 1 operation.
[5] Not Calculated for Cycle 14

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TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 14)

		Cycle 1	Cycle 14
26.	Partial Length	Removed	None
27.	Number of Absorber Rods per RCC Assembly	20	Same as Cycle 1
28.	Total Rod Worth, BOL, %	See Table 3.2-2	See Table 3.2-2
29.	Fuel Loading Shut down; Rods in (k = 0.85) (k = 0.90) (k = 0.95)	2000 ppm 1810 ppm N/A[7]	N/C[6] N/C[6] 1747 ppm[8]
30.	Shutdown (k=0.99) with Rods Inserted, Clean, cold	1000 ppm	N/C[6]
31.	Shutdown (k=0.99) with Rods Inserted, Clean, Hot	548 ppm	N/C[6]
32.	Shutdown (k=0.99) with No Rods Inserted, Clean, Cold	1500 ppm	N/C[6]
33.	With No Rods Inserted, Clean, Hot to Maintain k=1 at HFP, BOL	1476 ppm	1365 ppm
34.	Clean	1228 ppm	N/C[6]
35.	Xenon Eq.	934 ppm	983 ppm (150 MWD)
36.	Xenon and Samarium, Eq. (1000 MWD)	899 ppm	1022 ppm
37.	Shutdown, All But One Rod Inserted, Clean, Cold	1099 ppm (k=0.99)	1206 ppm (k = 0.987, BOL)
38.	Shutdown, All But One Rod Inserted, Clean, Hot	669 ppm (k=.99)	713 ppm (k=.987, BOL)

[6] Not Calculated for Cycle 14
[7] Not Applicable to Cycle 1
[8] Calculated Value – Actual value used is 2050 ppm per the COLR.

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TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 14)

		Cycle 1	Cycle 14
BURNABLE POISON RODS			
39.	Number and Material	1434 Borosilicate Glass	128 Unclad Hafnium 1264 WABA 10912 IFBA
40.	Worth Hot, !p	10.0%	N/C[1]
41.	Worth Cold, !p	8.0%	N/C[1]
KINETIC CHARACTERISTICS			
42.	Moderator Temperature Coefficient at Hot Full Power, pcm/°F	-3 to -35	-3 to -30
43.	Moderator Pressure Coefficient, psi^{-1}	$0.3x10^{-6}$ to $4.0x10^{-6}$	N/C[1]
44.	Moderator Density Coefficient, $\Delta k \cdot cm^3/gm$	-0.1 to 0.47	N/C[1]
45.	Doppler Coefficient, pcm/°F	-1.0 to -2.0	-1.63 (BOL, HZP)
46.	Delayed Neutron Fraction, %	0.44 to 0.72	0.51 to 0.62
47.	Prompt Neutron Life, seconds	$1.4x10^{-5}$ to $2.6x10^{-5}$	$1.30x10^{-5}$ to $1.56x10^{-5}$

[1] Not Calculated for Cycle 14

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TABLE 3.2-2

REACTIVITY REQUIREMENTS FOR CONTROL RODS
(For Cycle 1 and Cycle 14)

Requirement	Percent Δp Beginning of Life[10]	Percent Δp End of Life
CYCLE 1		
Control Banks		
Total Control	2.31	3.43
Shutdown Banks		
Shutdown	1.00	1.72
Total Requirement	3.31	5.15
CYCLE 14		
Control Groups		
Total Control Bank Requirement	2.04	2.95
Control Rod Worth (HZP)		
All Rods Inserted Less Most Reactive Stuck Rod[11]	5.97	6.08
(2) Less 10%	5.37	5.47
Shutdown Margin		
Calculated Margin [(3)-(1)]	3.33	2.52
Required Shutdown Margin	1.3	1.3

Note: A) Rod worths are calculated at ARO, HFP boron concentration.
B) T_{mod} at 547.0°F at HZP and 569.0°F at HFP.

[10] 150 MWD/MTU.
[11] N-11 is the most reactive stuck rod at BOL (0.85%Δp) and K-10 is the most reactive stuck rod at EOL (0.76%Δp).

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TABLE 3.2-3

CALCULATED ROD WORTHS, Δρ
(For Cycle 1 and Cycle 14)

CYCLE 1

Core Condition	Rod Configuration	Worth*	Less 10%**	Design Reactivity Requirements	Shutdown Margin
BOL, HFP	53 Rods in	9.76%			
BOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	7.75%	6.97%	2.31%	4.66%
EOL, HFP	53 Rods in	9.55%			
EOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	7.57%	6.81%	3.43%	3.38%

CYCLE 14

Core Condition	Rod Configuration	Worth*	Less 10%**	Reactivity Requirements	Shutdown Margin
BOL, HFP	53 Rods in	7.51%			
BOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	5.97%	5.37%	1.30%	3.33%
EOL, HFP	53 Rods in	8.23%			
EOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	6.08%	5.47%	1.30%	2.52%

Legend

BOL = Beginning-of-Life
EOL = End-of-Life
HFP = Hot Full Power
HZP = Hot Zero Power

* The values for worth are for rods at the insertion limit.
** Calculated rod worth is reduced by 10% to allow for uncertainties.

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TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 14 (Stretch Power Uprate))

	Cycle 1	Cycle 14 (SPU)
Total Heat Output, MWt	3025	3216
Total Heat Output, Btu/hr	$10{,}324 \times 10^6$	10973×10^6
Heat Generated in Fuel, %	97.4	Same as Cycle 1
Nominal System Pressure, psia	2250	Same as Cycle 1
Hot Channel Factors		
Heat Flux		
Engineering, F_q^E	1.03	Same as Cycle 1
Total, F_q^{T*}	2.32	2.50
Enthalpy Rise – Nuclear $F_{\Delta H}^N$	1.55	1.70
Total Flow Rate, lbm/hr	130.2×10^6	134.8×10^6
Average Velocity Along Fuel Rods, ft/sec	14.9	14.2
Average Mass Velocity, lbm/hr•ft^2	2.43×10^6	2.42×10^6

*The total heat flux hot channel factor is a generic limit. The actual value is documented in the COLR.

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TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 14 (Stretch Power Uprate))

	Cycle 1	Cycle 14
Coolant Temperature, °F		
Nominal Inlet	541.2	541.0
Average Rise in Vessel	60.5	62.0
Average Rise in Core	63.1	66.5
Average in Core	574.1	575.8
Average in Vessel	571.5	572.0
Nominal Outlet of Hot Channel	635.7	N/C*
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	52,200	52,100
Average Heat Flux Btu/hr•ft^2	193,000	205,200
Maximum Heat Flux, Btu/hr•ft^2	448,000	513,000
Maximum Thermal Output, $^{kw}/_{ft}$	12.7	16.6
Maximum Clad Surface Temperature BOL at Nominal Pressure, °F	657	N/C*
Maximum Average Clad Temperature BOL at Rated Power, °F	715	N/C*

*Not Calculated for Cycle 14.

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TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 14 (Stretch Power Uprate))

	Cycle 1	Cycle 14 (SPU)
Fuel Central Temperatures for nominal Fuel rod dimensions, °F		
Maximum at 100% Power	4100	3670*
Maximum at Overpower Power	4350 (112% power)	4250* (120% power)
DNB Ratio		
Minimum DNB Ratio at nominal operating Conditions	1.80	2.50
Pressure Drop, psi		
Across Core	21	26

*The fuel central temperatures were calculated using the new PAD model (Ref. 80)

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TABLE 3.2-5

CORE MECHANICAL DESIGN PARAMETERS[1]
(For Cycle 1 and Cycle 14)

	Cycle 1	Cycle 14
Active Portion of the Core		
Equivalent Diameter, inches	132.7	Same as Cycle 1
Active Fuel Height, inches	144.0	Same as Cycle 1
Length-to-Diameter Ratio	1.09	Same as Cycle 1
Total Cross-Section Area, ft^2	96.06	Same as Cycle 1
Fuel Assemblies		
Number	193	Same as Cycle 1
Rod Array	15 x 15	Same as Cycle 1
Rods per Assembly	204*	Same as Cycle 1
Rod Pitch, inches	0.563	Same as Cycle 1
Overall Dimensions, inches	8.426 x 8.426	Same as Cycle 1
Fuel Weight, MTU	88.60	86.60
Number of Grids per Assembly	7	10
Number of guide Thimbles	20	Same as Cycle 1
Diameter of Guide Thimbles (upper part) inches	0.545 O.D. x 0.515 I.D.	0.533 O.D. x 0.499 I.D.
Diameter of Guide Thimbles (lower part) inches	0.484. O.D. x 0.454 I.D.	0.490 O.D. x 0.455 I.D.
Fuel Rods		
Number	39,372	Same as Cycle 1
Outside Diameter, inches	0.422	Same as Cycle 1
Diametral Gap, inches	0.0075	0.0075**
Clad Thickness, inches	0.0243	Same as Cycle 1
Clad Material	Zircaloy-4	ZIRLO™
Overall Length	151.8	156.3
Length of End Cap, overall, inches	0.688	0.450 (top), 0.680 (bottom)
Length of End Cap, inserted in rod, inches	0.250	0.13

NOTE: All dimensions are for cold conditions.
*Twenty-one rods are omitted: Twenty provide passage for control rods and one to contain in-core instrumentation.
**Nominal gap.

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TABLE 3.2-5
(Cont.)

CORE MECHANICAL DESIGN PARAMETERS[1]
(For Cycle 1 and Cycle 14)

	Cycle 1	Cycle 14
Fuel Pellets		
Material	UO_2 sintered	Same as Cycle 1
Density (% of Theoretical)		
Region 1, 2, & 3 (typical of reload regions)	95 (10.4 g/cc)	
Region 13A & 13B		95.5
Feed Enrichments w/o		
Region 1	2.28	(14B) 4.950
Region 2	2.80	(15A) 4.950
Region 3	3.30	(16A) 4.600 (16B) 4.950
Diameter, inches		
Region 1, 2, & 3	0.3659	Same as Cycle 1
Length, inches	0.600	0.439
Rod Cluster Control Assemblies		
Neutron Absorber	5% Cd, 15% In, 80% Ag	Same as Cycle 1
Cladding Material	Type 304 SS-Cold Worked	Same as Cycle 1
Clad Thickness, inches	0.019	Same as Cycle 1
Number of Clusters		
Full Length	53	Same as Cycle 1
Number of Control Rods per Cluster	20	Same as Cycle 1
Length of Control Rod, inches	156.436 (overall)	Same as Cycle 1
	149.136 (insertion length)	Same as Cycle 1
Length of Absorber Section, inches	142.000	Same as Cycle 1

*Note: All dimensions are for cold conditions.

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TABLE 3.2-5
(Cont.)

CORE MECHANICAL DESIGN PARAMETERS[1]
(For Cycle 1 and Cycle 14)

	Cycle 1	Cycle 14	
Core Structure			
Core Barrel, inches			
I.D.	148.0	Same as Cycle 1	
O.D.	152.5	Same as Cycle 1	
Thermal Shield, inches			
I.D.	158.5	Same as Cycle 1	
O.D.	164.0	Same as Cycle 1	
Burnable Poison Rods and Flux Suppressors			
Material	Borosilicate Glass	IFBA	
		WABA	
		Hafnium	
Number	1434 (first cycle)	10912	(IFBA)
	1056	1264	(WABA)
		128	(Hafnium)
		12304	Total
Outer Tube, inches (outer diameter)	0.4390	N/A	(IFBA)
		0.532	(WABA)
		0.533	(Hafnium)
Inner Tube, inches (outer diameter)	0.2365	N/A	(IFBA)
		0.381	(WABA)
		N/A	(Hafnium)
Clad Material	Type 304 SS	N/A	(IFBA)
		ZIRLO	(WABA)
		N/A	(Hafnium)
Inner Tube Material	Type 304 SS	N/A	(IFBA)
		ZIRLO	(WABA)
	Type 304 SS	N/A	(Hafnium)
Boron loading (natural) gm/cm	0.0576	0.001043	(IFBA)
		0.00603	(WABA)
		N/A	(Hafnium)

*Note: All dimensions are for cold conditions.

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TABLE 3.2-6

NOMINAL COMPOSITION OF ZIRLO™ AND ZIRCALOY-4 CLADDING

Element	Zircaloy-4 (wt %)	ZIRLO™ (wt %)
Sn	1.6	1.0
Fe	0.21	0.1
Cr	0.1	0.0
Nb	0.0	1.0
Zr	>97.0	>97.0

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3.3 TESTS AND INSPECTIONS

3.3.1 Physics Tests

1. Tests to Confirm Reactor Core Characteristics

   A detailed series of startup physics tests were performed from zero power up to and including 100% power. As part of these tests, a series of core power distribution measurements were made by means of the core movable detector system. These measurements were analyzed and the results compared with the analytical predictions upon which safety analyses were based.

2. Tests Performed During Operation

   To detect and eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, is normalized to accurately reflect actual core conditions. When full power is reached initially, and with the control groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation continues, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted and corrected if necessary. This normalization is completed before a cycle burnup of 60 Effective Full Power Days (EFPD) is reached. Thereafter, actual boron concentration can be compared with the predicted concentration, and the reactivity prediction of the core can be continuously evaluated and adjusted.

   Any reactivity anomaly greater than one percent would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

3.3.2 Thermal and Hydraulic Tests and Inspections

General hydraulic tests on models have been used to confirm the design flow distributions and pressure drops (1,2). Fuel assemblies and control and drive mechanisms are also tested in this manner. Appropriate on-site measurements are made to confirm the design flow rates.

Vessel and vessel internals inspections were also reviewed to confirm such thermal and hydraulic design values as bypass flow.

A summary report of appropriate plant testing shall be submitted to the NRC following (1), an amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the testing and comparison of these values with acceptance criteria.

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Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup programs, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

3.3.3 Core Component Tests and Inspections

To ensure that all materials, components and assemblies conformed to the design requirements, a release point program was established with the manufacturer. This required surveillance of raw materials, special processes, i.e., welding, heat treating, nondestructive testing, etc. and those characteristics of parts which directly affect the assembly and alignment of the reactor internals. The surveillance was accomplished by the issuance of Inspection Reports by the quality control organization after conformance had been verified.

A resident quality control representative performed a surveillance/audit program at the manufacturer's facility and witnessed the required tests and inspections and issued the inspection reports.

Components and materials supplied by Westinghouse to the assembly manufacturer were subjected to a similar program. Quality Control engineers developed inspection plans for all raw materials, components, and assemblies. Each level of manufacturing was evaluated by a qualified inspector for conformance, i.e., witnessing the ultrasonic testing of core plate raw material. Upon completion of specified events, all documentation was audited prior to releasing the material or component for further manufacturing. All documentation and inspection releases are maintained in the quality control central records section. All materials are traceable to the mill heat number.

In conclusion, a set of "as built" dimensions were taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.

3.3.2.1 Fuel Quality Assurance

Nuclear fuel is purchased under the Authority's Quality Assurance Program. Surveillance audits of the fuel fabrications and the Westinghouse Quality Assurance Program are performed by the Authority.

Quality Assurance Program

The Westinghouse Nuclear Fuel Division's quality assurance program plan is included in Reference 3 that is summarized below.

The program provides for control over all activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage and transportation. The program also provides

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for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawing and product, process, and material specifications identify the inspections to be performed.

Quality Control

Quality Control (QC) philosophy is based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted:

1) Fuel system components and parts.

   The characteristics inspected depend upon the component parts; the QC program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

   All material used in this core is accepted and released by QC.

2) Pellets.

   Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips and surface conditions according to approved standards.

   Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a specified sample basis throughout pellet production.

3) Rod inspection.

   The fuel rod inspection consists of the following nondestructive examination techniques and methods, as applicable:

   a) Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.

   b) All weld enclosures are X-rayed or Ultrasonically tested. X-rays are taken in accordance with Westinghouse specifications meeting the requirements of ASTM-E-142.

   c) All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.

   d) All of the fuel rods are inspected by X-ray or other approved methods to ensure proper plenum dimensions.

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e) All of the fuel rods are inspected by gamma scanning, or other approved methods to ensure that no significant gaps exist between pellets.

f) All fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.

g) Traceability of rods and associated rod components is established by QC.

4) Assemblies:

Each fuel assembly is inspected for compliance with drawing and/or specification requirements. Other incore control component inspection and specification requirements are given in Reference 3.

5) Other inspections:

The following inspections are performed as part of the routine inspection operation:

a) Tool and gage inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.

b) Audits are performed of inspection activities and records to ensure that prescribed methods are followed and that records are correct and properly maintained.

c) Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

6) Process control:

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The UO_2 power is kept in sealed containers. The contents are fully identified by labels completely describing the contents. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by QC. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Substandard pellets are reprocessed and utilized for recycle material.

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A serialized traceability code label is electroetched into the tube, to provide unique identification. This identification is maintained and used throughout assembly fabrication and inspection.

Loading of pellets into the clad, which previously had bottom end plug welded in place, is performed in isolated production lines, and again only one density and enrichment are loaded on a line at a time except that the axial blanket pellets are usually a lower enrichment.

The "annular" axial blanket pellets are sintered with a hole in the center to increase plenum volume. The axial blanket pellets are carefully controlled on separate trays located in separate cabinets and have the added characteristic of the centerline hole to minimize improper placement in the pellets stacks.

The loading of IFBA burnable poison rods containing ZrB_2 coated pellets is performed in a separate production line physically separated from standard UO_2 production line. All other operations are similar with the same close control to prevent the misloading of pellets with different enrichment.

The top end plugs are inserted and then welded to seal the tube. At the time of installation into an assembly, a matrix is generated to identify each rod's position within a given assembly. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

The preceding discussion and the references stated describe the efforts in the application of quality control and the use of reliability techniques during the development, design, fabrication, and shipment of fuel assemblies.

Upon delivery of the fuel to the site, an inspection is performed for evidence of shipping damage, loose parts and debris. The shipping container internals, seals, container bolts, clamping fixture, and shock overload probe indicators are inspected. After removal from the container, the fuel itself is inspected for evidence of shipping damage to approved Inspection Procedures. The above inspections, as a minimum, are accomplished for all fuel assemblies and are appropriately documented.

3.3.2.2 Burnable Poison Rod Tests and Inspections

The end plug seal welds are checked for integrity by visual inspection, and X-ray or Ultrasonic testing. The finished rods are helium leak checked.

REFERENCES

1. Hetsroni, G., "Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, 1964.

2. Hetsroni, G., "Studies of the Connecticut-Yankee Hydraulic Model," WCAP-2761, 1965.

3. Moore, J., "Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800-Revision 5, November 1979.

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CHAPTER 4

REACTOR COOLANT SYSTEM

4.1 DESIGN BASIS

The Reactor Coolant System, shown in Plant Drawings 9321-F-27338 and -27473 [Formerly Figure 4.2-2A & -2B], consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control.

4.1.1 Performance Objectives

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance presented in Chapter 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material, and limits, to acceptable values, its uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation the system's heat capacity attenuates thermal transients. The Reactor Coolant System accommodates coolant volume changes within the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal-hydraulic effects are reduced to a safe level during the pump coast-down which would result from a loss of flow situation. The layout of the system assures natural circulation capability following a loss of flow to permit decay heat removal without overheating the core. Part of the system's piping is used by the Safety Injection System to deliver cooling water to the core during a Loss-of-Coolant Accident.

4.1.2 General Design Criteria

General design criteria which apply to the Reactor Coolant System are given below.

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1976, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980,

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and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1 of 7/11/67)

The Reactor Coolant System is of primary importance in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication and inspection conformed to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.7). Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Section 4.3.1 and Section 4.5. Particular emphasis was placed on the quality assurance of the reactor vessel to obtain material whose properties were uniformly within tolerances appropriate to the application of the design methods of the code.

Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces, that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect; (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2 of 7/11/67)

All piping, components and supporting structures of the Reactor Coolant System were designed to seismic Class I requirements (SEE Chapter 16). They are capable of withstanding:

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a) The Operational Basis Earthquake accelerations within code allowable working stresses.

b) The Design Basis Earthquake accelerations acting in the horizontal and vertical direction simultaneously with no loss of function.

The Reactor Coolant System is located in the Containment, whose design, in addition to being a seismic Class I structure, also considered accidents or other applicable natural phenomena with sufficient margin to compensate for the limits of accuracy on measurements, the quantity of data and the period of time in which historical data have been accumulated on the natural phenomena. Details of the containment design are given in Chapter 5.

Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5 of 7/11/67)

Records of the design of the major Reactor Coolant System components and the related engineered safety features components are maintained by the Authority and will be retained throughout the life of the plant as dictated by plant procedures.

Records of fabrication were maintained in the manufacturers' plants as required by the appropriate code. They will be available to the Authority throughout the life of the plant. Construction records are available at the site and/or in the office of the Authority where they will be retained for the life of the plant.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40 of 7/11/67)

The dynamic effects during blowdown following a Loss-of-Coolant Accident were evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Support structures were designed with consideration given to fluid and mechanical thrust loadings.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the Engineered Safety Features is not impaired.

4.1.3 Principal Design Criteria

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The criteria which apply solely to the Reactor Coolant System are given below.

Reactor Coolant Pressure Boundary

Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (GDC 9 of 7/11/67)

The Reactor Coolant System in conjunction with its control and protective provisions was designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System was carried out in strict accordance with the applicable codes. In addition, there were areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes. Details are given in Section 4.5.1.

The materials of construction of the pressure retaining boundary of the Reactor Coolant System are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible.

The system is protected form overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolatable sections of the system are provided with overpressure relieving devices discharging to closed systems, such that the system code allowable relief pressure within the protected section is not exceeded.

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16 of 7/11/67)

Positive indications in the Control Room of leakage of coolant from the Reactor Coolant System to the Containment are provided by equipment which permits continuous monitoring of containment air activity and humidity, and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. This equipment provided indication of normal background which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the Containment, and the equipment provided is capable of monitoring this change. The basic design criterion was the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, and condensate runoff. In addition, assuming no operator action, the liquid inventory in the process systems and containment sump can be used for gross indication of leakage. However, sensitivity of the processing systems and containment sump system can be

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improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours).

Further details are supplied in Section 4.2.7.

Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33 of 7/11/67)

The Reactor Coolant Pressure Boundary is capable of accommodating, without further rupture, the static and dynamic loads which would be imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Chapter 14.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since core depletion is primarily followed with boron dilution, only the rod cluster control assemblies in the controlled groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to assure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the primary system pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core was evaluated as a theoretical, thought not a credible accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the Reactor Coolant System and the reactor containment. The environmental consequences of rod ejection are less severe than from the hypothetical loss of coolant, for which public health and safety is adequately protected. (See Chapter 14)

Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given: (a) to the provision for control over service temperature and irradiation effects which may require operational restrictions; (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation; and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GC 34 of 7/11/67)

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The Reactor Coolant Pressure Boundary was designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

Assurance of adequate fracture toughness in the reactor vessel material was provided by compliance, insofar as possible, with the requirements for fracture toughness testing included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it as not possible to perform all tests in accordance with these requirements, conservative estimates of material fracture toughness were made using information available at the time of design.

Assurance that the fracture toughness properties remain adequate throughout the service life of the plant is provided by a radiation surveillance program.

Safe operating heatup and cooldown limits are established according to Section III, ASME Boiler and Pressure Vessel Code, Appendix G 2000, Protection Against Nonductile Failure, issued in the Summer 1972 Addenda.

Changes in fracture toughness of the core region plates, weldments, and associated weld heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels[1]. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, dropweight test, and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. Further details are given in Section 4.1.6.

All pressure containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

Reactor Coolant Pressure Boundary Surveillance

Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leak-tight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided. (GDC 36 of 7/11/67)

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. Further details are given in Section 4.5.

4.1.4 Design Characteristics

Design Pressure

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The Reactor Coolant System design and operating pressures together with the safety, power relief and pressurizer spray valves set points, and the protection system set point pressures are listed in Table 4.1-1. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-2 through 4.1-6. Table 4.1-7 gives the design pressure drop of the system components.

Design Temperature

The design temperature for each component was selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 4.1-2 through 4.1-6.

Seismic Loads

The seismic loading conditions were established by the "Operational Basis Earthquake" and "Design Basis Earthquake". The former was selected to be typical of the largest probable ground motion based on the site seismic history. The latter was selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the "Operational Basis Earthquake" (OBE) loading condition, the Nuclear Steam Supply System was designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose were required to operate within normal design limits. The seismic design for the "Design Basis Earthquake" (DBE) was intended to provide a margin in design that assured capability to shut down and maintain the nuclear facility in a safe condition. In this case, it was only necessary to ensure that the reactor coolant system components did not lose their capability to perform their safety function. This is referred to as the "no-loss-of-function" criteria and the loading condition as the "no-loss-of-function earthquake" loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Chapter 16. These criteria assure the integrity of the Reactor Coolant System under seismic loading.

For the combination of normal and Operational Basis Earthquake loadings, the stresses in the support structures were kept within the limits of the applicable codes.

For the combination of normal and Design Basis Earthquake loadings the stresses in the support structures were limited to values as necessary to assure their integrity and to maintain the stresses in the Reactor Coolant System components within the allowable limits as previously established.

4.1.5 Cyclic Loads

All components in the Reactor Coolant System were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown

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operation. The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes (40 year life) and are not intended to be an accurate representation of actual transients or actual operating experience.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses were based upon engineering judgement and experience. Those transients were chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 40 years operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

1. Heatup and Cooldown

   For design evaluation, the heatup and cooldown cases were represented by continuous heatup or cooldown at a rate of 100 F per hour which corresponds to a heatup or cooldown rate under abnormal or emergency conditions. The heatup occurs from ambient to the no load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100 F per hour will not be usually attained because of the Over Pressure Protection System (OPS) discussed in Section 4.3 and other limitations such as:

   a) Criteria for protection against non-ductible failure which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature

   b) Slower initial heatup rates when using pumping energy only

   c) Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

   The heatup and cooldown rates, administratively imposed by plant operating procedures, are limited to 50 F per hour for normal operation. Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional scheduled and unscheduled plant cooldowns may be necessary for plant maintenance.

2. Unit Loading and Unloading

   The unit loading and unloading cases were conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature varies with load as

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prescribed by the temperature control system. The number of each operation was specified at 14,500 times or once per day for the 40-year plant design life. In practice, the plant is operated at base load.

3. Step Increase and Decrease of 10%

The ±10% step change in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The Reactor Control System was designed to restore plant equilibrium without reactor trip following a ±10% step change in demand. The turbine load power range for automatic reactor control initiated from nuclear plant equilibrium conditions, is in the range between 15% and 100% of full load. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs so that peak reactor coolant temperature is minimized. Concurrently, the reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the Reactor Coolant System average temperature and pressurizer pressure also initially increase.

Because of the power mismatch between the turbine and reactor, the increase in reactor coolant temperature is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage (inlet) turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant's decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease.

Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature is raised to a value above its initial equilibrium value at the beginning if the transient.

The number of each operation was specified at 2000 times or 50 per year for the 40-year plant design life.

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4. Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature automatically initiates a secondary side steam dump that prevents a reactor shutdown or the lifting of steam generator safety valves.

This transient capability definition brackets the transient design bases used for the Regulating Systems as discussed in Section 7.3.

The number of occurrences of this transient was specified at 200 times or 5 per year for the 40-year plant design life.

5. Reactor Trip from Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the Reactor Coolant System and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam removes the core residual heat and prevents the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor protection system causes the control rods to move into the core.

The number of occurrences of this transient was specified at 400 times or 10 per year for the 40-year plant design life.

6. Hydrostatic Test Conditions

The pressure tests outlined below apply to field pressure tests conducted on the erected Reactor Coolant System. The number of tests given below does not include any allowance for pressure tests conducted on a specific component in the manufacturer's shop in accordance with vessel code requirements.

a. Primary Side Hydrostatic Test before Initial Startup at 3110 psig

This hydrostatic test was performed at a minimum water temperature of 100 F, imposed by a reactor vessel material NDTT value of 100 F at beginning of life, and a maximum test pressure of 3110 psig. In this test, the primary side of the steam generator was pressurized to 3110 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System was designed on the basis of 5 cycles of this hydro test.

b. Reactor Coolant System Leakage Test

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This test is performed at normal operating pressure following each refueling outage prior to startup in accordance with ASME Section XI. Additional tests are performed following repairs, replacements or modifications of the RCS in accordance with ASME Section XI.

7. Loss of Load Without Immediate Turbine or Reactor Trip

This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip, and represents the most severe transient on the Reactor Coolant System. The reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System. Since redundant means of tripping the reactor are provided as a part of the Reactor Protection System, transients of this nature are not expected but are included to insure a conservative design.

8. Loss of Flow

This transient applies to a partial loss of flow from full power in which a Reactor Coolant Pump is tripped out of service as a result of a loss of power to the pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal results in reactor coolant being passed at cold leg temperature through the steam generator and being cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop.

All components in the Reactor Coolant System were designed to withstand the effects of transients that result in system temperature and pressure changes.

Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature were determined for each of these transients through systematic analytical procedures. These stress intensity values S_{ij} (i, j = 1,2,3) were plotted against a time interval for each cycle. This plot may represent one or more stress cycles. For each cycle, extreme values of S_{max} and S_{min} were determined. From these values, the largest S_{alt} (alternating stress intensity) was found.

For this largest value of S_{alt}, an allowable number of cycles (N) was determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (N) gave the usage factor ^{u}i (i = 1,2,3..., n). Usage factor was determined in this manner for all transients. The cumulative usage factor was determined by summing the individual usage factors. The cumulative usage factor ($U = {}^{u}1^{+}\,{}^{u}2^{+u}3^{+}\ldots{}^{+u}n$) was never allowed to exceed a value of 1.0.

Although loss of flow and loss of load transients were not included in the tabulation, since the tabulation was only intended to represent normal design transients, the effects of these transients were analytically evaluated and were included in the fatigue analysis for primary system components.

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Over the range from 15% of full power up to and including, but not exceeding, 100% of full power, the Reactor Coolant System and its components were designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes without reactor trip. The Reactor Coolant System can accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump systems make it possible to accept a 10% to 50% change in load, at a maximum turbine unloading rate of 200% per minute from approximately 100% load with steam dump without reactor trip (load rejection capability depends on full power Tavg; see Section 7.3.2). However, for component stress analysis purposes, this was analyzed as a step change in load from 100% to 50% load.

4.1.6 Service Life

The service life of the Reactor Coolant System pressure components depends upon the end-of-life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the reactor Coolant System which is exposed to a significant level of neutron irradiation and it is, therefore, the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM-185 standards.

Reactor vessel design was based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class "A" vessels, the unit operating conditions were established for the 40-year design life. These operating conditions included the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

4.1.7 Codes and Classifications

All pressure containing components of the Reactor Coolant System were of U.S. manufacture and were designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-9.

The Reactor Coolant System is classified as Class I for seismic design, requiring that there will be no loss of function of such equipment in the event of the assumed maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously, when combine with the primary steady state stresses.

References

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1) WCAP-8475, "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program", Westinghouse Class 3, January 1975.

TABLE 4.1-1

REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valve (Begin to open)	2260
(Fully open)	2310
High Pressure Trip	2385
Low Pressure Trip	1800
Hydrostatic Test Pressure	3110

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TABLE 4.1-2

REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3110
Design Temperature, F	650
Overall Height of Vessel and Closure Head, ft-in (Bottom Head O.D. to top of Control Rod Mechanism Housing)	43-9 $\frac{11}{16}$
Water Volume, (with core and internals in place, ft^3)	4647
Thickness of Insulation, min, in	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in	7
ID of Flange, in	167$\frac{1}{16}$
OD of Flange, in	205
ID at Shell, in	173
Inlet Nozzle ID, in	27½
Outlet nozzle ID, in	29
Clad Thickness, min, in	$\frac{5}{32}$
Lower Head Thickness, min, in	5$\frac{5}{16}$
Vessel Belt-Line Thickness, min, in	8$\frac{5}{8}$
Closure Head Thickness, in	7
Reactor Vessel Inlet Temperature, F	517.3
Reactor Vessel Outlet Temperature, F	611.7
Reactor Coolant Flow, lb/hr	1.388×10^8

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TABLE 4.1-3

PRESSURIZER & PRESURIZER RELIEF TANK DESIGN DATA

Pressurizer	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design/Operating Temperature, F	680/653
Water Volume, Full Power, ft^{3*}	1080
Steam Volume, Full Power, ft^3	720
Surge Line nozzle Diameter, in/Pipe Schedule	14/Sch 140
Shell ID, in/Calculated Minimum Shell Thickness, in	84/4.1
Minimum Clad Thickness, in	0.188
Electric Heaters Capacity, kW	1800
Heatup rate of Pressurizer using Heaters only, F/hr	55 (approximately)
Power Relief Valves	
Number	2
Set Pressure (open), psig	2335
Capacity, lb/hr Saturated steam/valve	179,000
Safety Valves	
Number	3
Set Pressure, psig	2485
Capacity, lb/hr Saturated steam/valve	420,000
Pressurizer Relief Tank	
Design pressure psig	100
Rupture Disc Release Pressure psig	100
Design temperature, F	340
Normal water temperature, F	Containment Ambient
Total volume, ft^3	1800
Rupture Disc Relief Capacity, lb/hr	1.224×10^6

* 60% of net internal volume

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TABLE 4.1-4

STEAM GENERATOR DESIGN* DATA

Number of Steam Generators	4
Design Pressure, Reactor Coolant/Steam, psig	2485/1085
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107
Design Temperature, Reactor Coolant/Stem, °F	650/600
Reactor Coolant Temperature to Steam Generator, °F	602.5
Reactor Coolant Temperature from Steam Generator, °F	540.7
Reactor Coolant Flow, (gpm)	88,600
Total Heat Transfer Surface Area, ft^2	43,467
Heat Transferred at Design (Licensed) Power Level, (807.5 Mwt) Btu/hr	2755×10^6
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, 10^6 lb/hr	3.2914 / 3.4906
Steam Temperature, °F	510.9 / 510.8
Steam Pressure, psia	750.4 / 749.6
Feedwater Temperature, °F	390.0 / 433.6
Overall Height, ft-in	63 – 1.62
Shell OD, upper/lower, in	166/127
Shell Thickness, upper/lower, (minimum), in	3.5/2.62
Number of U-tubes	3214
U-tube Diameter, in	0.875
Tube Will Thickness, (Average) in	0.050
Number of manways/ID, in	4/16
Number of handholes/ID, in	6/6
Number of inspection ports/ID, in	1/3

*The values on this table apply at the design full-load power level of 3230 MWt (four loops) unless noted otherwise. The values shown are for a single steam generator.

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TABLE 4.1-4
(Cont.)

STEAM GENERATOR DESIGN* DATA

	3230 MWt	Zero Power
Reactor Coolant Water Volume ft^3 (cold)	924.1	924.1
Primary Side Fluid Heat Content, Btu	24.1×10^6	23.86×10^6
Secondary Side Water Volume, ft^3**	1626.9	2666
Secondary Side Steam Volume, ft^3**	3100	2061
Secondary Side Fluid Heat Content, Btu**	45.05×10^6	72.8×10^6

* The values on this table apply to the design full load power level of 3230 MWt (four loops) unless noted otherwise. The values shown are for a single steam generator.

** These values correspond to a normal operating water level of 52% narrow range span, and may vary with changes in water level.

*** Values provided for 3230 MWt correspond to a feed temperature of 433.6°F.

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TABLE 4.1-5

REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	4
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design Temperature (casing), F	650
RPM at nameplate Rating	1189
Suction Temperature, F	555
Net Positive Suction Head, Ft	170
Developed Head, ft	272
Capacity, gpm	89,700
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump discharge Nozzle ID, in	27½
Pump Suction Nozzle, ID, in	31
Overall Unit Height, ft	28.38
Water Volume, ft^3	192
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	82,000
Motor Data:	
Type	AC Induction, Single Speed, Air Cooled
Voltage	6600
Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, cps	60

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TABLE 4.1-5
(Cont.)

REACTOR COOLANT PUMPS DESIGN DATA

Starting Current, amp	2950
Input (hot reactor coolant), kW	4250
Input (cold reactor coolant) kW	5600
Power, hp (nameplate)	6000

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TABLE 4.1-6

REACTOR COOLANT PIPING DESIGN DATA

Reactor Inlet Piping, ID, in	27½
Reactor Inlet Piping, nominal thickness, in	2.375
Reactor Outlet Piping, ID, in	29
Reactor Outlet Piping, nominal thickness,	2.50
Coolant Pump Suction Piping, ID, in	31
Coolant Pump Suction Piping, nominal thickness, in	2.625
Pressurizer Surge Line Piping, ID, in	14
Pressurizer Surge Line Piping, nominal thickness,	1.25
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design Temperature, F	650
Design Temperature (pressurizer surge line) F	680
Water Volume (all 4 loops including surge line), ft^3	1156

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TABLE 4.1-7

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop, psi
Across Pump Discharge Leg	1.1
Across Vessel, including nozzles	46.7
Across Hot Leg	1.3
Across Steam Generator	32.3
Across Pump Suction Leg	3.0
Total Pressure Drop	84.4

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TABLE 4.1-8

THERMAL AND LOADING CYCLES

	Transient Condition	Design Cycles+**	Loading Conditions
1.	Plant heatup at 100 F per hour	200 (5/yr*)	Normal
2.	Plant cooldown at 100 F per hour	200 (5/year)	Normal
3.	Plant loading at 5% of full power per minute	14,500 (1/day)	Normal
4.	Plant unloading at 5% of full power per minute	14,500 (1/day)	Normal
5.	Step load increase of 10% of full power (but not to exceed full power)	2,000 (50/year)	Normal
6.	Step load decrease of 10% of full power	2,000 (50/year)	Normal
7.	Step load decrease of 50% of full power	200 (5/year)	Normal
8.	Reactor trip	400 (10/year)	Upset
9.	Hydrostatic test at 3110 psig pressure, 100 F temperature	5 (pre-operational)	Test
10.	Hydrostatic test at 2485 psig pressure and 400 F temperature	200 (post-operational)	Test
11.	Steady state fluctuations – the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6 F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur		
12.	Loss of load, without immediate turbine trip or reactor trip	80 (2 /year)	Upset
13.	Partial loss of flow, one pump only	80 (2/year)	Upset

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TABLE 4.1-8
(Cont.)

THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles+**	Loading Conditions
14. Operating Basis Earthquake (OBE)	5++	Upset
15. Design Basis Earthquake (DBE)	1++	Faulted

\+ Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience. See Section 4.1.5.

* This transient includes pressurizing to 2235 psig.

** Piping and Valves included in the Reactor Coolant System boundary are designed, analyzed and fabricated in accordance with their applicable codes.

++ The upset conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status. The faulted conditions include the earthquake for which safe shutdown is required. For fatigue studies, Class I components were analyzed for five OBE's and one DBE in addition to other fatigue producing events in the above listed four loading conditions. Each earthquake is considered to produce ten peak stress magnitudes.

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TABLE 4.1-9

REACTOR COOLANT SYSTEM – CODE REQUIREMENTS

The edition of the ASME Code, Section III and addenda to which the major components in the Reactor Coolant System were designed and fabricated are:

Component	Code Edition	Class	Applicable Addenda
Reactor Vessel	1965	A	Winter 1965 and Code Cases 1332, 1335, 1339, 1359
Rod Drive Mechanism	1965	A	Summer 1966
Replacement Steam Generators			
- Tube side	1983	1	Summer 1984
- Shell side	1983	1	Summer 1984
Pressurizer	1965	A	Summer 1966
Pressurizer Relief Tank	1965	C	Summer 1966
Pressurizer Safety Valves	1965		Summer 1966
Reactor Coolant Pump Volute	- Westinghouse design per ASME III Article 4.		

In addition the reactor coolant pipe was designed to ANSI B31.1 – 1955.

The loop 32 hot leg elbow replaced during the cycle 6/7 refueling outage in conjunction with the steam generator replacement was designed and fabricated to ASME Code Section III, 1983 edition, class 1 requirements, including addenda through summer 1984, although the elbow was required to meet only B31.1-1955 criteria.

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4.2 SYSTEM DESIGN AND OPERATION

4.2.1 General Description

The Reactor Coolant System consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system and auxiliary system connections is shown in Plant Drawing 9321-F-27383 and -27473 [Formerly Figures 4.2.2A and 4.2-2B]. A schematic flow diagram denoting principal parameters under normal steady state full power operating conditions is shown in Figure 4.2-2.

The containment boundary shown on the flow diagram in Plant Drawing 9321-F-27473 [Formerly Figure 4.2-2B] indicates those major components which are to be located inside the Containment. The intersection of a process line with this boundary indicates a functional penetration.

Reactor Coolant System design data are listed in Tables 4.1-2 through 4.1-6. A power level of 100% rated output for 80% of the time is considered an estimate of ideal operation over the service life of the system.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Chapter 7. Spring-loaded steam safety valves and power operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

4.2.2 Components

Reactor Vessel

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. The vessel was designed in accordance with Section III (Nuclear Vessels) of ASME Boiler and Pressure Vessel Code. Figure 4.2-3 is a schematic of the reactor vessel. The materials of construction of the reactor vessel are given in Table 4.2-1.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. Approximately ninety-five percent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the RCC guide thimbles, the leakage across the outlet nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum, and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

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A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. It is attached to the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield minimizes thermal stresses in the vessel which result from heat generated by the absorption of gamma energy. This protection is further described in Section 3.2.3.

Forty-eight core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by fifty-four 7 in diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leakoff connection is also provided beyond the outer O-ring seal.

The vessel is insulated with metallic reflective-type insulation supported from the nozzles. Insulation panels are provided for the reactor closure head which are supported on the refueling seal ledge and vent shroud support rings.

The reactor vessel internals were designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position. The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control cluster assemblies, surveillance specimens, and incore instrumentation.

Surveillance specimens made from reactor vessel steel are located between the reactor vessel wall and the thermal shield. These specimens are examined at selected intervals to evaluate reactor vessel material NDTT changes as described in Section 4.5.2.

The reactor internals are described in detail in Section 3.2.3 and the general arrangement of the reactor vessel and internals is shown in Figure 3.2-23.

Reactor vessel design data are listed in Table 4.1-2.

Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.2-4, and the design characteristics are listed in Table 4.1-3.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The basis for establishing the pressure relieving capacity for the Reactor Coolant Pressure Boundary is the loss of 100 percent of the heat sink i.e. steam flow to the turbine, when the thermal output of the reactor is at 100 percent of its rated power, with appropriate credit taken for operation of the secondary system safety valves and the Reactor Protection System.

Overpressure protection is described in Section 4.3.4.

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The pressurizer contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves and instrumentation. The electric heaters located in the lower section of the vessel regulate the Reactor Coolant System pressure by keeping the water and steam in the pressurizer at saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55 F/hr during startup of the reactor.

The pressurizer was designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer connects the pressurizer to the hot leg of a reactor coolant loop. During a positive surge, caused by a decrease in plant load, the pressurizer spray system, which is fed from the cold leg of a coolant loop, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. The spray valves on the pressurizer are power operated. In addition, the spray valves can be operated manually by a switch in the Control Room. Spray valve position is monitored by indicating lights in the Control Room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping. Isolation valves and bypasses are provided at each pressurizer spray valve. The outer bypass, which was provided to allow a small continuous spray flow to the pressurizer during on line maintenance of the valves, was removed by a subsequent modification.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of low alloy steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel.

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

Steam Generators

Each loop contains a Westinghouse Model 44F vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.2-5. Principal design parameters are listed in Table 4.1-4. The steam generators were designed and manufactured in accordance with Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code and were installed during the cycle 6/7 refueling outage as replacements for the original Model 44 steam generators.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Primary side manways are provided to permit access into the channel head and to the U-tubes.

Feedwater to the steam generator enters just above the top of the U-tubes through an elevated feedwater ring. The water exits the feedwater ring through inverted J-tubes installed on the top of the ring. This minimizes the potential for damaging water hammer events to occur. The water flows downward through an annulus between the tube wrapper

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and the shell and then upward through the tube bundle where it is converted to a steam-water mixture.

The steam-water mixture from the tube bundle passes through a low pressure drop, swirl vane modular primary separator assembly which imparts a centrifugal motion to the mixture and reduces the water content in the mixture. The separated water passes through a perforated section of riser tube, impinges on a small external downcomer sleeve, and combines with the feedwater for another pass through the tube bundle.

The remaining higher quality steam-water mixture rises through additional secondary separators which limit the moisture content of the steam to one tenth of one percent or less under all design load conditions. The steam outlet nozzle is equipped with an integral steam flow limiting device. Manways, handholes, and an inspection port are provided for maintenance and inspection of the secondary side.

The steam generator pressure boundary is constructed of low alloy steel. The heat transfer tubes are thermally treated Inconel 690. The interior surface of the channel head and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube-to-tubesheet joint is welded.

Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.2-6 and the principal design parameters for the pumps are listed in Table 4.1-5. Auxiliary system piping connections to the reactor coolant pumps are shown in Plant Drawing 9321-F-27383 [Formerly Figure 4.2-2A]. The reactor coolant pump estimated performance and net positive suction head characteristics are shown in Figure 4.2-7. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the 'knee' at about 45% design flow introduces no operational restrictions, since the pumps operate at full speed.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, a second seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water and vapor from the pump into the containment atmosphere.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the second seal, is also collected and removed from the pump. Pump seal injection flow is indicated in the Control Room as described in Section 9.2

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Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier cooling coil.

Each of the reactor coolant pumps is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of eleven pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers.

After the motor has come to a stop, one pawl engages the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. After the motor has come up to speed, the ratchet plate is returned to its original position by the spring return.

When the motor is started, the pawls drag over the ratchet plate until the motor reaches approximately 80 revolutions per minute. At this time, centrifugal force holds the pawls in the elevated position until the speed falls below the above value.

Normal loop flow, with all pumps running is approximately 9474 pounds/second (per loop). The following table indicates the loop flow distribution for less than four pumps running:

	Running Pump Loop Flow	Stopped Pump Loop Flow	Core Flow
3 Pumps Running*	10,244 (lb/sec)	-3,048 (lb/sec)	27,686 (lb/sec)
2 Pumps Running*	10,771 (lb/sec)	-1,944 (lb/sec)	17,655 (lb/sec)
1 Pump Running*	11,035 (lb/sec)	- 913 (lb/sec)	8,295 (lb/sec)

The above table assumes that the anti-reverse flow mechanism functions properly.

In the highly unlikely event that the anti-reverse flow device does not function properly and allows the pump to rotate freely in reverse, the following flow distributions would be realized:

	Running Pump Loop Flow	Stopped Pump Loop Flow	Core Flow
3 Pumps Running*	10,360 (lb/sec)	-4,837 (lb/sec)	26,244 (lb/sec)
2 Pumps Running*	10,856 (lb/sec)	-2,924 (lb/sec)	15,863 (lb/sec)
1 Pump Running*	11,067 (lb/sec)	-1,313 (lb/sec)	7,127 (lb/sec)

Each loop is provided with flow and temperature instruments. The loop temperature information is valid for all loops, active or inactive. The loop flow information is valid only for active loops. Inactive loop flow can be estimated by calculating total flow from a heat balance and then subtracting the active loop flow.

It is important to note that the Reactor Control and Protection Systems were designed to prevent abnormally induced transients or abnormal operating conditions. Plant operating instructions describe the various combinations of plant power level and reactor coolant flows which will result in plant trips.

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft.

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*NOTE: Reactor power operation is not permitted with less than four loops operating.

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An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests were conducted on less than full scale prototype seals as well as on full size seals. Operating experience with other large size, controlled leakage shaft seal pumps was also available to confirm the seal design.

The primary coolant pump flywheel is shown in Figure 4.2-8. The flywheel was fabricated from two rolled, vacuum-degassed, ASTM A-533 steel plates. The plates are bolted together with bolts aligned perpendicular to the plane of the plates. Thus the bolts carry no stress during operation.

The material specification is ASTM A-533 Grade B Class I, plus supplementary material testing requirements and Charpy tests, as detailed in Section 4.3.

Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. It is shown on Figure 4.2-9.

Steam and water discharge from the power relief and safety vales pass to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. The tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray and a drain to the Waste Disposal System which are operated to cool the tank following a discharge.

The tank size is based on the requirement to condense and cool a discharge equivalent to 110 percent of the full power pressurizer steam volume.

The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the Reactor Containment. The rupture discs on the relief tank have a combined relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture discs setting) is twice the calculated pressure resulting from the maximums safety valve discharge described above. This margin is to prevent deformation of the disc. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 percent of the set point pressure at full flow.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor coolant System which might collect in the pressurizer vessel. The tank is constructed of carbon steel with a corrosion resistant coating on the internal surface.

Piping

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawing in Chapter 1. Piping design data are presented in Table 4.1-6.

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The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 inch ID in the hot legs, 27 inches ID in the cold legs and 31 inches ID between each loop's steam generator outlet and its reactor coolant pump suction. The pressurizer relief line, which connects the pressurizer safety and relief valves' outlet to the inlet nozzle flange on the pressurizer relief tank was constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains and connections to other systems are austenitic stainless steel. All piping connections are welded except for flanged connections at the pressurizer relief tank and at the relief and safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

1) Return lines from the residual heat removal loop (safety injection lines)

2) Both ends of the pressurizer surge line,

3) Pressurizer spray line connection to the pressurizer, and

4) Charging lines and auxiliary charging line connections.

Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded.

Valves that perform a modulating function are equipped with either two sets of packing and an intermediate leakoff connection or have been designed with live-loaded packing, which will either control or migrate the potential for valve stem leakage due to modulating service.

Component Supports

The support structures for the reactor coolant system components are described in Appendix 4B.

4.2.3 Pressure-Relieving Devices

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. The relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Plant Drawing 9321-F-27473 [Formerly Figure 4.2-11] and the valve design parameters are given in Table 4.1-3.

A reactor head vent system is also provided to exhaust non-condensible gases or steam from the primary system that could inhibit natural circulation core cooling.

Two power operated relief valves (PORVs) and three code safety valves (SVs) are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. The PORVs also operate from the Overpressure Protection System described in Section 4.3 to prevent RCS pressure from exceeding the limits of Appendix G of Section III of the ASME Pressure Vessel and Boiler Code (1972 Summer Addenda) and ASME Code Case N-514.

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The instrument N_2 system for the PORVs is tapped from the N_2 supply line to the safeguards accumulators. The actual take-off point for this N_2 system is downstream of the pressure regulator valve NNE-863. Each PORV is supplied with its own accumulator. The PORV accumulators individually hold 6 cu ft of N_2 at a minimum pressure of 550 psig. During low temperature shutdown operations, the Overpressure Protection System requires an N_2 supply of sufficient capacity which, in case of loss of the main N_2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N_2 supply is provided by one Safety Injection Accumulator having its associated N_2 fill valve blocked open.

A platform within the pressurizer missile shield provides access to the pressurizer safety valves for testing purposes. This access platform does not interfere with maintenance which may require removal of the safety and/or power operated relief valves.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure valve by two rupture discs which discharge into the Reactor Containment. The rupture disc relief conditions are given in Table 4.1-3.

The design criteria require that transient flow loads from valve discharge be combined with deadweight loads, seismic loads, and loads due to internal pressure. Thrust, bending, and torsion (as applicable) from each of these loads was determined. Primary stress in the supports under this load combination was limited to 1.33 times allowable stress or rated load in accordance with Manufacturer's Standardization Society Standard MSS-SP-58 for standard supports or AISC-69 requirements for non-standard supports. In those instances where the integrity of supports is also dependent on reinforced concrete anchorage, the concrete behavior limits and anchor bolt behavior were in accordance with I.E. Bulletin 79-02. (8)

In response to NUREG-0737, Item II.D.1, the Electric Power Research Institute (EPRI), under contract with the Westinghouse Owners Group, performed full scale tests on typical safety and relief valves and associated piping configurations under simulated transient conditions.

EPRI issued the results of the test program to participating utilities along with evaluation guidelines for applying the test results to plant specific pressurizer safety and relief valves and associated piping. These guidelines have been used to evaluate the adequacy and integrity of the pressurizer safety and relief valve piping during plant transients which challenge the safety and relief valves.

The evaluation of the pressurizer safety and relief valve piping considered the following two transient cases:

Case 1: Sequential actuation of the power operated relief valves (PORV's) and safety valves (SV's) at their respective set pressures with pressurizer pressure increasing at a rate of 130 psi/sec and loop seal temperature at the inlet to the SV's of 260°F.

Case 2: Block valves upstream of the PORV's closed and only SV's open at their set pressures with pressurizer pressure increasing at a rate of 144 psi/sec and loop seal temperature at the inlets to the SV's of 260°F.

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The peak transient pressures, temperatures and flow rates in the discharge piping were computed using the RELAP-5 MOD 1, Cycle 14 computer code which has been verified during the EPRI test program to yield results which closely match actual pressures, temperatures and flow rates witnessed during the testing. A post processor code (FORCE) was used to translate the thermal hydraulic results of the RELAP code into transient forces on the piping and a structural code (STARDYNE) was used to determine the resulting pipe stresses and support transferred loads. In addition, the structural code NUPIPE-II was used to determine the pipe stresses and support transferred loads due to deadweight loads, normal pressure loads, thermal loads and seismic loads.

These evaluations indicated the need to upgrade the original design of the pressurizer safety and relief valve discharge system for purposes of complying with NUREG-0737 Item II.D.1. This upgrade was accomplished by a series of modifications which included: discharge piping support upgrade, discharge piping branch connection reinforcement, replacement of pressurizer safety valve internals, and re-routing of pressurizer safety valve loop seal drain lines to provide for continuous drainage back to the pressurizer. Continuous drainage is provided to preclude the occurrence of water hammer in the discharge piping in the event the valves open.

These modifications served to reduce the stresses in the safety and relief valve discharge piping to acceptable levels which are below the stress limits in the EPRI guidelines for the various load combinations considered. In addition, the stresses in the piping due to thermal loads are below the ANSI B31.1 1967 Power Piping Code stress limits. The constant supports were designed and fabricated in accordance with ANSI B31.1 whereas the hydraulic restraints were designed and fabricated in accordance with the ANSI B31.7 Nuclear Power Piping Code. Finally, the weld reinforcing pads were manufactured and installed in accordance with the requirements of ANSI B31.1.

Subsequent to the analyses and modifications described above, various pipe supports were further upgraded based on reanalyses of the pressurizer PORV inlet and outlet piping and supports. These reanalyses were performed to assess the effect to upgrading the PROV block valve operators with larger and heavier operators to enhance their design capabilities. Upgrade of the supports was deemed necessary to ensure that the integrity of the piping they support and the safety equipment they service will remain intact and to demonstrate that the design criteria used for Seismic Class I pipe supports is not violated with the added loading.

4.2.4 Protection Against Proliferation of Dynamic Effects

Engineered Safety Features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a Loss-of-Coolant Accident. Protection is provided by missile shielding and/or segregation of redundant components. This is discussed in detail in Chapter 6.

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into the Containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate.

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The concrete deck over the Reactor Coolant System also provides for shielding and missile damage protection.

Lateral bracing is provided near the steam generator upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing was provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the Reactor Coolant System is illustrated in the containment structure drawings which are presented in Chapter 5.

4.2.5 Materials of Construction

Each of the materials used in the Reactor Coolant System was selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

All reactor coolant system materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steels and Inconel, and they were chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The water in the secondary side of the steam generators is held within the appropriate chemistry specifications to control deposits and corrosion inside the steam generators. Secondary side chemistry is monitored as described in the Technical Specifications.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time.

It is characteristic of stress corrosion that combinations of alloy environment which result in cracking are usually quite specific. Environments which have been shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having high residual stresses resulting from manufacturing procedures.

The use of lead in the materials of the secondary side of Indian Point 3 was minimized to the practical limit of that occurring as trace elements in alloys and as such is insignificant.

All external insulation employed in the Reactor Coolant System is compatible with the structural materials of the component. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is insulated with low halide-content insulating material.

Assurance of adequate fracture toughness of the reactor vessel is provided by compliance with the requirements for fracture toughness included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it is not

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practicable to perform all tests in accordance with this Code, conservative estimates of material fracture toughness were made to demonstrate compliance with the Code requirements.

The techniques used to measure and predict the integrated fast neutron (E greater than 1 MeV) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum fast neutron exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the samples and vessel inner surface are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel is obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron exposure of the vessel inner surface was computed to be 0.922×10^{19} n/cm^2 at license expiration, or 27.1 effective full power years, at a power of 3216 MWt). Similarly, the maximum integrated fast neutron exposure at the ¼T location was computed to be 5.5×10^{18} n/cm^2 at EOL (11). With this exposure the end-of-life ΔRT_{NDT} was estimated to be 170.6°F, and RT_{PTS} temperature was not expected to be over 268°F, which is below the NRC screening criteria of 270°F. Each core redesign is evaluated to assure that leakage is less than assumed in analyses to predict the effect of neutron embrittlement.

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported. (1) Based on the Westinghouse evaluation, heatup and cooldown curves were developed for up to 9.26 EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, revision 2. "Radiation Embrittlement of Reactor Vessel Material", to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A of 10 CFR part 50, Appendix G. Capsules X and Z were analyzed (9)(10) and new pressure-temperature curves were developed using this methodology.

The maximum shift in RT_{NDT} after 20 EFPYs of operation is projected to be 230.1°F at the 1/4T and 188.8°F at the 3/4T vessel wall locations for Plate B2803-3 the controlling plate. Plate B2803-3 was also the controlling plate for the operating periods of 2 EFPYs, 9 EFPYs, 11.00 EFPYs, 13.3 EFPYs and 16.2 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value or RT_{NDT} at the end of 20 EFPYs. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table 4.4.1.

Changes in fracture toughness of the core region plates or forgings, weldments, and associated heat treated zones due to radiation damage will continue to be monitored by a surveillance program which conforms with ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. (3) Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. To this extent, the second, third and forth reactor vessel material surveillance capsules were

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removed in 1982, 1987 and 2003 respectively. These capsules have been tested and the results have been evaluated and reported. (4)(5)(6)(9)(10) No change to the technical specification heatup and cooldown limit curves was required as a result of the Capsule Y and Capsule Z analyses. However, the new analytical methodology introduced in Reg. Guide 1.99, Rev. 2, resulted in the generation of new heatup-cooldown curves for 9, 11, 13.3, 16.2 and 20 EFPY.

The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, drop weight and tensile specimens and post-irradiation testing by Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel.

Information from the radiation surveillance program described above is also used for evaluations required by the NRC's Pressurized Thermal Shock (PTS) Rule (10 CFR 50.61). Using the prescribed PTS Rule methodology, RTPTS values were generated for allbeltline region materials of the reactor vessel as a function of several fluence values and pertinent vessel lifetimes. All of the RTPTS values remain below the NRC screening values for PTS using the projected fluence exposure through the expiration date of the operating license. (7)

4.2.6 Maximum Heating and Cooling Rates

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.5. The normal system heat-up rate is $\leq 60°F/hr$, and cooling rate is $\leq 50°F/hr$. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurized liquid homogeneous with the coolant.

The fastest cooldown rates which result form the hypothetical case of a break of a main stream line are discussed in Chapter 14. Refer to Section 4.4 for further information.

4.2.7 Leakage

The existence of leakage from the Reactor Coolant System to the Containment, regardless of the source of leakage, is detected by one or more of the following conditions:

1) Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

2) A third instrument used in leak detection is the humidity detector. This provides a means of measuring overall leakage from all water and steam systems within the containment but furnishes a less sensitive measure. The humidity monitoring method provides backup to the radiation monitoring methods.

3) A leakage detection system is included which determines leakage losses from all water and steam systems within the Containment including that from the Reactor Coolant System. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the fan cooling units. It relies on the principle that all leakages up to sizes permissible with continued plant

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operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary.

4) An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer. Assuming no operator action, the increase in containment sump level provides a less sensitive means of detecting leakage. However, sensitivity of the processing systems and containment sump system can be improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours).

Leakage detection methods are described in detail and evaluated in Chapter 6.

Leakage Prevention

Reactor Coolant System components were manufactured to exacting specifications which exceeded normal code requirements (as outlined in Section 4.1.3). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it was subjected (as outlined in Section 4.5), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the Reactor Coolant System is permitted by the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device was selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. Leakage from the reactor through its head flange will leak off between the double o-ring seal and actuate an alarm in the Control Room.

Whereas leakage prevention is important from the standpoint of RCS inventory loss, it can also be of concern with regard to corrosion of carbon steel components within the reactor coolant pressure boundary. Leakage past the canopy seals of the reactor vessel closure head penetrations has been reported at several plants including Indian Point 3. While this type of leakage is not a safety issue due to the minimal amounts possible, it does represent a boric acid corrosion concern. For these reasons, CRDM and CET Seal Clamp Assemblies designed by the reactor manufacturer have been installed at the twelve spare reactor vessel head penetrations and at the five core exit thermocouple penetrations to prevent canopy seal weld leaks in these locations. The CRDM and CET Seal Clamp Assemblies meet all of the ASME Boiler and Pressure Vessel Code requirements applicable to the Indian Point 3 reactor vessel.

Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location which can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the

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leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and deposited by the evaporation process.

4.2.8 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces.

Following a significant boration or dilution, the operator normally will sample both the Reactor Coolant System hot legs and the pressurizer liquid for record purposes and to check that homogenization of the pressurizer liquid with the recirculating reactor coolant has been completed. For a cold shutdown, the operator borates the system prior to the start of cooldown.

All materials exposed to reactor coolant are corrosion resistant. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water qualify listed in Table 4.2-2. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and the Sampling System which are described in Chapter 9.

4.2.9 Reactor Coolant Flow and Temperature Measurements

Elbow taps are used in the primary coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation $\Delta P/\Delta P_o = (w/w_o)^{2.0}$ where ΔP_o is the referenced pressure differential with the corresponding referenced flow rate w_o and ΔP is the pressure differential with the corresponding flow rate w. The full flow reference point was established during initial plant startup. The low flow trip point was then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse plants. The expected absolute accuracy of the channel is within 10% and field results have shown the repeatability of the trip point to be within 1%. The analysis of the loss of flow transient presented in Section 14.1.6 provides for an allowance of 3% measurement inaccuracy.

As a result of the calibration techniques used, the absolute accuracy of the coolant flow measurement is not relevant. As indicated in Chapter 14, the limiting trip point assumed for analysis was 87% loop flow. As indicated above, this represents a 3% of flow allowance below the allowable value (low flow trip point) of 90% or higher which is dictated by the Technical Specifications. Since the trip point is calibrated as a function of full flow output of the instrument and since the flow rate of the reactor is verified to be equal or greater than the design flow rate during startup testing, the actual trip point would be 89% or greater based on the 1% repeatability.

Startup tests provide a means for verifying that reactor coolant flow is equal to or greater than the design flow rate. The core flow rate cap be verified with an accuracy better than 10% by correlating a secondary system heat balance and the inlet and outlet core temperatures. In addition measurements of pump input power and loop ΔP can be made at

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hot shutdown condition for various configurations of running pumps, and the absolute flow rate of each pump is verified to be greater than the design flow.

Hot leg temperature measurement of each loop is accomplished with three fast response, narrow range single element RTD's mounted in the three scoops of each hot leg. Cold leg temperature measurement of each loop is accomplished with one fast response, narrow range duel element RTD mounted in the manifold of each cold leg. In addition a single wide range RTD is provided in each cold leg at the discharge of the RCP and near the entrance to each steam generator.

4.2.10 Reactor Coolant System Vent Collection System

A vent collection system consisting of stainless steel pipes is provided as a means to collect water and contain gases during venting of the Reactor Coolant System during cold shutdown.

The system is located inside the Containment and consists of a vent collecting header which is connected to the Reactor Coolant System venting points. The vent header discharges gases inside the Containment in the area of the purge exhaust system. Liquids are continuously drained to the Reactor Coolant Drain Tank or to the Containment Sump.

The Vent Collection System is used only when the reactor is in cold shutdown; vent connections are disconnected and the Reactor Coolant System venting points flanged off before the reactor is put into hot shutdown mode.

4.2.11 Reactor Head Vent System

The basic function of the Reactor Vessel Head Vent System is to remove non-condensible gases or steam from the Reactor Vessel Head. This system is designed to mitigate a possible condition of inadequate core cooling due to a loss of natural circulation resulting from the accumulation of non-condensible gases in the Reactor Coolant System. The Reactor Head Vent System connects to the reactor head via an existing part length control rod drive mechanism conoseal port. The vent piping downstream of the concoseal port connects to two parallel paths. Each path contains two series normally closed valves. The two flow paths join together and discharge into the pressurizer safety and relief valve discharge header. The common vent line downstream of the valves contains a flanged spool piece which is removable for refueling. These flanged connections are therefore outside the reactor coolant pressure boundary.

The Reactor Head Vent System is operated manually from the main control room. The solenoid operated isolation valves are full open/close type valves which are controlled from individual control switches in the main control room.

4.2.12 Reactor Vessel Level Indication System (RVLIS)

The basic function of the RVLIS is to monitor the water level in the reactor vessel or relative voids in the RCS during accident conditions. However, the system is designed to function under all plant normal, abnormal, and accident conditions, including LOCA, post-LOCA, and during and after a seismic event. Refer to Section 7.5.2 for more detailed information about the operation of this system.

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(7) Perone, V. A. et al. "Indian Point Unit 3 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-11045, January, 1986 and Revision 1 thereto dated June,1989.

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(10) Entergy Report to NRC "Capsule X Material Surveillance Report, NL-04-092," 07/29/04.

(11) Calculation CN-RCDA-03-88, Rev. 0, "Indian Point Unit 3 Stretch Power Uprate Evaluation for Reactor Vessel Intergrity," Westinghouse Electric Co., December 2003.

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TABLE 4.2-1

MATERIALS OF CONSTRUCTION OF THE
REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-302, Gr. B
	Shell & Nozzle Forgings	A-508 Class 2
	Cladding, Stainless Weld Rod	type 304 equivalent
	Thermal Shield and Internals	A-240, type 304
	Insulation	Stainless steel, Aluminum
Replacement Steam Generator	Pressure Plate	SA-533, Gr. B Class 1
	Cladding, Stainless Weld Rod	type 304 equivalent
	Cladding for Tube Sheets	Inconel Weld Deposit
	Tubes	Inconel (SB-163)
	Channel Head Forgings	SA-508 Class 3
Pressurizer	Shell	SA-302 Gr. B
	Heads	SA-216 WCC
	External Plate	SA-302, Gr. B
	Cladding, Stainless	type 304 equivalent
	Internal Plate	SA-240 type 304
	Internal Piping	SA-376 type 316
Pressurizer Relief Tank	Shell	A-285 Gr. G
	Heads	A-285 Gr. C
	Internal surface coating	Amercoat 55
Piping	Pipes	A-376 type 316
	Fittings	A-351, CF8M
	Nozzles	A-182 F316
Pump	Shaft	type 304
	Impeller	A-351, CF8
	Casing	A-351, CF8M
Valves	Pressure Containing Parts	A-351, CF8M and A-182 F316

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TABLE 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Chemistry Parameter	MODE	Requirement
Electrical Conductivity	1, 2, 3, 4, 5	Determined by the concentration of boric acid and alkali present.
Solution pH (at 25 C)	1, 2, 3, 4, 5	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration).
Oxygen, ppm, max. (>250 F)	1, 2, 3, 4	0.10
Chloride, ppm, max. (>250 F)	1, 2, 3, 4	0.15
Flouride, ppm, max. (>250 F)	1, 2, 3, 4	0.15
Hydrogen, cc (STP)/kg H_20	1, 2	25-50
Total Suspended Solids, ppm, max.	1, 2, 3, 4, 5	1.0
pH Control Agent (Li^7OH)	1, 2	Varying to maintain minimum RCS $pH_{(t)}$ of 6.9, not to exceed 3.5 ppm Lithium
Boric Acid as ppm Boron	1, 2, 3, 4, 5, 6	Variable from 0 to 4000

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4.3 SYSTEM DESIGN EVALUATION

4.3.1. Safety Factors

The safety of the reactor vessel and all other Reactor Coolant System pressure containing components and piping is dependent on several major factors including the design and stress analyses, the material selection and fabrication, and the quality control and operations control.

Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Boiler and Pressure Vessel Code. The evaluation demonstrates that stress levels are within the stress limits of the Code. Table 4.3-1 presents a summary of the results of the stress evaluation.

The State of New York has adopted ASME Code Section III and imposes no additional design requirements beyond those listed in this code.

A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2.

For the Indian Point 3 reactor vessel, the maximum thermal stress due to gamma ray heating occurs in the cylindrical portion of the vessel adjacent to the core and its value is about 2500 psi. This additional thermal stress does not augment the stress intensity values considerably. The maximum stress intensity values under steady state and transient operating conditions are still far below the allowable limits of N-414 of ASME Boiler and Pressure Vessel Code Section III. The effect of gamma ray heating on the cumulative usage factor is negligible.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants presently in service. These cycles include five heatup and cooldown cycles per year, a conservative selection when the vessel may not complete more than one cycle per year during normal operation. A complete list of thermal and loading cycles and their frequencies used for design purposes are given in Table 4.1-8.

The vessel design pressure is 2485 psig while the normal operating pressure will be 2235 psig. The resulting operating membrane stress is, therefore, amply below the code allowable membrane stress to account for operating pressure transients.

Assurance of adequate fracture toughness of the Reactor Coolant System is provided by compliance, insofar as possible, with the requirements for fracture toughness included in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

The heatup and cooldown curves for the plant are based on the actual measured fracture toughness properties of the vessel materials (see the Technical Specifications) determined in accordance with the above mentioned fracture toughness requirements. Where sufficient tests to comply with these requirements for fracture toughness testing were not performed, conservative estimates of fracture toughness properties are used. Maximum allowable pressures as a function of the rate of temperature change and the actual temperature

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relative to the vessel RT_{NDT} are established according to the methods given in Appendix G, Protection Against Nonductile Failure, published in Section III of the ASME Boiler and Pressure Vessel Code. Curves incorporating allowances for instrument error in measurement of temperature and pressure are given in the Technical Specifications.

These curves are based on the predicted RT_{NDT} of the vessel and include appropriate estimates of ΔRT_{NDT} caused by radiation. Estimated ΔRT_{NDT} values are derived by using Figure 4.4-1 and the fluence at 1/4T corresponding to the maximum for the service period applicable. (See Figure 4.2-10)

The results of the radiation surveillance program are used to verify the ΔRT_{NDT} predicted from Figure 4.4-1 as discussed in Section 4.4.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material automatically provides additional conservatism for the non-irradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

Change in fracture toughness of the core region plate or forging, weld metal and of the associated heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. For additional details of the irradiation program, see Section 4.5. In addition, assurance of adequate fracture toughness is further provided by the evaluations required by the NRC's Pressurized Thermal Shock (PTS) Rule (10CDF50.61). Information from the radiation surveillance program referenced above is used to demonstrate that the RT_{PTS} values generated in accordance with PTS rule methodology remain below the NRC screening values for PTS.

The vessel closure contains fifty-four 7-inch studs. The stud material has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 54,210 psi. This means that twenty eight of the fifty four studs have the capability of withstanding the hydrostatic end load on the vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

As part of the plant operator training program, supervisory and operating personnel are instructed in reactor vessel design, fabrication and testing, as well as present and future precautions necessary for pressure testing and operating modes. The need for record keeping is stressed as such records are helpful for future summation of time at power levels and temperatures which tend to influence the irradiated properties of the material in the core region.

The following components of the reactor pressure vessel were analyzed in detail through systematic analytical procedures:

Control Rod Housing

An interaction analysis was performed on the CRDM housings. The flange was assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes were taken into account in the analysis. The

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local flexibility was considered at appropriate locations. The closure head was treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head were assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the "J" weld.

Closure Head Flange and Shell

The closure head, closure head flange, vessel flange, vessel shell and closure studs were all evaluated in the same analysis. An analytical model was developed by dividing the actual structure into different elements such as long sphere, ring, long cylinder, cantilever beam, etc. An interaction analysis was performed to determine the stresses due to mechanical and thermal loads. These stresses were evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code Section III.

Main Closure Studs

A similar analysis to the one described for the closure head flange and shell was performed for the vessel flange to vessel shell juncture and for the main closure studs.

Inlet Nozzle and Vessel Support

For the analysis of nozzle and nozzle to shell juncture, the loads considered were internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel, earthquake loading, expansion and contraction, etc. A combination of methods was used to evaluate the stresses due to mechanical and thermal loads and external loads resulting from seismic pipe reactions, earthquake, pipe break, etc.

For fatigue evaluation, peak stresses resulting from external loads and thermal transients were determined by concentrating the stresses as calculated by the above described methods. Combining these stresses enabled the fatigue evaluation to be performed.

Outlet Nozzle and Vessel Support

The method of analysis for the outlet nozzle and vessel support was the same as for the inlet nozzle.

Vessel Wall Transition

Vessel wall transition was analyzed by means of a standard interaction analysis. The thermal stresses were determined by the skin stress method where it was assumed that the inside surface of the vessel was at the same temperature as the reactor coolant and that the mean temperature of the shell remained at the steady state temperature. This method was considered conservative.

Core Barrel Support Pads

Thermal, mechanical and pressure stresses were calculated at various locations on the pad and at the vessel wall. Mechanical stresses were calculated by the flexure formula for bending stress in a beam, pressure stresses were taken from the analysis of the vessel to bottom head juncture and thermal stresses were determined by the conservative method of

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skin stresses. The stresses due to the cyclic loads were multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

Bottom Head to Shell Juncture

The standard interaction analysis and skin methods were employed to evaluate the stresses due to mechanical and thermal stresses, respectively. The fatigue evaluation was made on a cumulative basis where superposition of all transients was taken into consideration.

Bottom Head to Instrument Penetrations

An interaction analysis was performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head were assumed to be local only. It was also assumed that, for any condition where there was interference between the tube and the head, no bending at the weld could exist. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the "J" weld.

The location and geometry of the areas of discontinuity and/or stress concentration are shown in Figures 4.3.-1, 4.3.-2 and 4.3-3.

In addition to the analyses of reactor vessel components, the following transients were analyzed since they cause temperature and pressure excursions influencing the cumulative fatigue of the reactor vessel:

Transient Analyses

Loss of Load Transient. This is the most severe anticipated transient on the Reactor Coolant System. This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip. The reactor and turbine eventually are assumed to trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System.

Figure 4.3-4 gives the pressure-temperature transient assumed in the analysis for usage factor. This design basis transient is more severe than that reported in Section 14.1.8.

Loss of Flow Transient. This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop. Figure 4.3-5 gives the temperature transient assumed in the analysis for usage factor. This design basis transient is more severe than that reported in Section 14.1.6.

The number of occurrences of the loss of load and the loss of flow transients was generally specified at two (2) for each year of plant design life.

Reactor Coolant System Stress Analyses

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All components in the Reactor Coolant System were designed to withstand the effects of these and other transients that would result in system temperature and pressure changes.

Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature were determined for each of these transients through systematic analytical procedures. These stress intensity values $S_{ij}(i, j = 1, 2, 3)$ were plotted against a time interval for each cycle. This plot would represent one or more stress cycles. For each cycle, extreme values of S_{max} and S_{min} were determined. From these values, the largest S_{alt} (alternating stress intensity) was found.

For this value of S_{alt} , an allowable number of cycles (N) was determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (n) gave the usage factor (u). The usage factor was determined in this manner for all transients. The cumulative usage factor was determined by summing the individual usage factors. The cumulative usage factor ($U=u_1 + u_2 + u_3....$) was never allowed to exceed a value of 1.0.

Subsection N415.2 of the 1965 Edition of the ASME Code was used for calculating the usage factors.

Details of thermal and seismic analyses are summarized below[15]. These analyses are directly applicable to Indian Point 3.

Thermal Stresses

Maximum thermal stresses in the core barrel would occur if cold water were injected from the accumulators due to the occurrence of a Loss-of-Coolant Accident. The barrel is exposed to cold water in the downcomer annulus and to somewhat hotter water in the compartments between barrel and baffle, producing a thermal gradient across the barrel wall.

The lower support structure is cooled more uniformly because of the large and numerous flow holes, and consequently, thermal stresses are lower.

The method used to obtain the maximum barrel stresses was as follows:

1) Temperature distribution across the barrel wall was computed as a function of time taking into consideration water temperatures and film coefficients

2) Assuming that the obtained thermal gradients were axisymetrically distributed, which is conservative for stresses, maximum thermal stresses were computed in the barrel which was considered as an infinite cylinder

3) Thermal stresses were added to primary stresses, including seismic, in order to obtain the maximum stress state of the barrel.

Results of the analyses showed that the maximum thermal stresses in the barrel wall were well below the allowable criteria given for design by Section III of the ASME Code.

Seismic Analysis of Reactor Internals

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The maximum stresses were obtained by combining the contributions from the horizontal and vertical earthquakes in the most conservative manner. The following paragraphs describe the horizontal and vertical contributions.

Horizontal Earthquake Model and Procedure

The Containment Building along with the reactor vessel support, the reactor vessel, and the reactor internals were included in this analysis. The mathematical model of the building, attached to ground, was identical to that used to evaluate the building structure. The reactor internals were mathematically modeled as beams, concentrated masses, and linear springs.

All masses, water, and metal were included in the mathematical model. All beam elements had the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components were attached somewhat uniformly, their mass was included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements was included as a distributed mass.

Horizontal components were considered as concentrated masses acting on the barrel. These concentrated masses also included components attached to the horizontal members since these were the media through which the reaction was transmitted. The water near and about these separated components was considered as being an additive at these concentrated mass points.

The concentrated masses attached to the barrel represented the following: a) the upper core support structure, including the upper vessel head and one-half the upper internals; b) the upper core plate, including one-half the thermal shield and the other half of the upper internals: c) the lower core plate, including one-half of the lower core support columns; d) the lower one-half of the thermal shield; and e) the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity was chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional areas were selected along with a value for Poisson's ratio. The fuel assembly moment of inertia was derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provided stiffness values for use in this analysis.

The fuel assemblies were assumed to act together and were represented by a single beam. The following assumptions were made with regard to connection restraints. The vessel is pinned to the vessel support which is the surrounding concrete structure and part of the Containment Building. The barrel is clamped to the vessel at the barrel flange and spring connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

Model analysis plus the response spectrum method[16] were used in this analysis. The modal analysis was studied by the use of a transfer matrix method.

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The maximum deflection, acceleration, etc., were determined at each particular point by summing the absolute values obtained for modes. With the shear forces and bending moments determined, the earthquake stresses were then calculated.

Figure 4.3-6 shows the mathematical model studied.

Analytical Model for Vertical Earthquake Model and Procedure

The reactor internals were modeled as a single degree of freedom system for vertical earthquake analysis. The maximum acceleration at the vessel support was increased by the amplification due to the building-soil interaction.

There were no interfaces in the analysis (e.g., dynamic to static, elastic to plastic).

Reactor Coolant Pumps Evaluation

Pump Motor Overspeed

During normal operation, the reactor coolant pumps are supplied from the unit auxiliary bus and therefore are tied to the turbine generator frequency (speed). On occurrence of unit (turbine) trip, the pump electrical buses are transferred from the auxiliary transformer without any intentional delay.

On most electrical and mechanical events which cause the turbine to be tripped, the reactor coolant pump buses and the unit are tripped simultaneously and the pumps will therefore not exceed their normal or pretrip running speed. If for some unlikely reason the only plant trip is a turbine overspeed trip (mechanical – hydraulic trip), then the pump trip will be initiated by the turbine hydraulic system and the trip point will be between 106 and 110 percent of the turbine generator synchronous speed. The turbine overspeed trip point is set at a value not to exceed 106 percent of synchronous speed (1908 rpm).

Missile Prevention

The reactor coolant pump motor bearings are of conventional design, the radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated – the lower radial bearing and thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner. Low oil levels would signal an alarm in the Control Room and require shutting down the pump. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, would be indicated and alarmed in the Control Room as a high bearing temperature. This, again, would require pump shutdown. Even if these indications were ignored and the bearing proceeded to failure, the low melting point Babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing would occur. In this event, the motor would continue to drive, as it has sufficient reserve capacity to operate even under such conditions. However, it would demand excessive currents and, at some stage, would be shut down because of high current demand.

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the

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motor. This would constitutes a loss of coolant flow in the one loop, the effect of which is analyzed in Chapter 14.

Following the seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar into the bearing. Any seizure in the seals would result in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indication of pump malfunction under these conditions would be initially by high temperature signals from the bearing water temperature detector and later by excessive No. 1 seal leakoff indications. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble. The pump would then be shut down for investigation.

The design specifications for the reactor coolant pumps included as a design condition the stresses generated by a design basis earthquake ground acceleration of 0.15g. Besides examining the externally produced loads from the nozzles and support lugs, an analysis was made of the effect of gyroscopic reaction on the flywheel and bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during the maximum seismic disturbance.

The pump would continue to run, unaffected by such conditions. In no case would any bearing stress in the pump or motor exceed or even approach a value which the bearing could not carry.

Precautionary measures, taken to preclude missile formation from primary coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

Reactor Coolant Pump Flywheel

a) Methods of Fabrication

The pump flywheels were fabricated from vacuum degassed steel accordance with ASTM A-533 CL-1, Grade B.

The flywheel blanks were flame-cut from 8 inch and 5 inch plates with allowance for exclusion of flame-affected metal. They were then machined to the specific dimensions, and the bolt holes were drilled. These plates were subjected to 100 percent volumetric ultrasonic examination.

The two plates, machined to 7.25 inches and 4.45 inches in thickness, were then bolted together, the finished flywheel was attached to the motor shaft,

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and the whole unit was balanced to yield vibration levels at operating speed less than 0.001 inches double amplitude.

The supplier certification data, presented in Table 4.3-5, shows the Charpy V-notch test results for the Indian Point 3 flywheels at +10° F.

Acceptability of the above flywheel material for Indian Point 3, in comparison to Safety Guide No. 14 toughness criteria, was determined by the following two steps:

1) A reference curve describing the lower bound fracture toughness behavior for the material in question was established

2) Charpy (C_V) impact energy values obtained in certification tests (Table 4.3-5) at 10° F were used to fix the position of the heat in question on the reference curve.

A lower bound K_{Id} reference curve (see Figure 4.3-8) was constructed from dynamic fracture toughness data generated by Westinghouse on A-533 Grade B, Class I steel. [13][14] All data points were plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound, below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static represents a large degree of conservatism.

The applicability of a 30 ft-lb Charpy energy reference value was derived from sections on Special Mechanical Property Requirements and Tests in Article 3, Section III of the ASME Boiler and Pressure Vessel Code. The implication was that the test temperature lies a safe margin above NDT. Indian Point 3 flywheel plates exhibited an average value greater than 30 ft-lbs in the weak direction and, therefore, met the specific requirement "a" stated in Safety Guide No. 14 that NDT must be no higher than 10° F. Making the conservative assumption that all materials in compliance with the Code requirements were characterized by an NDT temperature of 10° F, one was able to reassign the "zero" reference temperature position in Figure 4.3-8 a value of 10° F.

Flywheel operating temperature at the surface is 120° F. The lower bound toughness curve indicated a value of 116 ksi-in$^{1/2}$ at the (NDT + 110) position corresponding to operating temperature. Safety Guide No. 14 requirement "c" was fulfilled with considerable margin for safety. The flywheel analysis was reviewed by Westinghouse (Reference 20) and it was concluded that a 10°F increase in flywheel surface temperature to 130° F has no adverse effect on the flywheel integrity or the conclusions of the analysis.

By assuming a minimum toughness at operating temperature in excess of 100 ksi-in$^{1/2}$, it was seen by examination of the correlation in Figure 4.3-9 that the C_V upper shelf energy must be in excess of 50 fl-lb; therefore, the requirement "b" that the upper shelf energy must be at least 50 ft-lb was satisfied.

Based on the above discussion, the flywheel materials met the Safety Guide No. 14 toughness criteria on the basis of supplier certification data.

The design requirements of the bearings were primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. To this end, the surface bearing stresses were held at a very low value, and even

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under the most severe seismic transients or other accidents, would not begin to approach loads which cannot be adequately carried for short periods of time.

Because there were no established criteria for short-time stress-related failures in such bearings, it was not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gave assurance of the adequacy of the bearings to operate without failure.

As was generally the case with machines of this size, the shaft dimensions were predicated on avoidance of shaft critical speed conditions rather than actual levels of stress.

There are many machines as large as and larger than the reactor coolant pumps designed to run at speeds in excess of first shaft critical. However, it was considered desirable in a superior product to operate below first critical speed, and the Reactor Coolant Pumps were designed in accordance with this philosophy. This resulted in a shaft design which, even under the severest postulated transient, gave very low values of actual stress. While it would be possible to present quantitative data of imposed operational stress relative to maximum tolerable levels, if the mode of postulated failure were clearly defined, such figures would have little significance in a meaningful assessment of the adequacy of the shaft to maintain its integrity under operational transients. However, a qualitative assessment of such factors gave assurance of the conservative stress levels experienced during these transients.

So, in each of these cases, where the functional requirements of the component controlled its dimensions, it was seen that if the requirements discussed previously were met, the stress-related failure cases were more than adequately satisfied.

It was thus considered to be out of the bounds of reasonable credibility that any bearing or shaft failure could occur that would endanger the integrity of the pump flywheel.

b) Methods of Quality Assurance

An NDTT less than +10 F was specified. A minimum of three Charpy tests were made from each plate, parallel and normal to the rolling direction, to determine that each blank satisfied design requirements.

The finished flywheels were subjected to 100% volumetric ultrasonic inspection.

The finished machined bores were also subjected to magnetic particle, or liquid penetrant examination.

The design-fabrication techniques yielded flywheels with primary stress at operating speed (as shown in Figure 4.3-7) which was less than 50% of the minimum specified material yield strength at room temperature (100 to 150 F).

The flywheel calculated stresses at operating speed were based on stresses due to centrifugal forces (Figure 4.3-7). The stress resulting from the interference fit of the flywheel on the shaft was less than 2000 psi at zero speed, but this stress became zero at approximately 600 rpm because of radial expansion of the hub.

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A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

1) Maximum tangential stress at an assumed overspeed of 125% compared with maximum expected overspeed of 109%

2) A crack through the thickness of the flywheel at the bore

3) 400 cycles of startup operation in 40 years of plant design life.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030" to 0.060" per 1000 cycles.

An ultrasonic inspection capable of detecting at least 1/2" deep cracks from the ends of the flywheel and a dye penetrant or magnetic particle test of the bore will be performed when the RCP motor is sent out for refurbishment in accordance with the Preventive Maintenance Program.

c) Normal Operating Speed

The primary coolant pumps run at 1189 rpm and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of outside load.

For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. For the overspeed condition, which would not persist for more than 30 seconds, pump operating temperatures would remain at about the design value.

d) Bursting Speed

Bursting speed of the flywheels has been calculated on the basis of Robinson's results[1] to be 3900 rpm, more than three times the operating speed. This is confirmed using Griffith-Irwin theory.[2]

Steam Generators

The pressure boundary integrity of the steam generator, including the primary to secondary pressure boundary is assured by compliance of the design, fabrication, analysis, inspection, and testing activities with the criteria and requirements of the ASME Boiler and Pressure Vessel Code. The stress report for the Model 44F steam generators currently installed in Indian Point Unit 3 included an evaluation for faulted conditions including large break LOCA and steam line break (loss of secondary pressure). The stress intensities calculated for these conditions are less than the applicable limits from the ASME Code. The criteria and requirements of Section III of the ASME Code (1965 edition, Winter 1965 Addenda) were used for the evaluation.

The evaluation of the stress intensity levels in the tubesheet and channel head for these faulted conditions is based on an evaluation of interactions of the complex structure of the channel head, tubesheet and lower shell. A finite element computer program is used for the evaluation. The structure is modeled in terms of discrete elements with loading and boundary conditions applied to these elements. The system of simultaneous linear

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equations resulting from the modeling is solved to determine the stress conditions. This method of stress analysis is well established for reactor coolant system components. For the tubesheet calculations, the guidelines of Article I-9, ASME Code, Section III were used to calculate the ligament efficiency based on nominal pitch dimension and maximum hold dimensions.

The postulated rupture of a primary pipe is assumed to impose a maximum pressure differential of 1100 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 21,620 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition. The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The postulated rupture of a secondary pipe is assumed to impose a maximum pressure differential of 2485 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 54,650 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition (Tables 4.3-3 and 4.3-4). The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The stress intensity in the tubes has also been considered for the postulated faulted conditions. In addition to the pressure differential stresses, LOCA blowdown forces result in a bending load on the tubes. The fatigue due to vibration of the tubes during a steam line break does not need to be considered in the evaluation. The requirements of the ASME Code are met for these faulted conditions.

In addition to analytical results, results of destructive pressure testing or representative tubes demonstrates a factor of safety against tube collapse due to external pressure. The results of the pressure testing have been used to calculate a collapse pressure for tubes of the size and material in the Model 44F steam generators. The lower bound collapse pressure for the tubes in the Indian Point steam generators is 2369 psi considering tube ovality, tube wear, and tube corrosion.

The structural evaluation of the tubes uses an allowance of 2 mils of uniform wall thinning. This value is based on published values and operating experience for corrosion and erosion-corrosion. The plugging limit for indication of tube degradation is based on the requirements of IWB-3521 of the ASME Boiler and Pressure Vessel Code Section XI or an analysis meeting the requirements of Regulatory Guide 1.121 and not on the allowance for uniform thinning in the structural evaluation.

The loading conditions considered include the maximum potential earthquake loading conditions superimposed on the loss of secondary pressure effects. The dynamic effects of the fluid and the acceleration of the steam generator result in a small increase in the equivalent pressure loading compared to the base pressure differential. The stress intensity for the combined loading condition is well below limits.

The fluid dynamic load on the tube support plate for the steam line break conditions has been considered. The analysis has determined that the tube supports will be restrained without deformation of the tubes.

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In addition, the secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70 °F.

4.3.2 Reliance on Interconnected Systems

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Power Conversion, the Safety Injection and the Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators and upon the steam, feedwater, and condensate system for decay heat removal under normal operating conditions until RHR can be placed in service (reactor coolant temperature between 250° F and 350° F). The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of all main reactor coolant pumps.

The flow diagram of the Steam and Power Conversion System is shown on Figure 10.2-1. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The Axiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative.

The Safety Injection System is described in Section 6.2. The Residual Heat Removal System is described in Section 9.3.

4.3.3 System Integrity

A complete stress analysis, which reflected consideration of fall design loadings detailed in the design specification, was prepared by the manufacturer. The analysis showed that the reactor vessel, steam generators, reactor coolant pump casings and pressurizer complied with the stress limits of Section III of the ASME Code. A similar analysis of the piping showed that it complied with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness tests were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generators and pressurizer to provide assurance that hydrotesting and operation will be in the ductile region at all times. In addition, dropweight tests were performed on the reactor vessel plate material.

As an assurance of system integrity, all components in the system were hydrotested at 3110 psig prior to initial operation. Replacement steam generators were shop tested on the primary side at 3107 psig, and hydrotested after installation in accordance with ASME Section XI requirements.

A summary of Charpy V-notch and dropweight test results for the reactor vessel plates and forgings is given in Section 4.4.

Furnace Sensitized Components

The following pressure or strength-bearing stainless steel component parts in the Reactor Coolant System have become furnace-sensitized during the fabrication sequence:

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Reactor Vessel

Eight primary nozzle safe ends (forgings) which were overlaid in the field with stainless steel weld metal

Steam Generators

Two primary nozzle safe ends per generator – weld metal buttering

Pressurizer

All nozzle safe ends (forgings) in top and bottom heads.

Vibration and Cycle Loads

Vibration loads were considered in the design for the reactor internals, the steam generator tube bundles, and the reactor coolant pipe. Reactor coolant pump vibration is insignificant. Instrumentation is provided to check the vibration level of these pumps if an abnormal condition is suspected.

Reactor Internals

Model tests of the Indian Point 3 reactor internals were performed for normal operating and transient conditions. Results of the combined analytic and experimental work were factored into the design.

Predicted stresses and deflections were in agreement with tests on reactors having similar internals design. The results of the vibration tests performed on the Ginna rector (reported in WCAP-7408-L, Westinghouse proprietary report) [18] confirmed that the tests agree very closely with the predicted performance and margins. A more extensive testing program was performed during pre-operational testing for Indian Point 2.

Allowable stress amplitude for flow induced vibration was established on the basis of the material fatigue properties (endurance limit of 20,000 psi for 10^{10} cycles). Since infinite cycle fatigue was a criterion, no limits were then necessary for frequency. Displacement amplitudes for reactor internals vibration were not governing; stress limits were more restrictive.

An analysis of the dynamic response of the Indian Point 2 internals under seismic and blowdown loads was made. Allowable criteria were established and stresses and deflections were determined to assure that seismic and blowdown loads will not prevent core shutdown or will not interfere with the effectiveness of the emergency core cooling system (reported in detail in WCAP-7822, Westinghouse non-proprietary report).[15] This analysis applies directly to Indian Point 3.

Steam Generators

a) Tube Vibration Analysis

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In the design of Westinghouse Model 44F steam generators used in Indian Point 3, the possibility of tube degradation due to either mechanical or flow-induced excitation was considered. This evaluation included detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

Consideration was given to potential sources of tube excitation including primary fluid flow within the U-tubes, mechanically-induced vibration, and secondary fluid flow on the outside of the tubes. The effects of primary fluid flow and mechanically-induced vibration are considered to be negligible during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation by the secondary fluid on the outside of the tubes, and this area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms due to hydrodynamic excitation by the secondary fluid on the outside of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluidelastic vibration mechanisms.

Vortex shedding is possible, at most, only for the outer few rows in the wrapper inlet region of steam generators such as the Model 44F for which non-uniform, two-phase turbulent flow exists throughout most of the tube bundle. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the wrapper inlet region. Bounding calculations consistent with laboratory tests parameters confirmed that vibration amplitudes would be less than .001 inches, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the steam generator.

Flow-induced vibrations due to flow turbulence are also small: root mean square amplitudes are less than allowances used in tube sizing, and these vibrations cause stresses which are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation is anticipated due to secondary flow turbulence in the Model 44F design configuration.

Fluidelastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism: relatively large tube amplitudes can feedback proportional driving forces if an instability threshold is exceeded. Tube support spacing incorporated into design of both the tube support plates and the anti-vibration bars in the U-bend region provides tube response frequencies such that the instability threshold is not exceeded for secondary fluid flow conditions. This approach provides large margins against initiation of fluidelastic vibration for tubes which are effectively supported by the Model 44F tube support configuration.

Small clearances between the tubes and supporting structure are required for steam generator fabrication. These clearances introduce the potential that any given tube support location may not be totally effective in restraining tube motion

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if there is a finite gap around the tube at the location. Fluidelastic tube response within available support clearance is therefore theoretically possible if secondary flow conditions exceed the instability threshold assuming no support at the location with a gap around the tube.

This potential has been investigated both with tests and analyses for the U-bend region where secondary flow conditions have the potential to exceed the instability threshold if a tube does not contact two or more sequential supports as a result of fabrication tolerances. Tube vibration response is shown to have wear potential within available design margins even for limiting tube fit-up conditions which are not expected. Corresponding tube bending stresses remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. These analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

Analyses and tests therefore demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the Model 44F steam generators at Indian Point 3. Operating experience with similar steam generators supports this conclusion.

b) Tubesheet Analysis

The evaluation of the Indian Point Unit 3 steam generator tubesheets was performed according to the rules of the ASME Boiler and Pressure Vessel Code. The design criteria encompassed consideration of both steady state, transient and emergency operations specified in the steam generator Design Specification for the plant.

The evaluation of the tubesheet involved the heat conduction and stress analysis of the tubesheet, channel head, and secondary shell structure for particular steady design conditions for which the Code stress limitations were to be satisfied, and for discrete points during transient operation for which the temperature/pressure conditions were to be known in order to evaluate stress minima and maxima for fatigue life usage.

The stress analysis of the tubesheet complex consisted of performing an interaction analysis between the tubesheet and the channel head and attached lower shell to determine the interaction forces and moments between parts of the structure. A finite element computer program is used to calculate the stress intensities. The tubesheet calculations were made with a ligament efficiency determined using guidelines from Article I-9 from the ASME Code, Section III (1965) Edition, Winter 1965 Addenda). The stress analysis considered stress due to symmetric temperature and pressure differential as well as asymmetric temperature distribution due to temperature drop across the tubesheet divider lane.

The fatigue analysis of the tubesheet was performed at potentially critical regions, such as the junction between the tubesheet and the channel head and secondary shell, as well as representative locations in the perforated region of the tubesheet. The nominal stress results from the primary chamber

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components interaction analysis are used as the basis for the fatigue evaluation. These stresses are modified by stress concentration factors which are a function of position around the circumference of the hole. The fatigue evaluation is performed at the hole evaluated for the nominal location. The Model 44F steam generators for Indian Point Unit 3 did not have any misdrilled holes to be evaluated. Fatigue usage is evaluated for several locations around the circumference of the hole.

In all cases evaluated, the Indian Point Unit 3 steam generator tubesheet complex met the stress limitations and fatigue criteria.

c) Tube-Tubesheet Analysis

The tubes in the Indian Point Unit 3 Model 44F steam generators are hydraulically expanded the full depth of the tubesheet. This expansion of the tubes into the tubesheet adds to the strength of the tubesheet for some loading conditions. However, the effect of the tubes is not accounted for in the analysis.

Due to the expansion of the tube into the tubesheet, the effects of fluid induced vibration of the tube terminate at or near the secondary surface of the tubesheet. The fatigue usage is calculated for the tube at the tube to tubesheet juncture at the secondary surface of the tubesheet. An appropriate fatigue strength reduction factor is used at that location. The calculated fatigue usage for the tube to tubesheet juncture on the secondary side is less than the Code limit.

d) Tube to Tubesheet Weld

At the primary surface of the tubesheet the tube is joined to the tubesheet with a weld of the tube to the tubesheet cladding. The weld is not subject to any effects of tube vibration on the secondary side. The stress analysis of the weld considers thermal stresses due to increasing and decreasing transient temperature conditions. A finite element model was used to determine the states of stress in the tube to tubesheet weld. The model included the effect of adjacent tube holes o n the stiffness of the tubesheet. The stress state of the tubesheet surrounding the tube was based on the tubesheet analysis. The weld was evaluated for fatigue usage using an appropriate fatigue strength reduction factor. The calculated fatigue usage for the weld is less than the Code limit.

Reactor Coolant Pumps

The RCP motors are equipped with a Bently-Nevada vibration monitoring system that provides continuous monitoring of both shaft and motor frame vibration. Continuous Control Room recording can be provided by this system via selector switches and chart recorders for trending and troubleshooting RCP vibration. Additionally, vibration alarms are provided for the RCP's. A Control Room alarm will be annunicated if an RCP vibration exceeds the predetermined level. A noise detector is also located adjacent to each motor and may be placed in service via a selector switch in the control room.

Each of the reactor coolant pumps is also equipped with two International Research and Development vibration pickups mounted at the top of the motor support stand. They are mounted in a horizontal plane and pick up radial vibrations of the pump. One is aligned

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parallel to the pump discharge; the other is aligned perpendicular to the pump discharge. Their signals are taken to a multi-point selector switch mounted outside the reactor containment. Both signals form each reactor coolant pump are taken to this selector switch. Also supplied is a vibration meter. This is a portable device that may be plugged into the selector switch, so the signal from any one pickup may be monitored at one time. The vibration levels of the reactor coolant pumps can thus be checked and the pumps were checked during full flow tests for flow induced vibration (Reference 19).

No analysis for vibration loading was performed for the reactor coolant pumps.

Primary System Piping

The reactor Coolant System piping was designed for normal and faulted conditions. For faulted conditions, the piping was designed and analyzed for seismic loads and blowdown forces due to a Loss-of-Coolant Accident. By design, the main piping of the reactor coolant loop is not subjected to induced pressure pulse vibration from the reactor coolant pump impeller or the pistons of the charging pump.

The perturbating frequency of the reactor coolant pump is quite high when compared to the piping natural frequency. Frequency separation, therefore, insures a very small probability for self-excited or sympathetic vibration. This is borne out of satisfactory operation of several representative coolant loops.

4.3.4 Overpressure Protection

An overpressure Protection System (OPS) is required to prevent the reactor vessel pressure from exceeding the Technical Specifications (Appendix G) and ASME Code Case N-514 limits. The problem arises inasmuch as the reactor vessel steel has less ductility at low temperatures. As the reactor vessel is irradiated during its lifetime, the limitations become even more stringent.

The OPS is a three-channel analog curve tracking arrangement which can initiate an appropriate chain of coincidence logic for the purpose of automatically preventing a violation of the technical specification temperature/pressure limit curve for the reactor vessel.

Wide range RCS temperature signals will be used to perform two primary functions in this system:

1) Provide the arming and disarming function, and

2) Serve as the independent variable in computing the reference curve which is the system pressure limit that must be adhered to. The basis of the OPS curve is the Appendix G isothermal cooldown curve increased by 10% (as allowed by ASME Code Case N-514) and then adjusted to allow for effects of the design basis heat input and mass input events.

   The arming function of the Overpressure Protection System is activated when the RCS temperature is below a predetermined value as specified in the Technical Specifications. Below this temperature, one half of a two-out-of-two (temperature-pressure) coincidence logic is satisfied to allow the air operated valves (PCV-455C, 456) to open automatically in the event of an

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impending overpressure condition. This automatic opening of the valves is initiated by the other half of the two-out-of-two coincidence logic which is described below.

These same temperature signals are also fed into three respective function generators whose task is to output values of pressure as a function of the input temperature which are the maximum RCS pressures allowed at those temperatures. The difference between the Appendix G/ASME Code Case N-514 curve pressures and the actual RCS pressure transmitted by three 0 to 1500 psig transmitters is computed in each of the three channels, and if any two-out-of-three of these differences is smaller than a preset minimum, a trip open condition will be initiated for each pressurizer power operated relief valve. The power operated relief valves are normally operated using the N2 system with accumulators at each valve.

However, the Overpressure Protection System requires an N2 supply of sufficient capacity which, in case of loss of the main N2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N2 supply is provided from one Safety Injection Accumulator having its associated N2 fill valve blocked open.

The system also includes appropriate accessory equipment to provide adequate testability, calibration facilities and operator surveillance instrumentation.

The main function of the power operated relief valves (PORVs) is to open automatically at 2335 psig to protect against pressure surges. Under the OPS, the PORVs will also open automatically when the reactor coolant temperature is within the temperature range specified in the Technical Specifications, to relieve the RCS from exceeding the Appendix G/ASME Code Case N-514 curves. Under this mode of operation, the PORVs will relieve solid water rather than steam-water mixture at 2335 psi.

Westinghouse conducted a generic reactor vessel overpressure study and the analytical results showed that only one PORV is necessary to turn around the design mass input overpressure incident and the design thermal input overpressure incident. The motor operated valves and power operated relief valves feed to the pressurizer relief tank. The tank has sufficient capacity to accept the expected short term flow from the OPS.

The electrical activation uses two-out-of-three logic for valve activation. The electrical supply is from the instrument buses with backup by the station emergency batteries. Thus, the system is single failure proof in addition to having a secured power supply.

The system allows for accurate control of system heatup and cooldown. Spurious opening and/or closing of the PORVs is essentially eliminated by the new two-out-of-three logic (if one channel were to fail the valve would not be able to malfunction).

4.3.5 System Incident Potential

The potential of the Reactor Coolant System as a cause of accidents was evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Section 14.1 and 14.2. Reactor coolant pipe rupture is evaluated in Section 14.3.

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4.3.6 Loose Parts Monitoring

The metal impact monitor was designed to enable early detection of loose metallic parts which may be in the steam generator or the reactor vessel. Upon the occurrence of an impact of loose metallic parts, a pressure wave is generated in the reactor system component causing minute displacements in the component material. The step excitation of the impact produces a broadband frequency response with peak amplitude response at resonant frequencies. Many of these resonant frequencies lie in the audible frequency range and are called "Bell" frequencies. Certain of these "Bell" frequencies are especially sensitive to impact excitation due to differing modes of preferred vibration response.

The displacements attendant with an impact wave are insignificant. However, the accelerations caused by the moderately high audio frequencies are very significant; for this reason, acceleration is the parameter chosen to indicate impact.

Acceleration Measurement

Acceleration is measured by the use of special transducers that convert accelerations to electrical signals. These transducers are mounted at specially selected monitoring points on the exterior of the Reactor Coolant System. Monitoring points normally in use during plant operation are at the top and bottom of the reactor vessel and above and below each steam generator tube sheet with transducers mounted on the generator shell. Additional monitoring points are available above and below each steam generator transition cone and above each feedwater nozzle.

Each transducer contains a piezoceramic material fabricated in a manner that provides for changes in compression of the material in response to accelerations. A small electric charge proportional to acceleration is generated by piezoelectric behavior. The charge is converted to a voltage signal by a charge preamplifier that operates in a feedback manner to continuously compensate for charge input changes to maintain a near-zero differential input voltage. A small current flows in the cable connecting the transducer to the charge pre-amplifier resulting in virtual elimination of cable effects on the acceleration signal. The charge preamplifier output voltage can then be treated as a normal instrument signal requiring only normal shielding and cable considerations.

The sensitivity of this measurement to impact is determined by the impact energy (determined by mass and impact velocity) and the distance from the point of impact to the measurement point (damping and geometry changes attenuate the traveling wave).

Impact Monitoring

The voltage signal from the charge preamplifier is further amplified and then fed to a monitoring device (Digital Metal Impact Monitor). This device excludes all frequency information except for the "Bell" frequencies from the impact. The selected signals are further processed to provide an output DC signal proportional to the impact energy (as seen by the transducer) and a DC signal indicating the rate of occurrence of impact repetition.

Records and Displays

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Every 24 hours and whenever the alarm setpoint is exceeded the impact energy level and rate of occurrence can be displayed on a printer. These records serve to establish a history for establishing when and where impacts were observed. The rate at which impacts occur gives an indication of the amount of debris present in the monitored area while the impact energy is a measure of the weight of the debris.

A test circuit is provided for determining the continuity of the monitoring system. The test signal inputs a representative wave form similar to a metallic impact.

Cabling

The signal cables are an integral part of the accelerometer. They are constructed of high temperature and radiation resistant materials and transmit the accelerometer signal to the charge preamplifier.

4.3.7 Cold Shutdown RCS Level Indication

The original system used to monitor RCS level indication during cold shutdown conditions was replaced and upgraded with a new system in accordance with NRC Generic Letters 87-12 – "Loss of RHR While RCS Is Partially Filled" and 88-17 – "Loss of Decay heat Removal." The new system provides two independent RCS water level indicators and the Westinghouse Ultrasonic Level Measuring System for use during reduced inventory conditions. Filling the RCS while under vacuum was a process introduced in RO11. In order to maintain compliance with Generic Letter 88-17, another RCS Level Monitoring system (Mansell Level Monitoring System, i.e. MLMS) was introduced during RO11 which was capable of providing RCS level indication while in reduced inventory and mid-loop conditions while the RCS is under vacuum as well as non-vacuum conditions. Hand held UT is required to be used with MLMS in mid-loop condition to increase the range of level indication in the hot leg. The four system includes:

1) Two redesigned, independent water level columns, with the supply and vent lines for the level gauges being constructed of stainless steel piping for strength and compatibility with the REC. The level gauges are constructed of 2 inch diameter transparent rigid polyvinyl chloride (PVC) tube and span a range of the RCS from the 78 foot elevation (=10% in the pressurizer) down to the 60.8 foot elevation (inside bottom of hot leg piping), well below the RCS "midloop" elevation of 62' – 0". The columns are independently supplied from the low point drain lines of the RCS intermediate loop piping, one level column from loop #32 and the other from loop #34. The level gauges can be viewed in the Control Room by remotely operated cameras.

2) A Westinghouse Ultrasonic Level measuring System (ULMS) that provides control room indications of water level in the reactor coolant piping during non-power operations. The ULMS consists of the following three basic components.

   a) The sensor, which transmits and receives the ultrasonic waves,

   b) The preamp/pulsar module, which drives the sensor and preamplifies the received echo signal, and

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c) The signal processing module, which provides an analog output signal proportional to the ultrasonic wave travel time in the water.

3) A MLMS that is capable of providing the Control Room indications of water level in the reactor coolant piping during non-power operation. The MLMS utilizes sensing locations above and below where the RCS water levels are expected to remain during outages. The MLMS consists of two independent indication loops, a wide range and a narrow range loop. The wide range loop has an indication range between 121 foot elevation down to the 62 foot elevation. The wide range high point connects to a vent on the pressurizer relief line. The wide range loop low point connects to a drain off of PT-413. The narrow range loop has an indication range between 77 foot elevation down to 62 foot elevation. The narrow range high point connects to a drain off of PT-433. The pressure transducer assembly (consisting of two redundant transducers per assembly) is connected to each of the four sensing locations. The pressure transducers transmit signal to two computers located in the Control Room, i.e. one narrow range and one wide range computer. The pressure difference between the high and low points is calculated by the computers and is displayed as water level on computer screens, as well as LED displays in the Control Room. The computers have an alarm feature.
4) Two hand held UT devices capable of providing indication of water level inside the reactor coolant piping during non-power operations. The UT devices are manually positioned at the 6:00 position of the RCS hot legs and level in the pipe is read locally and communicated back to the Control Room. This method of level indication is only used while the RCS is in mid-loop and may be used to comply with Generic Letter 88-17, but must be used in conjunction with the MLMS system in mid-loop operation.

In order to comply with the requirements of Generic Letter 88-17, two independent continuous RCS water level indications are necessary when the RCS is in a reduced inventory condition. Water level indication should be periodically checked and recorded by an operator or automatically and continuously monitored and alarmed. The ULMS and MLMS system provide an alarm function. The water level columns and UT devices do not. Any combination of two independent indicators may be used for level indication provided they are capable of performing within the RCS pressure (vacuum) ranges. Examples include: one level column and ULMS; one MLMS and one ULMS; two MLMS and two UT devices; two UT devices, etc. Operations procedures delineate specific actions required for the particular combination of level indicators in service.

4.3.8 Saturation Alarm and Recorder (Analog System)

The saturation recorder provides subcooling trend data to the operator which will note if saturation conditions in the Reactor Coolant System occur.

The system involves a calculating device to develop a saturation curve, an alarm duplex bistable (alarm in CCR disabled), an isolating amplifier, a high selector, a low selector and a summing unit to form a difference signal between reactor coolant temperature and saturation temperature.

The RCS temperature is determined by the use of four RCS temperature loops taken from all four hot leg instrument loops. The four signals from these loops are channeled into a

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device which selects the highest temperature signal and feeds it to a duplex bistable and a summing unit.

The programmed RCS saturation temperature is determined by the use of two RCS pressure loops. The two signals from these loops are channeled into a device which selects the lowest pressure signal and feeds it to a characterizer where a corresponding saturation temperature signal is derived. This temperature signal is fed into the duplex bistable and summing unit. The duplex bistable trips when the difference in temperature (RCS vs. saturation) approaches saturation conditions and when the difference in temperature is at saturation. These alarm functions are disabled from alarming in the CCR.

This system is operational during the natural circulation cooldown mode because the temperature readings in this mode are derived from the RCS loops.

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3) DELETED 6/2000

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18) WCAP-7408-L Topical Report, "Robert E. Ginna Unit #1 Internals Assurance Programs."

19) Westinghouse NSD Technical Bulletin TB 75-3, "Reactor Coolant Pump (RCP) Vibration Limits for Type 93 & 93A Pumps"

20) Westinghouse Letter No. INT-98-227, "RCP Flywheel Integrity," dated October 5, 1998.

21) Dominion Engineering Inc. Report R-4147-00-1, "Reactor Vessel Tensioning Optimization Stress Report – Indian Point Units 2 & 3."

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TABLE 4.3-1

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY
FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity (psi)	Allowable Stress 3Sm (psi) (Operating Temperature)
Control Rod Housing	55,300	69,900
Head Flange	51,820	80,100
Vessel Flange	52,890	80,100
Closure Studs	109,400	110,300
Primary Nozzles – Inlet	45,500	80,100
Outlet	49,390	80,100
Core Support Pad	55,740	69,900
Bottom Head to Shell	37,700	80,100
Bottom Instrumentation	56,800	69,900
Nozzle Belt to Shell	42,630	80,100
CRDM Adapter Plug	27,630	48,600

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TABLE 4.3-2

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*
Control rod housing	0.124
Head Flange	0.024
Vessel Flange	0.023
Stud Bolts	0.908
Primary Nozzles – Inlet	0.049
Outlet	0.259
Core Support Pad (Lateral)	0.052
Bottom Head to Shell	0.020
Bottom Instrumentation	0.206
Nozzle Belt to Shell	0.002
CRDM Adapter Plug	0.0036

*As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessel.

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TABLE 4.3-3

STRESSES DUE TO MAXIMUM STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL (2485 PSI)

Stress	(668 F) Computed Value	Allowable Value
Primary Membrane Stress	6,960 psi	56,000 psi (0.70 Su)
Primary Membrane plus Primary Bending Stress	54,650 psi	84,000 psi (1.05 Su)

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TABLE 4.3-4

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL OF 2485 PSI

Component Part	Stress Ratio
Channel Head	1.90
Channel Head-Tube Sheet Joint	1.67
Tubes	2.08
Tube Sheet (Average Ligament)	1.54

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TABLE 4.3-5

CHARPY V-NOTCH TEST RESULTS IP-3 FLYWHEELS AT +10°F
(CERTIFICATION DATA)

1. Flywheel No. 1

A. 5 In Thick Plate — Heat No. C4344 Slab No. 3

	1	2	3
"Weak" direction (ft-lbs)	30	32	33
90° to "weak" direction (ft-lbs)	60	75	80

B. 8 In Thick Plate — Heat No. C4908 Slab No. 1A

	1	2	3
"Weak" direction	57	57	53
90° to "weak" direction	38	44	49

2. Flywheel No. 2

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

	1	2	3
"Weak" direction	96	101	104
90° to "weak" direction	50	47	53

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

	1	2	3
"Weak" direction	58	52	57
90° to "weak" direction	95	78	77

3. Flywheel No. 3

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

Results as for Flywheel No. 2 (Case 2A)

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TABLE 4.3-5
(Cont.)

CHARPY V-NOTCH TEST RESULTS IP-3 FLYWHEELS AT +10°F
(CERTIFICATION DATA)

B. 8 In Thick Plate Heat No. C4909 Slab No. 1

Results as for Flywheel No. 2 (Case 2B)

4. Flywheel No. 4

A. 5 In Thick Plate Heat No. C4679 Slab No. 3

Results as for Flywheel No. 2 (Case 2A)

B. 8 In Thick Plate Heat No. C4909 Slab No. 1

Results as for Flywheel No. 2 (Case 2B)

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4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 System Heatup and Cooldown Rates

The operating limits for the Reactor Coolant System heatup and cooldown rates are defined in the Technical Specifications. These limits are recalculated periodically using methods derived from Appendix G, Protection Against Non-Ductile Failure, of Section III of the ASME Boiler and Pressure Vessel Code and ASME Code case N-514. The ASME approach utilizes fracture mechanics concepts and is based on the reference nil-ductility transition temperature, RT_{NDT}. The method calculation for the Indian Point 3 operating limits for heat up and cooldown rates is described in detail in Reference 10.

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

RT_{NDT} and, turn, the operating limits of the nuclear power plant can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of the pressure vessel steel are monitored by the "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program" (WCAP-8475, Reference 2). Surveillance capsules are periodically removed from the reactor, one at a time, and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} as an adjustment for radiation embrittlement. The radiation induced ΔRT_{NDT} can be estimated from Figure 4.4-1. The adjusted RT_{NDT} is used to index the material to the K_{IR} curve and, in turn, to reset the operating limits of the plant to take into account the effects of irradiation on the reactor vessel materials.

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. The analysis of this capsule (T) (Reference 3) forms the basis of the original technical specification operating limits for RCS heatup and cooldown rates for up to 9.26 Effective Full Power Years (EFPY's). The second and third reactor vessel material surveillance capsules were removed in 1982 and 1987, respectively. These capsules have been tested and the results have been evaluated and reported (References 4,5,6,7). No change to the technical specification heatup and cooldown limit were required as a result of the Capsule Y and Capsule Z analyses.

In May 1991, 10 CFR 50.61 (Reference 9) was amended to include the provisions of Reg. Guide 1.99, Rev. 2 (Reference 8), which provided a slightly different methodology for calculation of reactor vessel embrittlement.

The heatup-cooldown curves were accordingly regenerated using this new rule for plant operating periods of 9, 11, 13.3, 16.2, and 20 EFPY. Future Technical Specifications Amendment proposals for greater operating periods will be submitted, using this new rule,

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as appropriate (References 10 and 12). The fourth surveillance capsule (Capsule X) was removed in 2003. No changes to the Technical Specifications were required as a result of Capsule X analysis (Ref. 13).

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure will continue to be obtained form ongoing and future capsule analyses in accordance with the Indian Point 3 reactor vessel radiation surveillance program.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material, automatically provides additional conservatism for the non-irradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

Table 4.4-1 provides the material toughness test requirements and data that were specified and reported for plates, forgings, piping, and weld material prior to plant operation. Specifically, the following data were provided:

1) The maximum NDT temperature as obtained from DWT results.
2) The maximum temperature corresponding to the 50 ft-lb value of the Cv fracture energy.
3) The minimum upper shelf Cv energy value for the weak direction (WR direction in plates) of the material.

For the limiting reactor vessel beltline materials, the end-of-life RT_{PTS} was estimated to be within the screening criteria of 10 CFR 50.61 for plate metal and welds (Reference 11). (Reg. Guide 1.99, Rev. 2 defines the property RT_{PTS} as an indicator of vessel embrittlement.)

The analysis of the reactor vessel material contained in surveillance Capsule T showed that the irradiated properties of the limiting reactor vessel beltline materials exceeded those predicted. However, due to the effects of low-leakage core loading strategy, measured beltline properties were within predicted values by the time Capsule Z was analyzed.

4.4.2 Reactor Coolant Activity Limits

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the coolant could constitute a hazard only if the reactor coolant system barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The plant systems were designed for operation with activity in the Reactor Coolant System corresponding to 1 percent fuel defect. The waste gas system is designed such that rupture of a gas decay tank, following a refueling shutdown wherein the gaseous activity is removed from the reactor coolant to the waste gas tanks for decay, will not result in an offset whole body exposure in excess of 0.5 rem. In the event of a steam generator tube rupture a high activity level signal at the condenser air ejector exhaust will divert the radioactive discharge back into Containment.

4.4.3 Maximum Pressure

The Reactor Coolant System serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor

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Coolant System is assured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III. Reactor Coolant System pressure settings are given in Table 4.1-1.

4.4.4 System Minimum Operating Conditions

Minimum operating conditions for the Reactor Coolant System for all phases of operation are given in the Technical Specifications and Technical Requirements Manual.

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10) "Final Report on Pressure Temperature Limits for Indian Point 3 Nuclear Power Plant," ABB Combustion Engineering, July 1990.

11) Memorandum REC:92-003, dated February 10, 1992, F. Gumble to P. Kokolakis

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12) "Indian Point Unit 3 Section XI Enable Temperatures for 13 and 15 EFPY," ABB Combustion Engineering, August 14, 1997.

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TABLE 4.4-1

MATERIAL TOUGHNESS TEST REQUIREMENTS AND DATA

Component	Material Grade	Cu (%)	(Drop Wt.) NDTT (F)	50 Ft-lb/35 Mil Temp. Long. (F)	50 Ft-lb/35 Mil Temp. Trans. (F)	RT_{NDT} (F)	Minimum Upper Shelf Long. (Ft-lb)	Minimum Upper Shelf Trans. (Ft-1b)
I. Reactor Vessel								
a. Closure Hd. Dome	A533,B,C1 1	0.13	+10	93	148[a]	88	85	55[b]
b. Clos. Hd. Peel Seg.	A533,B,C1 1	0.14	0	83	112[a]	52	108	70[b]
c. Clos. Hd Peel Seg.	A533,B,C1 1	0.14	+10	51	80[a]	20	128	83[b]
d. Clos. Hd. Peel Seg.	A533,B,C1 1	0.13	+10	78	99[a]	39	117	76[b]
e. Head Flange	A508, C1 2	NA	+ 3*	-22	- 8[a]	3	117	76[b]
f. Vessel Flange	A508, C1 2	NA	+38*	-5	16[a]	38	141	92[b]
g. Inlet Nozzle	A508, C1 2	NA	+20*	-28	-2[a]	20	154	100[b]
h. Inlet Nozzle	A508, C1 2	NA	+45*	-8	40[a]	45	120	78[b]
i. Inlet Nozzle	A508, C1 2	NA	+40*	-7	20[a]	40	158	103[b]
j. Inlet Nozzle	A508, C1 2	NA	+12*	-14	0[a]	12	155	101[b]
k. Outlet Nozzle	A508, C1 2	NA	+60*	60	150[a]	90	72.5	47[b]
l. Outlet nozzle	A508, C1 2	NA	+60*	4	44[a]	60	105.5	69[b]
m. Outlet Nozzle	A508, C1 2	NA	+60*	4	44[a]	60	96	62[b]
n. Outlet Nozzle	A508, C1 2	NA	+60*	-2	43[a]	60	123.5	80[b]
o. Upper Shell	A533, B,C1 1	NA	-50	74	128[a]	68	90 (95% shear)	58[b]
p. Upper Shell	A533, B,C1 1	0.20	-40	84	130[a]	70	100	65[b]
q. Upper Shell	A533, B,C1 1	NA	-40	43	82[a]	22	127	82.5[b]
r. Inter. Shell	A533, B,C1 1	0.20	-50	24	65**	5	134	97**

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TABLE 4.4-1
(Cont.)

MATERIAL TOUGHNESS TEST REQUIREMENTS AND DATA

Component	Material Grade	Cu (%)	(Drop Wt.) NDTT (F)	50 Ft-lb/35 Mil Temp. Long. (F)	50 Ft-lb/35 Mil Temp. Trans. (F)	RT_{NDT} (F)	Minimum Upper Shelf Long. (Ft-lb)	Minimum Upper Shelf Trans. (Ft-1b)
I. Reactor Vessel								
s. Inter. Shell	A533, B,C1 1	0.22	-50	23	56**	-4	113	86**
t. Inter. Shell	A533, B,C1 1	0.20	-40	40	77**	17	113	85**
u. Lower Shell	A533, B,C1 1	0.19	0	74	109**	49	90	65**
v. Lower Shell	A533, B,C1 1	0.22	-20	6	55**	-5	134	89**
w. Lower Shell	A533, B,C1 1	0.24	-10	68**	134**	74	105**	62**
x. Bot. Hd. Peel Seg.	A533, B,C1 1	0.13	-40	23	62[a]	2	103	67[b]
y. Bot. Hd. Peel Seg.	A533, B,C1 1	0.16	-40	33	56[a]	-4	108	70[b]
z. Bot. Hd. Peel Seg.	A533, B,C1 1	0.13	-40	38	69[a]	9	106	69[b]
aa. Bottom Hd. Dome	A533, B,C1 1	0.13	-30	60	107[a]	47	80	52[b]
ab. WELD	-	0.15**	0*	-	5**	0	-	112**
ac. HAZ	-	NA	NA	-	10**	-	-	111**

NA – Not Available

* Estimated (60F or 100 Ft-lb temp., whichever is less for forgings; OF or 30 Ft-1b temp, whichever is higher for welds).
a) Estimated when no transverse data are available. (77 Ft-lb/54 mil longitudinal temp.)
b) Estimated when no transverse data are available. (65% of longitudinal shelf)
** Westinghouse data (all other data provided by the vessel fabricator).

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4.5 INSPECTIONS AND TESTS

4.5.1 Inspection of Materials and Components Prior to Operation

Table 4.5-1 summarizes the quality assurance program for all Reactor Coolant System components. In this table, all of the non-destructive tests and inspections which were required by Westinghouse specifications on Reactor Coolant System components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which were more stringent in some areas than those requirements specified in the applicable codes, are also included. The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used for the reactor vessel.

Westinghouse required, as part of its reactor vessel specification, that certain special tests not specified by the applicable codes be performed. These tests are listed below:

1) Ultrasonic Testing – Westinghouse required that a 100% volumetric ultrasonic test (both shear wave and longitudinal wave) of reactor vessel plate be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test. The 100% volumetric ultrasonic test by both techniques was a severe requirement, but it assured that the plates were of the highest quality.

2) Radiation Surveillance Program – In the surveillance program, the evaluation of the radiation damage was based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) fracture mechanics test specimens.

4.5.2 Reactor Vessel Surveillance

The reactor vessel surveillance program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature and the fracture mechanics approach. The reactor vessel surveillance program includes specimens from the most limiting plate used in the core region of the reactor vessel. Three capsules meet the requirements of ASTM-E-185-70 except that the limiting plate specimen orientation is transverse (weak direction) rather than longitudinal (strong direction). Five additional capsules do not meet E-185-70 since they do not include HAZ specimens. The program is essentially in accordance with ASTM-E-185-70, and ASTM-E-185-79 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," except that three capsules contain, in addition to the specified Charpy specimens taken from the weld metal, the heat affected zone and the ASTM reference plate, core region base metal specimens oriented normal (transverse) to the principal rolling direction of the plate rather than parallel to the principal rolling direction.

Five additional capsules, not required by ASTM-E-185 but included in the program, contain both longitudinal and transverse Charpy specimens taken from the limiting core region material, and longitudinal tensile and Charpy specimens of one of the other core region plates; however, they do not contain weld heat affected zone specimens of ASTM reference correlation monitor specimens. The surveillance program does not include thermal control specimens. These specimens were not required since the surveillance specimens are exposed to the combined neutron irradiation and temperature effects and the test results

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provide the maximum transition temperature shift. Thermal control specimens as considered in ASTM-E-185 would not provide any additional information on which the operational limits for the reactor are set.

Test stock from four plates (three intermediate shell plates and the most limiting lower shell plate) used in the core region of the reactor vessel, and a weldment containing representative (as deposited) weld metal (but no HAZ representing the limiting plate) were retained as per Section 3.1.2 of ASTM E-185-70.

Chemical analyses (excluding nitrogen and iron) as per Section 3.1.3 of ASTM #-185-70 were obtained for the limiting core region plate and weld metal.

The reactor vessel surveillance program uses eight specimen capsules. The capsules are located about 3 inches from the vessel wall directly opposite the center of the core and are retained in guide baskets welded to the outside of the thermal shield. Sketches of an elevation and plan view showing the location and dimensional spacing of the capsules with relation to the core, thermal shield and vessel are shown in Figures 4.5.1 and 4.5.2, respectively. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The capsules contain reactor vessel steel specimens from shell plates located in the core region of the reactor and from associated weld metal and heat affected zone metal. In addition, three capsules contain correlation monitors made from fully documented specimens of SA-533 Grade B, Class 1 material obtained through Subcommittee II of ASTM Committee E10, "Radioisotopes and Radiation Effects". The capsules contain tensile specimens, Charpy V-notch specimens (which include weld metal and heat affected zone material) and WOL specimens.

Sixty-four Charpy V-notch specimens (oriented with respect to the weak direction) for one of the lower shell plates were included in the reactor vessel surveillance program. This lower shell plate is the limiting material in the core region as defined by E-185.

Dosimeters, including Ni, Cu, Co-Al, Ed shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238, for capsules V, Y & S were placed in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys were included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and to ensure good thermal conductivity.

The complete capsule was helium leak tested. Vessel material sufficient for at least 2 capsules will be kept in storage. This material represents four plates (three intermediate shell plates and the most limiting lower shell course plate) used in the core region of the reactor vessel and a representative weldment. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01% is made for surveillance material and as deposited weld metal.

Each of three capsules contain the following specimens:

CAPSULES V. Y***** and S

Material	No. of Charpys	No. of Tensiles	No. of WOL's

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Plate B2803-3*	8	2	-
Weld Metal	8	2	4
Heat affected zone metal	8	-	-
ASTM Reference	8	-	-

The following dosimeters and thermal monitors are included in each of the three capsules:

DOSIMETERS

Copper

Nickel

Cobalt – Aluminum (0.15% Co)

Cobalt – Aluminum (Cadmium shielded)

U-238 (Cadmium shielded)

Np-237 (Cadmium shielded)

THERMAL MONITORS

97.5% Pb, 2.5% Ag (579 F Melting Point)

97.5% Pb, 1.75% Ag, 0.75% Sn (590° F Melting Point)

Five additional capsules contain the following specimens:

Capsules	Material	No. of Charpys	No. of Tensiles	No. of WOL's
W and T ****	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-1***	8	1	6
	Weld Metal	8	-	-
X and U	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-2***	8	2	6
	Weld Metal	8	-	-

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Z*****	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-3***	8	2	6
	Weld Metal	8	-	-

* Lower shell plate specimens oriented normal (transverse) to the principal rolling direction of the plate.

** Lower shell plate specimens oriented parallel (longitudinal) to the principal rolling direction of the plate.

*** Intermediate shell plate specimens oriented parallel (longitudinal) to the principal rolling direction of the plate.

**** Capsule T has been removed and analyzed.

***** Capsule Z has been removed and analyzed.

The following dosimeters and thermal monitors are included in each of the five capsules:

Dosimeters

Copper
Nickel
Cobalt-Aluminum (0.15% Co)
Cobalt-Aluminum (Cadmium shielded)
Iron

Thermal Monitors

97.5% Pb, 2.5% Ag (579°F Melting Point)
97.5% Pb, 1.75% Ag, 0.75% Sn (590°F Melting Point)

The fast neutron exposure of the specimens occurs at a rate equal to or faster than the maximum exposure experienced by the vessel wall with the specimens being located between the core and the vessel. Since some of these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the RT_{NDT} determinations of these specimens are representative of the vessel at a later time in service.

Data from the WOL fracture toughness specimens provide additional information for use in determining allowable stresses for irradiated material.
The calculated average fast neutron exposure at the vessel clad-base metal interface was 5.86×10^8 n/cm^2 (E greater than 1 MeV at the end of Cycle 12). The reactor vessel surveillance capsules are located at 4°, 40°, and 220° as shown in Figure 4.5-2. The design basis lead factor and the plant specific lead factor are listed below.

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Capsules at	Design Basis Lead Factor	Plant Specific Lead Factor
4°	1.07	1.30
40°	3.74	3.74
220°	3.46	3.44

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Appendix 4A. The analysis of the reactor vessel material contained in Surveillance Capsule T for Indian Point 3 was reported in WCAP-9491, April 1979. The analysis of the reactor vessel material contained in Surveillance Capsule Y for Indian Point 3 was reported in WCAP-10300, March 1983. The analysis of the reactor vessel material contained in Surveillance Capsule Z for Indian Point 3 was reported in WCAP-11815, March 1988. The analysis of reactor vessel material contained in Surveillance Capsule X for Indian Point Unit 3 was reported in WCAP – 16251-NP, July 2004. The Capsule T report indicated that the damage rate of the plate and weld metal due to irradiation is in excess of that predicted by the Westinghouse trend curves and that the calculated lead factors were slightly higher than originally estimated. However, due to the effects of low-leakage core loading strategy, measured belting plate and weld metal properties were found to be slightly better than design by the time Capsule Z was analyzed.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and readjustment of the calculated wall exposure are made by use of data on all the capsules withdrawn as was done for Capsule T, Capsule Y, and Capsule Z.

The tentative schedule for removal of the capsules is as follows:

CAPSULE	REMOVAL TIME
T	Removed (1978 Refueling Outage, At the Replacement of the First Region of the Core, 1.34 EFPY*)
Y	Removed (1982 Refueling Outage, 3.13 EFPY)
Z	Removed (1987 Refueling Outage, 5.55 EFPY)
S	**
X	Removed (2203 Refueling Outage, 15.6 EFPY)
U	30 Years or 25.5 EFPY, assuming an 85% capacity)
V	Standby
W	Standby

*NOTE: Effective full power years from plant startup.

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**Capsule S, scheduled for removal in the 2001 outage, was found to be inaccessible due to equipment interference and has therefore been removed from the program. The schedule for specimen retrieval beyond Capsule Z was revised in 2003 in order to optimize the benefits gained from specimen analysis in the latter half of plant live.

This plan was developed assuming an 85% capacity factor over plant life. Accordingly, the times for removal may be extended to allow for historical capacity factor below 85%.

4.5.3 Primary System Quality Assurance Program

Table 4.5-1 summarizes the quality assurance program with regard to inspections performed on primary system components, including the replacement steam generators installed during the cycle 6/7 refueling outage. In addition to the inspections shown in Table 4.5-1, there were those performed by the equipment supplier to confirm the adequacy of material received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME Code requirements. The inspection procedures and acceptance standards required on original pipe materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements, and were equivalent to those performed on ASME coded vessels. The loop 32 RCS hot leg elbow replaced in conjunction with the steam generator was fabricated and inspected to ASME Code Section III requirements.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified Westinghouse engineers. These procedures were developed to provide the highest assurance of quality material and fabrication. They considered not only the size of possible flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabrication processes received a 100% surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed.

All reactor coolant plate material was subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. All forgings received the same inspection. In addition, 100% of the material volume was covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse Quality Control engineers monitored the supplier's work and witnessed key inspections not only in the supplier's shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, review of radiographs, performance of required test and qualification of supplier personnel. An independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, amongst other things, the original Reactor Coolant System components was carried out by the United States Testing Company for Consolidated Edison. Comparable independent surveillance was also carried out during fabrication and installation of the replacement steam generators.

Equipment specifications for fabrication required that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they were reviewed by qualified Westinghouse engineers. This, also, was done on the field fabrication procedures to assure that installation welds were of equal quality.

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Consolidated Edison engineers witnessed the hydrostatic test of the reactor vessel.

Field erection and field welding of the Reactor Coolant System during original plant construction were performed so as to permit exact fitting of the 31" ID closure pipe subassemblies between the steam generator and the reactor coolant pump. After installation of the pump casing and the steam generator, measurements were taken of the pipe length required to close the loop. Based on these measurements, the 31" ID closure pipe subassembly was properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle. During replacement steam generator fabrication and installation, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed in order to ensure restoration of the RCS to its original configuration.

Cleaning of RCS piping and equipment was accomplished before and/or during erection of various equipment. Stainless steel piping was cleaned in sections as specific portions of the systems were erected. Pipe and units large enough to permit entry by personnel were cleaned by locally applying approved solvents (Stoddard solvent, acetone and alcohol) and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove all trapped foreign particles.

Section III of the ASME B&PV Code required that nozzles carrying significant external loads be attached to the shell by full penetration welds. This requirement was satisfied for the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The Reactor Coolant System components were welded under procedures which required the use of both preheat and postheat. Preheat requirements, non-mandatory under Code rules, were performed on all weldments, including "P1" and "P3" materials which were the materials of construction in the reactor vessel, pressurizer and steam generators. Both preheat and postheat of weldments served a common purpose: the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

Quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used in the manufacture of the reactor vessel which conformed to Section III of the ASME Boiler and Pressure Vessel Code.

The piping was designed to the USAS B31.1 (1955) Code for Power Piping using the allowable stresses found in Nuclear Code Cases N-7 and N-10 for pipe and fittings, respectively. Results of piping reanalysis for seismic performance are presented in Section 16.3.5.

While the governing code for design, fabrication, inspection and testing of original RCS piping was USAS B31.1 (1955), the quality assurance requirements imposed by Westinghouse in the purchase and examination of the reactor coolant piping assured that the quality level of the plant is comparable to that delineated by USAS B31.7, Class I, Code for Nuclear Piping. This is demonstrated by the following comparison of original RCS quality assurance measures to selected provisions of USAS B31.7. The RCS reconnection

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activities associated with the replacement of steam generators during the cycle 6/7 refueling outage were governed by ANSI B31.1-1986 requirements but also met or exceeded the original RCS quality assurance measures, including the measures described below where applicable.

a) All materials conformed to ASTM specifications listed for B31.7 Class I, Nuclear Piping. In addition, all materials were certified, identified, and marked to facilitate traceability thus complying with the requirements of USAS B31.7, Class I, Code for Nuclear Piping

b) Piping base materials were examined by quality assurance methods having acceptance criteria which met the requirements set forth in USAS B31.7, Class I, Code for Nuclear Piping

c) All welding procedures, welding, and welding operators were qualified to the requirements of ASME Section IX, Welding Qualifications, which was in compliance with the requirements of USAS B31.7, Class I, Code for Nuclear Piping

d) All welds were examined by NDT methods and to the extent prescribed in USAS B31.7 for Class I, Nuclear Piping

e) All branch connection nozzle welds of nominal sizes of 3" and larger were 100% radiographed. This exceeded the requirements of USAS B31.7, Class I piping since it included nominal sizes of 6" and larger for 100% radiography

f) All finished welds were liquid penetrant examined on both the outside and inside (if accessible) surfaces as required by USAS B31.7, Class I. In addition, nozzle welds in nominal sizes 2" and smaller were progressively examined after each ¼ inch increment of weld deposit in lieu of radiography

g) Hydrostatic testing was performed on the erected and installed piping. This requirement was the same as in USAS B31.7, Class I.

Hence, the Westinghouse quality assurance requirements implemented in the procurement of Indian Point 3 piping and fittings were equal to and in some instances exceeded the requirements of USAS B31.7.

The design and stress criteria specified in USAS B31.7 are not directly comparable to that of USAS B31.1 (1955 for piping design and 1967 for piping stress qualification). The following described how USAS B31.1 (1955 and 1967) were used in the primary coolant piping and the ASME B&PV Code Section III, Subsection NB, 1986 Edition for the pressurizer surge line including the effects of thermal stratification in the Indian Point Unit 3 design. A thermal expansion flexibility analysis was performed on the main primary coolant piping and pressurizer surge line (including the effects of thermal stratification) in accordance with the criteria set forth in USAS B31.1 (1955 and 1967) for the reactor coolant piping and the ASME B&PV Code Section III 1986 Edition for the pressurizer surge line including the effects of thermal stratification. For the reactor coolant piping the analysis was performed to ensure that the stress range and number of thermal cycles (usage factor) are safely within the limits prescribed in B31.1. As per the requirements of USAS B31.1, no fatigue analysis is required and hence, no fatigue analysis of the reactor coolant piping is performed. For the pressurizer surge line including the effects of thermal stratification, the analysis was performed to ensure that the stress range and number of thermal cycles (usage factor) are

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safely within the limits prescribed in ASME B&PV Code Section III, Subsection NB, 1986 Edition. In addition, seismic analyses were performed on the composite piping, which included the combined effects of all the sustained (pressure and weight) loading plus seismic vertical/horizontal loading components. The resultant reactions of the piping due to separate and combined effects of thermal, sustained and seismic loading were factored into the checking of the final design of the equipment nozzles with which the piping is interconnected. In turn, the equipment supporting structures were checked for adequate design including the added effects of these same loadings. Thus the total design analysis including pipe, equipment and structures considered the effects of thermal expansion, sustained and seismic loadings with a normal usage factor.

For considering and protecting against the dynamic effects of postulated ruptures, the Reactor Coolant Loop (RCL) LOCA analysis is performed for postulated breaks in the following branch lines:

- The Surge and the Residual Heat Removal (RHR) lines ion the hot leg,
- and the Accumulator line in the cold leg.

The RCL is also evaluated for the secondary side breaks at the main steam line and feedwater line terminal end nozzle locations at the steam generator.

Thermally induced stresses arising from temperature gradients were limited to a safe and low order of magnitude in assigning a maximum permissible time rate of temperature change on plant heat up, cool down, and incremental loadings in the plant operation procedure.

An added margin of conservatism was obtained through the use of thermal sleeves in nozzles wherein a cold fluid is introduced into a pipe conveying a significantly hotter fluid or vice versa. Typical examples are the charging line, pressurizer surge, and residual heat return nozzle connections to the primary coolant loop piping.

The use of thermal sleeves was not a specific requirement in B31.7. The seismic reanalysis effort for the Reactor Coolant System piping is described in Section 16.3.5.

Shop and field fabrication requirements, documentation, and quality assurance examinations all complied with those found in USAS B31.7 for Class I Nuclear Piping.

Electroslag Welding

The 90° elbows were electroslag welded. The following were performed for quality assurance of these components:

1) The electroslag welding procedure employing "one-wire" technique was qualified in accordance with the requirements of ASME B&PV Code, Section IX, and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5 inch thick weldment and successfully tested:

   a) 6 Transverse Tensile Bars – as welded

   b) 6 transverse Tensile Bars – 2050 F, H_20 Quench

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c) 6 Transverse Tensile Bars – 2050 F, H_20 Quench + 750 F stress relief heat treatment

d) 6 Transverse Tensile Bars – 2050 F, H_20 Quench, tested at 650 F

e) 12 Guided Side Bend Test Bars

2) The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM E-186, Severity level 2 (except no category D or E defectiveness was permitted) and ASME Section III, Paragraph N-627, respectively.

3) The edges of the electroslag weld preparation were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Section III, Paragraph N-627.

4) The completed electroslag weld surfaces were ground flush with the casting surface. The, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Section III, Paragraph N-627.

5) Weld metal and base metal chemical and physical analysis were determined and certified.

6) Heat treatment furnace charts were recorded and certified.

Two of the Indian Point 3 reactor coolant pump casings were electroslag welded. The efforts discussed below were performed for quality assurance of the components:

1) The electroslag welding procedure employing "two- and three-wire" techniques was qualified in accordance with the requirements of the ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as required by Westinghouse. The following test specimens were removed from an 8 inch thick and from a 12 inch thick weldment and successfully tested for both the "2-wire" and "3-wire" techniques, respectively:

a) Two wire electroslag process – 8" thick weldment

1. 6 Transverse Tensile Bars – 750 F postweld stress relief

2. 12 Guide Side Bend Test Bars

b) Three wire electroslag process – 12" thick weldment

1. 6 Transverse Tensile Bars – 750 F postweld stress relief

2. 17 Guided Side Bend Test Bars

3. 21 Charpy Vee Notch Specimens

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4. Full section macroexamination of weld and heat affected zone

5. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions

6. Hardness survey across weld and heat affected zone.

2) A separate weld test was made using the "2-wire" electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8 inch thick weldment in the stop-restart-repaired region and successfully tested:

   a) 2 Transverse Tensile Bars – as welded

   b) 4 Guided Side Bend Test Bars

   c) Full section macroexamination of weld and heat affected zone.

3) All of the weld test blocks in 1) and 2) above were radiographed using a 24 MeV Betatron. The radiographic quality level obtained was between one-half of 1% to 1%. There were no discontinuities evident in any of the electroslag welds.

   a) The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 except no category D or E defectiveness was permitted for section thickness up to 4½ inches and ASTM E-280 severity level 2 for section thicknesses greater than 4½ inches. The penetrant acceptance standards were ASME B&PV Code Section III, Paragraph N-627.

   b) The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME B&PV Code Section III, Paragraph N-627.

   c) The completed electroslag weld surfaces were ground flush with the casing surface. Then the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME B&PV Code Section III, Paragraph N-627.

   d) Weld metal and base metal chemical and physical analyses were determined and certified.

   e) Heat Treatment furnace charts were recorded and certified.

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The two remaining Indian Point 3 reactor coolant pump casings were submerged arc welded. Quality Assurance procedures and Quality Assurance inspections equivalent to the above were also exercised on these casings.

4.5.4 Non-Destructive Testing

Section XI of the ASME Boiler and Pressure Vessel Code sets the requirements for both the pre-operational and operational non-destructive testing of nuclear reactor coolant system.

The plant was examined to the fullest extent practical in accordance with Section XI, IS-141 and IS-142, even though this plant was ordered and designed before the code was effective. Examinations nonetheless followed coed requirements wherever the design of the plant allowed.

Non-destructive testing was performed by one of several methods, as specified in Section XI and its applicable reference:

1) Visual Examination

   a) Direct Visual

   b) Remote Visual

   c) Indirect Visual

2) Surface Examination

   a) Magnetic Particle

   b) Liquid Penetrant

3) Volumetric Examination

   a) Radiographic

   b) Ultrasonic

Test personnel were qualified in accordance with all code requirements.

Pre-Service Inspection

Section XI, IS-232 required pre-operational examination of essentially 100% of the pressure containing welds within the reactor coolant system boundary.

The plant components were examined in accordance with the requirements wherever it was possible and practical to do so in order to provide base line date for subsequent inservice inspections.

The pre-service examination for the original plant components was performed at the plant site after the components had been installed. With the exception of the reactor coolant pipe to channel head weld which received a pre-service examination after the replacement steam generators were installed, pre-service examination of replacement steam generator

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pressure boundary welds was performed at the manufacturer's shop. Primary and secondary side hydrostatic tests were performed after installation. Personnel qualifications, equipment and records met the requirements of applicable codes. Onsite examinations were necessarily limited by the design and accessibility restrictions of the plant.

In-Service Inspection

Operational examinations as set forth in ASME Section XI are performed to the fullest extent practical at the required intervals.

The structural integrity of the Reactor Coolant System is maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence resulting from the inspections required by the ISI Program and indicating that potential defect implications have initiated or enlarged are investigated, including evaluation of comparable areas of the Reactor Coolant System.

Non-destructive test methods, personnel, equipment and records conform to the requirements of ASME Section XI.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several features were incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques as they become available:

1) Shop ultrasonic examinations were performed on all thermally clad surfaces to an acceptance and repair standard which assures an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed was ¼" x ¾"

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction

3) To establish baselines for Post-Operational Ultrasonic Testing of the Reactor Vessel, during the manufacturing stage selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate the in-service inspection program.

   The areas selected for ultrasonic testing mapping included:

   a) Vessel flange radius, including the vessel flange to upper shell weld

   b) Middle shell course

   c) Lower shell course above the radial core supports

   d) Exterior surface of the closure head form the flange knuckle to the cooling shroud

IPEC00035404

e) Nozzle to upper shell weld

f) Middle shell to lower shell weld

g) Upper shell to middle shell weld.

The pre-operational ultrasonic testing of these areas was performed after hydrostatic testing of the reactor vessel.

A qualified inspector employed by an insurance company authorized to write boiler and pressure vessel insurance certified all examinations.

Means of access to the Reactor Coolant Pressure Boundary were provided as necessary for the surveillance programs as detailed in the ISI Program. This inspection program is in compliance with Section XI of the ASME Code for in-service inspection of nuclear reactor coolant systems.

During the design phase, careful consideration was given to provide access for both visual and non-destructive in-service inspections of the reactor coolant primary and associated auxiliary systems and components within the boundaries established in accordance with the Section XI Code.

Specific provisions made for inspection access in the design of the reactor vessel, system layout and other major primary coolant components were:

1) All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided

2) The reactor vessel shell in the core areas was designed with a clean, uncluttered cylindrical inside surface to permit future positioning of test equipment without obstruction

3) The reactor vessel cladding was improved in finish by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with the Code

4) The cladding to base metal interface was ultrasonically examined to assure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface

5) The reactor closure head is stored in a dry condition on the operating deck during refueling, allowing direct access for inspection

6) All reactor vessel studs, nuts, and washers are removed to dry storage during refueling, allowing inspection in parallel with refueling operations

7) Access holes were provided in the core barrel flange, allowing access for the remote visual examination of the clad surface of the vessel without removal of the lower internals assembly

IPEC00035405

8) Removable plugs were provided in the primary shield, providing limited access for inspection of the primary nozzle safe-end welds

9) Manways were provided in the steam generator channel head to provide access for internal inspection

10) A manway was provided in the pressurizer top head to allow access for internal inspection

11) The insulation covering all component and piping welds 6 inches in diameter and larger and covering the adjacent base metal was designed for ease of removal and replacement in areas where external inspection is planned

12) Removable plugs were provided in the primary shield concrete above the main coolant pumps to permit removal of the pump motor and to provide internal inspection access to the pumps.

The Indian Point 3 reactor vessel was built to the 1965 edition of the ASME Code Section III and all addenda through the Summer 1965 issue. ASME Section XI in-service inspection was not a requirement at that time. However, accessibility and techniques are available for inspecting all welds requiring inspection by ASME Section XI on the vessel, except for the closure head dome and bottom head dome circumferential welds and the control rod mechanism housing and bottom instrumentation tube attachment welds which were completed prior to the issuance of Section XI. Particular design improvements applied to the reactor vessel to facilitate in-service inspection include Items 1) through 4) and 7) above.

The data and results of the pre-operational examination serve as baseline data for the in-service inspection program.

In-service inspection of seismic Class I pressure retaining components, such as vessels, piping, etc. within the Reactor Coolant Pressure Boundary is performed in accordance with Section XI of the ASME Code, as described in the IP3 Inservice Inspection Program Plan for the applicable interval, with certain exceptions whenever specific relief is granted by the NRC.

The engineered safety features, the reactor shutdown systems, the cooling water systems, and the radioactive waste treatment systems which are necessary for plant operation are provided as redundant systems. This redundancy provides the capability for system and/or component outage (per Technical Specification requirements) to perform operability tests/checks or repair/maintenance. Periodic testing is in accordance with Technical Specification requirements and the IP3 Pump and Valve Testing Plan. The Pump and Valve Testing Plan is in accordance with ASME Section XI with certain exceptions whenever specific relief is granted by the NRC.

Pre-operational Vibration Test Program

During hot functional testing, the piping was observed and any vibration problems were eliminated. Also, any other piping vibrations that were deemed excessive were eliminated. Observation for piping vibration was made by persons experienced in piping design when systems were operated in normal modes during hot functional testing. When piping vibrations were observed, and evaluation was made to determine corrective action.

IPEC00035406

Class I (seismic) systems were checked out and run prior to hot functional testing in accordance with Section 13.1.

During the normal course of the pre-operational test program, specific attention was directed to evaluating possible vibration problems during the performance of the following transients:

PRE-OPERATIONAL TEST	SPECIFIC TRANSIENTS
1. Reactor Coolant System Heatup	Operational Test of Charging Pumps (Step Changes) Reactor Coolant Pump Start Operation of Pressurizer Power-Operated Relief Valves Operation of Pressurizer Spray Valves Operation of Letdown Isolation Valves
2. Reactor Coolant System at Temperature	Operation of Pressurizer Power-Operated Relief Valve Reactor Coolant Pumps (Stopping and Starting)
3. Reactor Coolant System Cooldown	Initiation of Residual Heat Removal
4. Emergency Core Cooling Full Flow Test	Initiation and Termination of the Following: A. Safety Injection Pumps B. Residual Heat Removal Pumps
5. Chemical and Volume Control System Test	Operational Test of Positive Displacement Charging Pumps (Stop and Start)

Amplitudes of vibration will theoretically cause the pipe to reach its elastic limit. Charts or monographs were provided during pre-operational testing to define these amplitudes as a function of pipe size, span and schedule as an aide for the operator and cognizant engineer to determine acceptability.

The acceptance of an observed vibration was based on operator and cognizant engineer experience. In addition, systems and components were physically examined (visually) for the following types of deficiencies which are indicative of a possible vibration problem:

1) Cracks in the grouting of equipment foundations

2) Leaking gaskets in piping systems and pump seals

3) Leaks from flanged connections in piping systems

4) Metal to metal contact indications on piping systems restraints.

IPEC00035407

If the above types of indications were observed, further investigation was performed to establish and correct any adverse conditions.

IPEC00035408

TABLE 4.5-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
1. Steam Generator						
1.1 Tube Sheet						
1.1.1 Forging		yes		yes		
1.1.2 Cladding		yes(1)	yes			
1.2 Channel Head						
1.2.1 Forging		yes		yes		
1.2.2 Cladding		yes(1)	yes			
1.3 Secondary Sheet & Head						
1.3.1 Plates		yes				
1.3.2 Shell Transition Cone (forging)		yes		yes		
1.4 Tubes		yes			yes	
1.5 Nozzles (forgings)		yes		yes		
1.6 Weldments						
1.6.1 Shell, longitudinal	yes			yes		
1.6.2 Shell, circumferential	yes			yes		
1.6.3 Cladding (Channel Head-Tube Sheet joint cladding restoration)		yes(1)	yes			
1.6.4 Steam and Feedwater Nozzle to shell	yes			yes		
1.6.5 Support brackets				yes		
1.6.6 Tube to tube sheet			yes			yes
1.6.7 Instrument connections (primary and secondary)				yes		
1.6.8 Temporary attachments after removal				yes		
1.6.9 After hydrostatic test (all shell welds and Tube-sheet to channel head)				yes		
1.6.10 Nozzle safe ends (weld deposit)	yes		yes			
2. Pressurizer						
2.1 Heads						
2.1.1 Casting	yes			yes		
2.1.2 Cladding			yes			
2.2 Shell						
2.2.1 Plates		yes		yes		
2.2.2 Cladding			yes			

IPEC00035409

TABLE 4.5-1
(Cont.)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
2.3 Heaters						
2.3.1 Tubing(++++)		yes	yes			
2.3.2 Centering of element	yes					
2.4 Nozzle		yes	yes			
2.5 Weldments						
2.5.1 Shell, longitudinal	yes			yes		
2.5.2 Shell, circumferential	yes			yes		
2.5.3 Cladding			yes			
2.5.4 Nozzle Safe End (if forging)	yes		yes			
2.5.5 Nozzle Safe End (if weld deposit)			yes			
2.5.6 Instrument Connections			yes			
2.5.7 Support Skirt				yes		
2.5.8 Temporary Attachments after removal				yes		
2.5.9 All welds and cast heads after hydrostatic test				yes		
2.6 Final Assembly						
2.6.1 All accessible surfaces after hydrostatic test				yes		
3. Piping						
3.1 Fittings (Castings)	yes		yes			
3.2 Fitting (Forgings)		yes	yes			
3.3 Pipe		yes	yes			
3.4 Weldments						
3.4.1 Circumferential	yes		yes			
3.4.2 Nozzle to run pipe (no RT for nozzles less than 3 inches)	yes		yes			
3.4.3 Instrument connections			yes			
4. Pumps						
4.1 Casting	yes		yes			
4.2 Forgings		yes	yes			
4.2.1 Main Shaft		yes	yes			
4.2.2 Main Studs		yes	yes			
4.2.3 Flywheel (Rolled Plate)		yes				
4.3 Weldments						
4.3.1 Circumferential	yes		yes			
4.3.2 Instrument connections			yes			

IPEC00035410

TABLE 4.5-1
(Cont.)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
5. Reactor Vessel						
5.1 Forgings						
5.1.1 Flanges		yes		yes		
5.1.2 Studs		yes		yes		
5.1.3 Head Adapters		yes	yes			
5.1.4 Head Adapter Tube	yes	yes				
5.1.5 Instrumentation Tube		yes	yes			
5.1.6 Main Nozzles		yes		yes		
5.1.7 Nozzle Safe-Ends (If forging is employed)		yes	yes			
5.2 Plates		yes		yes		
5.3 Weldments						
5.3.1 Main Seam	yes			yes		
5.3.2 CRD Head Adapter Connection			yes			
5.3.3 Instrumentation tube connection			yes			
5.3.4 Main nozzles	yes			yes		
5.3.5 Cladding		yes(+++)	yes			
5.3.6 Nozzle Safe-Ends (If forging)		yes				
5.3.7 Nozzle Safe-Ends (If weld deposit)	yes		yes			
5.3.8 Head adapter forging to head adapter tube	yes		yes			
5.3.9 All welds after hydrotest				yes		
6. Valves						
6.1 Castings	yes		yes			
6.2 Forgings (No UT for valves two inch and smaller)		yes	yes			

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current
LT - Leak Testing (Helium)

(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-to-Base Metal
(++++) Or a UT and ET
(1) For clad defects and for bond to base metal

IPEC00035411

APPENDIX 4A

NEUTRON DOSIMETRY

1. MEASUREMENT OF INTEGRATED FAST NEUTRON FLUX

In order to obtain a correlation between fast neutron (E greater than 1.0 MeV) exposure and the changes observed in radiation induced properties in the test specimens, a number of fast neutron monitors are included as an integral part of the Reactor Vessel Surveillance Program. In particular, the surveillance capsules contain detectors employing the following reactions:

Fe^{54} (n,P) Mn 54 (Fe from test specimens)
Ni^{58} (n,P) Co^{58}
Cu^{63} (n, ~) Co^{60}
Np^{237} (n,f) CS^{137}
U^{238} (n,f) Cs^{137}

In addition, thermal neutron flux monitors, in the form of bare and cadmium shielded Co-Al wire, are included within the capsules to enable an assessment of the effects of isotopic burnup on the response of the fast neutron detectors.

The use of activation detectors such as those listed above does not yield a direct measure of the energy dependent neutron flux level at the point of interest. Rather, the activation process is a measure of the integrated effect that the time and energy dependent neutron flux on the target material. An accurate estimate of the average neutron flux level incident on the various detectors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

1. The operating history of the reactor
2. The energy response of the given detector
3. The neutron energy spectrum at the detector location
4. The physical characteristics of the detector

The procedure for the derivation of the fast neutron flux from the results of the $Fe^{54}(n,P)Mn^{54}$ reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron energy spectrum, is similar.

The Mn^{54} product of the $Fe^{54}(n,P)Mn^{54}$ reaction has a half-life of 314 days and emits gamma rays of 0.84 MeV energy which are easily detected using gamma spectrometry. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom from interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples, all of the interferences may be corrected for by the gamma spectrometric methods without any chemical separation.

IPEC00035412

The analysis of the sample requires that two procedures be completed. First, the Mn^{54} disintegration rate per unit mass of sample and the iron content of the sample must be measured as described above. Second, the neutron energy spectrum at the detector location must be calculated.

For this analysis, the DOT[1] two-dimensional multigroup discrete ordinates transport code is employed to calculate the spectral data at the location of interest. Briefly, the DOT calculations utilize a 21 group energy scheme, an S_8 order of angular quadrature, and P_3 (Reference 2) expansion of the scattering matrix to compute neutron radiation levels within the geometry of interest. The reactor geometry employed here includes a description of the radial regions internal to the primary concrete (core barrel, neutron pad, pressure vessel and water annuli) as well as the surveillance capsule and an appropriate reactor and core baffle description. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals are accounted for in the analytical approach.

Having the measured activity, sample weight, and neutron energy spectrum at the location of interest, the calculation of the threshold flux is as follows:

The induced Mn^{54} activity in the iron flux monitors may be expressed as

$$D = (No/A) f_i y \int_E \sigma(E)\phi(E)\, dE \sum_{J=1}^{N} F_J (1 - e^{-\lambda T_J}) e^{-\lambda(T - T_J)}$$

where:	D	=	Induced Mn^{54} activity	(dps/ gmFe)
	No	=	Avogadro's number	(atoms/ gm-atom)
	A	=	Atomic weight of iron	(gm/ gm-atom)
	f_i	=	Weight fraction of Fe^{54} in the detector	
	y	=	Number of product atoms produced per reaction	
	$\sigma(E)$	=	Energy dependent activation cross-section for the $Fe^{54}(n,P)Mn^{54}$ reaction	(barns)
	$\phi(E)$	=	Energy dependent neutron flux at the detector at full reactor power	(n/ cm^2-sec)
	λ |	=	Decay constant of Mn^{54}	(sec-1)
	F_J	=	Fraction of full reactor power during the Jth time interval, T_J	
	t_J	=	Length of the Jth irradiation period	(sec)
	T	=	Elapsed time between initial reactor startup and sample counting	(sec)

IPEC00035413

The parameters F_J, t_J, and T depend on the operating history of the reactor and the delay between capsule removal and sample counting.

The integral term in the above equation may be replaced by the following relation:

$$\int_E \sigma(E)\phi(E)dE = \bar{\sigma}\,\bar{\phi}E_{TH} = \overline{\phi}\ E_{TH}\sum_{E_{TH}}^{\infty}\sigma_S(E)\phi_S(E)\Big/\sum_{E_{TH}}^{\infty}\phi_S(E)$$

where: $\bar{\sigma}$ = Effective spectrum average reaction cross-section for neutrons above energy, E_{TH}

$\overline{\phi}\,E_{TH}$ = Average neutron flux above energy, E_{TH}

$\sigma_S(E)$ = Multigroup $Fe^{54}(n,P)Mn^{54}$ reaction cross-section

compatible with the DOT energy group structure

$\phi_S(E)$ = Multigroup energy spectra at the detector location

obtained

from the DOT analysis

Thus,

$$D = (No/A)f_i\,y\,\bar{\sigma}\,\bar{\phi}E_{TH}\sum_{J=1}^{n}F_J\ \left(1-e^{-\lambda T_J}\right)e^{-\lambda(T-T_J)}$$

or, solving for the threshold flux

$$\bar{\phi}E_{TH} = D/(No/A)f_i\,y\,\bar{\sigma}\ \sum_{J=1}^{n}F_J\left(1-e^{-\lambda T_J}\right)e^{-\lambda(T-T_J)}$$

The total fluence above energy E_{TH} is then given by

$$\phi_{E_{TH}} = \bar{\phi}E_{TH}\sum_{J=1}^{n}F_J T_J$$

where $\sum_{J=1}^{n}F_J T_J$ represents the total effective full power seconds of reactor

operation up to the time of capsule removal.

Because of the relatively long half-life of Mn^{54}, the fluence may be accurately calculated in this manner for irradiation periods up to about two years. Beyond this time, the calculated average flux begins to be weighted toward the later stages of irradiation and some inaccuracies may be introduced. At these longer irradiation times, therefore, more reliance must be placed on the Np^{237} and U^{238} fission detectors with their 30 year half-life product (Cs^{137}).

IPEC00035414

No burnup correction was made to the measured activities, since burnout of the Mn^{54} product is not significant until the thermal flux level is about 10^{14} n/cm^2-sec.

The error involved in the measurement of the specific activity of the detector after irradiation is estimated to be 6 5 percent.

2. CALCULATION OF INTEGRATED FAST FLUX

The energy and spatial distribution of neutron flux within the reactor geometry is obtained from the DOT(1) two dimensional Sn transport code. The radial and azimuthal distributions are obtained from an R, u computation wherein the reactor core as well as the water and steel annuli surrounding the core are modeled explicity. The axial variations are then obtained from an R, Z DOT calculation using the equivalent cylindrical core concept. The neutron flux at any point in the geometry is then given by

$$\phi(E,R,\theta,Z) = \phi(E,R,\theta) \quad F(Z)$$

Where $\phi(E,R,\theta)$ is obtained directly from the R,θ calculation and $F(Z)$ is a normalized function obtained directly from the R,Z analysis. The core power distributions used in both the R,θ and R,Z computations represent the expected average over the life of the station.

Having the calculated neutron flux distributions within the reactor geometry, the exposure of the capsule as well as the lead factor between the capsule and the vessel may be determined as follows:

The neutron flux at the surveillance capsule is given by

$$\phi c \quad = \quad \phi(E, R_c, \theta_c, Z_c)$$

and the flux at the location of peak exposure on the pressure vessel inner diameter is

$$\phi_{v-\max} \quad = \quad \phi(E, R_v, \theta_{v-\max}, Z_{v-\max})$$

The lead factor then becomes

$$LF = \quad \phi_c / \phi_{v-\max}$$

Similar expressions may be developed for points within the pressure vessel wall; and, thus, together with the surveillance program dosimetry, serve to correlate the radiation induced damage to test specimens with that of reactor vessel.

The specific activity of each of the activation monitors is determined by using established ASTM procedures.

References

IPEC00035415

1. R. G. Soltesz, et al., "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 – Two-Dimensional Discrete Ordinates Technique, " WANL-PR-(LL)-034, Aug. 1970

2. WCAP-11057, "Indian Point Unit 3 Reactor Vessel Fluence and RT-PTS Evaluations for Consideration of Life Extension," Westinghouse Electric Corp, June 1989 Rev. 1

APPENDIX 4B

EVALUATION OF REACTOR COOLANT SYSTEM AND SUPPORTS UNDER COMBINED SEISMIC AND BLOWDOWN LOADS

1. Description of reactor coolant system component support structures
2. Analysis of reactor coolant system and supports under combined loads

APPENDIX 4C

PROCEDURE FOR PLUGGING A TUBE IN A STEAM GENERATOR (DELETED)

IPEC00035416

APPENDIX 4D

SENSITIZED STAINLESS STEEL

Introduction

Westinghouse has evaluated the use of sensitized stainless steel for reactor components in pressurized water reactors. The results of this evaluation are summarized in WCAP-7477-L (Westinghouse proprietary) which covers the nature of sensitization conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse systems is presented in the report.

Sensitized stainless steel is subject to stress corrosion, and must not be exposed to certain environments which will cause cracking. Chlorides and flourides are the most important contaminants, although oxygen, low pH, elevated temperature and high stress generally must also be present to cause cracking. When subjected to environments that cause cracking, the cracks are usually intergranular in sensitized stainless steel.

The stainless steel safe ends on the reactor vessel, pressurizer, and steam generator nozzles may become somewhat sensitized during stress relief of the vessel. The Post Weld Heat Treatment (PWHT) temperatures and minimum time are consistent with ASME Section III requirements. The degree of sensitization of the safe ends varies from plant to plant, depending on the materials used and the detailed processing performed by the various vendors. For Indian Point 3, the specific design and construction practices are discussed in the following sections. The outer diameter and inner diameter safe ends of the reactor vessel were overlayed with type 312L and Inconel weld metal to eliminate the exposure of sensitized stainless steel in areas where there is limited accessibility for inservice inspection and plant maintenance. There is complete accessibility to the remaining RCS components. The pre-operational inspection of the RCS components provided assurance that there was no stress corrosion cracking of sensitized stainless steel.

All core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility of sensitization, with the exception of the core barrel itself, which required stress relief during manufacture at temperatures over 750 F. The stress relieving operation was conducted in a manner to minimize the possibility of severe sensitization, while maintaining the necessary conditions for relieving residual fabrication stresses. This consisted of heating to 1650 F to 1750 F, holding at this temperature for several hours, then cooling very slowly in the furnace. This treatment results in massive carbide precipitation at the grain boundaries, and agglomeration of the carbides, instead of the formation of detrimental continuous carbide films. Further, the long times at high temperatures cause diffusion of chromium into the grain boundary areas that were depleted in chromium by the precipitation of chromium carbides. This combination of formation of massive carbides, plus diffusion of chromium back into the depleted zone is referred to as "desensitization", and is commonly used to prevent severe sensitization of parts requiring heat treatments that otherwise would cause severe sensitization of the material. Stress tests run according to ASTM A393 were performed on core barrel material given this heat treatment, and results verified that severe sensitization is prevented. Material that does not or would not be expected to pass ASTM A393 is considered to be severely sensitized.

IPEC00035417

Severe sensitization of component parts of the Reactor Coolant Pressure Boundary was avoided by the following methods:

- Reactor Vessel - The type 316 safe ends were overlaid with Inconel and type 308L weld metal on the ID and OD after post welding heat treatment.
- Steam Generators - The safe ends were made of weld metal, type 309 and 308L, containing enough ferrite to preclude severe sensitization.
- Pressurizer - The nozzles were made of type 316 stainless steel, but the very short post welding heat treatment time used (less than 10 hours) is not expected to cause severe sensitization of type 316 which is more resistant to sensitization than type 304.

The welding processes employed for field use were shielded metal arc welding (SMAW) and gas tungsten arc welding (GTAW). Both welding processes were used individually or in combination to qualify procedures per the requirements of ASME Code Section IX. Quality controls employed included the use of qualified weld and inspection procedures as well as verification of the maximum interpass temperatures by use of Tempil-Stiks or contact pyrometers.

The effect of nitrogen addition on the corrosion resistance of stainless steel in the PWR environment is discussed in WCAP-7735, "Topical Report - Sensitized Stainless Steel in W PWR NSSS", August 1971.

Further justification that nitrogen addition does not adversely affect the corrosion resistance of sensitized austenitic stainless steel can be obtained from the following literature:

1) C. J. Smithells, Metal Reference Book, Vol. II, p. 621, Plenum Press (1967).

2) L. R. Scharistein, "Effects of Residual Elements on the General Corrosion Resistance of Austenitic Stainless Steels", Effects of Residual Elements on Properties of Austenitic Stainless Steels, ASTM, STP 418 (1967).

3) R. B. Gunia, G. R. Woodrow, "Nitrogen Improves Engineering Properties of Chromium-Nickel Stainless Steels", Journal of Metals , Volume 5, No. 2, p. 413, June (1970).

4) Jones and Laughlin data, sheet-type 304-N Stainless Steel, Jones and Laughlin Steel Corp., Stainless and Strip Division, Warren, Michigan.

All piping in Indian Point 3 was fabricated in a manner to assure that it will not be sensitized. All pipes and fittings were purchased in the sensitized (carbide solution treated) condition. Heat treatment after bending was also a carbide solution treatment at 1900 F or above. No heat treatment was permitted after welding. Welding was done with closely controlled interpass temperature, both in the shop and in the field, to assure freedom from sensitization.

Reactor Coolant System Nozzle Safe Ends

1. Reactor Vessel Primary Nozzle Safe Ends

   A. Method of Fabrication (See Figure 4D-1)

IPEC00035418

1) Wrought Stainless Steel - A-508, class 2 nozzle forging clad with 308 stainless steel welded to type 316 forging with Inconel weld metal. Attached prior to final post weld heat treatment.

2) Forging was overlayed on ID and OD with type 312L stainless and Inconel weld metal. This was performed in the shop after the final post weld heat treatment.

B. Inspection

1) Forging Safe Ends were examined by UT and PT at Combustion Engineering using Section III acceptance standards.

2) Weld overlay of the ID and OD surfaces was examined by UT and PT. The acceptance standards are shown below:

Ultrasonic Acceptance Standards

Rejectable Defect Indications:

a) Those exceeding 90% screen height and exceeding 1/2" length
b) Those exceeding 90% screen height and 1/2" or less in length if not separated by 2" from a similar indication
c) Those with range of 50% to 90% screen height and exceeding 1-1/2" in length
d) Those with range of 50% to 90% screen height and 1" to1-1/2" in length if not separated by 2" from a similar indication.

Penetrant Inspection Acceptance Standards

The following relevant indications were unacceptable:
a) Any cracks and linear indications
b) Rounded indications with dimensions greater than 3/16"
c) Four or more rounded indications in a line separated by 1/16 in or less edge-to-edge
d) Ten or more rounded indications in any six square inches of surface with the major dimension of this area not to exceed six inches with the area taken in the most unfavorable location relative to the indications being evaluated.

2. Steam Generator Primary Nozzle Safe Ends (See Figure 4D-2)

A. Method of Fabrication

308 stainless steel weld metal buttering applied to low alloy steel (SA508 Class 3 forging) nozzles prior to final post weld heat treatment. Stainless weld metal for the first layer is type 309L (modified) and for the balance is type 308L.

B. Inspection

Buttered safe ends were examined by PT and RT using ASME B&PV Code Section III acceptance standards.

3. Pressurizer (See 4D-3)

IPEC00035419

A. Method of Fabrication

Wrought stainless steel pipe or Type 316 forgings welded to carbon steel (A-216 Grade WCC Casting with 308 stainless steel cladding) nozzles with type 309 (modified) and 308L weld metal before PWHT. The surge nozzle safe end is fabricated from SA- 312 pipe, type 316 and the spray, relief, and safety nozzle safe ends from SA-182 forgings, type 316.

B. Inspection

Wrought material was examined by UT and PT using Section III acceptance standards.
Reactor Coolant System Construction

All primary piping and fittings were given a solution annealing treatment consisting of heating to 1900 - 1950 F, holding 1 hour per inch of thickness and water quenching. This assured that the material would not be sensitized.

Main coolant pipe welds are of type 308L or 316 stainless steel. Welding was performed during original plant construction by the manual metal arc process after the root pass was completed using an insert followed by three layers using the manual gas shielded tungsten arc process. The maximum energy input possible with the manual metal arc process is on the order of 20,000 joules per linear inch of weld. With the large heat sink available in this thick walled pipe (2.375 to 3.00"), and the interpass temperature control of 350 F maximum, there was no sensitization of the solution treated pipe during welding.

Comparable welding controls to avoid primary piping sensitization were also employed during steam generator replacement, however, automatic gas metal arc welding processes were used after the root and hot passes were manually completed. The use of inserts was not required during the reconnection of primary piping to the replacement steam generators.

Venting provisions were made at high points throughout the Reactor Coolant System to relieve entrapped air when the system is filled and pressurized. Principally, vents were installed on the reactor vessel head, the pressurizer, and the reactor coolant pumps. Additional vents are available on the control rod drive mechanisms, on instruments, and on a number of connecting pipes. For normal venting of the Reactor Coolant System, only the principal venting points are utilized. The amount of oxygen which could be trapped in the remaining small volumes becomes negligible as the system is pressurized and the oxygen is scavenged by the hydrazine, specifically added for this purpose prior to operation. During operation, the oxygen levels are kept low consistent with water chemistry requirements as described in the Technical Requirements Manual.

Reactor Coolant System Operational Stresses

To avoid unusual stresses in areas where nozzle safe ends are joined to the piping, precautions were taken to eliminate unnecessary stresses due to erection of the various components of the Reactor Coolant System. The primary coolant system piping closure pieces are two pipe fitting subassemblies located between the steam generator and the primary coolant pump. The 40 degree elbow of the loop piping was first installed on the steam generator outlet nozzles. Then the gap to be closed by the closure pieces was physically measured between the 40 degree elbow outlet and the inlet nozzle of the pump. These measured dimensions for each individual

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loop were compensated and adjusted for the expected field weld shrinkage. The resulting net true dimensions were then transmitted to the pipe shop fabricator who prepared the final closure pipe subassemblies for each primary coolant loop. Upon welding these specially dimensioned pipe subassemblies in place, the primary coolant system closure was accomplished for each loop in a condition which was free from cold spring. During steam generator replacement, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed to ensure restoration of the RCS to its original configuration.

As a precaution that the behavior of the Reactor Coolant System during operating conditions was as predicted, measurements were made during incremental temperature increases during the hot functional test. The measurements were made to check the movement of the components at temperature and pressure to insure interferences were not present. The data taken during the test were compared with the flexibility analysis predictions and evaluated.

Inservice Inspection Capability

As a final check on the adequacy of the precautions taken to avoid any Reactor Coolant System failure as a result of severely sensitized stainless steel, a post-operational inspection plan was developed for the nozzle safe ends within the Reactor Coolant System Boundary. The pressurizer and steam generator stainless steel safe ends which were subjected to the furnace atmosphere during final stress relief are accessible for visual, surface and volumetric inspection upon removal of the insulation at each safe end. The reactor vessel safe ends which were subjected to the furnace atmosphere are accessible for limited inspection by removal of the special access plugs provided in the primary concrete just above each nozzle. Upon removal of these plugs and the insulation of the safe end, approximately 120 degrees of the top segment of the safe ends are accessible for direct visual and surface examination.

A specially designed in-vessel, remote, ultrasonic, inservice inspection tool was developed which can be affixed to the upper vessel flange after removal of the head. This tool is intended for ultrasonic examination of the vessel circumferential and longitudinal welds, nozzle-to-vessel welds and nozzle-to-safe-end welds. Some of these examinations utilizing this invessel tool were performed in the 1979 refueling outage. No indications were revealed by this testing.

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CHAPTER 5

CONTAINMENT

5.1 CONTAINMENT SYSTEM STRUCTURES

5.1.1 Design Basis

The Reactor Containment completely encloses the entire reactor and Reactor Coolant System and ensures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the Reactor Coolant System were to occur. The liner and penetrations were designed to prevent any leakage through the containment. The structure provides biological shielding for both normal and accident situations.

The Reactor Containment was designed to safely withstand several conditions of loading and their credible combinations. The major loading conditions were:

a) Occurrence of a gross failure of the Reactor Coolant System which creates a high pressure and temperature condition within the containment.

b) Coincident failure of the Reactor Coolant System with an earthquake or wind.

5.1.1.1 Principal Design Criteria

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the basis for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed.

Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used

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shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1 of 7/11/67)

The containment system structure is of primary importance with respect to its safety function in protecting the health and safety of the public. Quality standards of material selection, design, fabrication, and inspection governing the above features conformed to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the reactor containment conformed to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is given in Section 5.1.1.5.

Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding conditions, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data of their suitability as a basis for design. (GDC 2 of 7/11/67)

All components and supporting structures of the Reactor Containment were designed so that there is no loss of function of such equipment in the event of design basis ground acceleration acting in the horizontal and vertical directions simultaneously.

The dynamic response of the structure to ground acceleration, based on the site characteristics and on the system damping, was included in the design analysis.

The Reactor containment is defined as a seismic Class I structure for purposes of seismic design (Chapter 16). Its structural members have sufficient capacity to accept without exceeding specified stress limits, a combination of normal operating loads, functional loads due to a Loss-of-Coolant Accident, and the loadings imposed by the design basis earthquake.

Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Non-combustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features. (GDC 3 of 7/11/67)

Fire protection in all areas of the nuclear electric plant is provided by structure and component design which optimizes the containment of combustible materials and maintains exposed

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combustible material below the ignition temperature. The station was designed on the basis of limiting the use of combustible materials in construction by using fire resistant materials to the greatest extent practical. The Reactor Containment System was designed to maintain its capability in case of fire to safely shut down and isolate the reactor. Since containment recirculation ventilation charcoal filters are required, special manually-actuated sprays are installed which are operable from the Control Room. Containment liner thermal insulation does not support combustion. Fire headers are provided inside Containment and the Reactor Coolant Pumps are provided with an oil collection system. For additional details on fire protection features, see Section 9.6.2.

Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5 of 7/11/67)

Records of the design, fabrication, construction and testing of the reactor containment are maintained throughout the life of the reactor.

Reactor Containment

Criterion: The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public, the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features, as may be necessary, to retain for as along as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public. (GCD 10 to 7/11/67)

The design pressure and temperature of the Containment exceed the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the Reactor Coolant system up to and including the hypothetical double-ended severance of reactor coolant pipe. Energy contribution from the steam system was included in the calculation of the containment pressure transient due to reverse heat transfer through the steam generator tubes. The supports for the Reactor Coolant System were designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible.

The containment structure and all penetrations were designed to withstand, within design limits, the combined loadings of the Design Basis Accident and design basis seismic conditions.

All piping systems which penetrate the vapor barrier are anchored at the liner. The penetrations for the main steam, feedwater, blowdown and sample lines were designed so that the penetration is stronger than the piping system and that the vapor barrier is not breached due to a hypothesized pipe rupture combined, for the case of the steam line, with the coincident internal pressure. The pipe capacity in the flexure was assumed to be limited to the plastic moment capacity based upon the ultimate strength of the pipe material. All lines, with the exception of small bore lines, 2" and smaller connected to the Primary Coolant System that penetrate the vapor barrier were also anchored at or within the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are each provided with

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at least one valve between the anchor and the Reactor Coolant System. These anchors were designed to withstand the thrust, moment and torque resulting from a hypothesized rupture of the attached pipe.

All isolation valves are supported to withstand, without impairment of valve operability, the combined loadings of the design basis accident and design basis seismic conditions.

Appendix 4B includes a discussion of the details of the design of primary system supports. In addition, the design pressure will not be exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources, such as residual heat and limited metal-water reactions, structural heat sinks and the operation of the engineered safeguards: the latter utilizing only the emergency electric power supply.

Reactor Containment Design Basis

Criterion: The reactor containment structure, including openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a Loss-of-Coolant Accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system will not result in undue risk to the health and safety of the public. (GDC 49 of 7/11/67)

The following general criteria were followed to assure conservatism in computing the required structural load capacity:

a) In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe were considered.

b) In considering post-accident pressure effects, various malfunctions of the emergency systems were evaluated. Contingent mechanical or electrical failures were assumed to disable one of the diesel generators, two of the five fan-cooler units and one of the two containment spray units. Equipment which can be run from diesel power is described in Chapter 8.

c) The pressure and temperature loadings obtained by analyzing various Loss-of-Coolant accidents, when combined with operating loads and maximum wind or seismic forces, do not exceed the load carrying capacity of the structure, its access opening or penetrations.

The most stringent case of these analyses is summarized below:

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only these engineered safety features which can run simultaneously with power from two of the three on-site diesel generators (two high head safety injection pumps, one recirculation pump, three fan cooler units, one spray pump), results in a sufficiently low radioactive materials leakage from the containment structure that there is not undue risk to the health and safety of the public.

NDTT Requirement for Containment Material

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Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50 of 7/11/67)

The selection and use of containment materials compiled with the applicable codes and standards tabulated in Section 5.1.1.5.

The concrete containment is not susceptible to a low temperature brittle fracture.

The containment liner is enclosed within the Containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during normal operation is between 50°F and 130°F. This includes both hot operating and cold shutdown conditions. The minimum service metal temperature of the containment liner is well above the NDT temperature of +30°F for the liner material. The Equipment Hatch, penetration sleeves and Personnel Lock meet the Charpy V-notch impact values for a minimum of 15 ft-lbs at -50°F. Penetration "SS" end plates were replaced in 1997 and the Charpy V-notch impact values were determined.

5.1.1.2 Supplementary Accident Criteria

Systems relied upon to operate under post-accident conditions, which are located external to the containment and were considered to be extensions of the leakage boundary.

The pressure retaining components of the containment structure were designed for the maximum potential earthquake ground motion of the site combined with the simultaneous loads of the design basis accident as follows:

1) The liner was designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads.

2) The mild steel reinforcement was generally designed to ensure that no strains greater than the strain at the guaranteed yield point occur at a cross section under the factored loads.

The pressure retaining components of containment subject to deterioration or corrosion in service were provided with appropriate protective means or devices (e.g., protective coatings).

5.1.1.3 Energy and Material Release

The design pressure is not exceeded during any subsequent long term pressure transient determined by the combined effects of heat sources such as residual heat and metal-water reactions, structural heat sinks and the operation of other engineered safety features utilizing only the emergency onsite electric power supply.

The design pressure and temperature on the containment structure are those created by the hypothetical Loss-of-Coolant Accident. The Reactor Coolant System contains approximately 512,000 lb of coolant at a weighted average enthalpy of 595 Btu/lb for a total energy of 304,000,000 Btu. In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident is the 29 inch ID section because rupture of the 31

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inch ID section requires that the blowdown go through both the 29 inch and the 27-1/2 inch ID pipes and would, therefore, result in a less severe transient.

Additional energy release was considered from the following sources:

a) Stored heat in the reactor core

b) Stored heat in the reactor vessel piping and other Reactor Coolant System components

c) Residual heat production

d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy

The following loadings were considered in the design of the containment in addition to the pressure and temperature conditions described above:

a) Structure dead load

b) Live loads

c) Equipment loads

d) Internal test pressure

e) Earthquake

f) Wind

g) Tornado

The capability of the Containment to withstand additional energy releases is discussed in Chapter 14.

5.1.1.4 Engineered Safety Features System Contributions

Five types of engineered safety features were included in the design of this facility to assure containment integrity. These systems are discussed in Chapter 6 and their effectiveness is analyzed in Chapter 14.

5.1.1.5 Codes and Classifications

The design, materials, fabrication, inspections, and proof testing of the containment vessel complies with the applicable parts of the following:

STRUCTURAL

	Code	Title
1.	ASTM A-333, Gr. 1	Specification for Seamless and Welded Steel Pipe for Low Temperature Service

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2. ASTM A-181	Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for General Service
3. ASTM A-300, C1. 1	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
Firebox A-201, Gr. B	Specification for Carbon Silicon Steel Plates of Intermediate Tensile Ranges for Fusion Welded Boilers and Other Pressure Vessels
4. ASTM A-36, Gr. C	Specification for Structural Steel
5. ASTM A-131, Gr. C	Specification for Structural Steel for Ships
6. ASTM A-240	Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
7. ASTM A-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
8. ASTM 442, Grade 60	Standard Specification for Carbon Steel Plates with Improved Transition Properties
9. ASME Boiler & Pressure Vessel Code-Section III	Nuclear Vessels
10. ASME Boiler & Pressure Vessel Code-Section VIII	Unfired Pressure Vessels
11. ASME Boiler & Pressure Vessel Code-Section IX	Welding Qualifications
12. ASTM C-33	Standard Specifications for Concrete Aggregates
13. ASTM C-150	Standard Specifications for Portland Cement
14. ASTM C-172	Method of Sampling Fresh Concrete
15. ASTM C-31	Method of Making and Curing Concrete Compression and Flexure Test Specimen in Field
16. ASTM C-39	Method of Test for Compressive Strength of Molded Concrete Cylinders

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17. ASTM C-350	Specification for Fly Ash for Use as an Admixture in Portland Cement Concrete
18. ASTM C-94	Recommended Practice for Winter Concreting
19. ASTM C-42	Methods of Securing, Preparing, and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
20. ASTM C-494	Specifications for Chemical Admixtures for Concrete
21. ASTM A-305	Specifications for Miminum Requirements for Deformation of Deformed Bars for Concrete Reinforcement
22. ASTM A-408	Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
23. ASTM A-432	Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength
24. Research Council of Reveted & Bolted Structural Joints of the Engineering Foundation	Specification for Structural Joints Using ASTM A-325 Bolts
25. ACI-613	Recommended Practice for Selecting Proportions for Concrete
26. ACI-306	Recommended Practice for Winter Concreting
27. ACI-318, Part IV-B	Structural Analysis and Proportioning of Members Ultimate Strength Design
28. ACI-318	Building Code Requirements for Reinforced Concrete
29. ACI-505	Reinforced Concrete Chimney Design
30. ACI-315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
31. ASME Nuclear Vessels Code	---
32. ASA N6.2	Safety Standards for the Design, Fabrication and Maintenance of Steel Containment

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	Structures for Stationary Nuclear Power Reactors
33. ASA A58.1	American Standard Code Requirements for Minimum Design Loads in Building and Other Structures
34. --	State Building and Construction Code for the State of New York
35. SSPC-SP-6	Commercial Blast Cleaning
36. ASME Boiler & Pressure Vessel Code-Section XI	Rules for in service inspection of Nuclear Power Plant Components.

5.1.2 Containment System Structure Design

5.1.2.1 General Description

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome. A welded steel liner with a minimum thickness of ¼ inch is attached to the inside face of the concrete to insure a high degree of leak-tightness. The design objective of the containment structure was to contain all radioactive material which might be released from the core following a Loss-of-Coolant Accident. The structure serves as both a biological shield and a pressure container.

The structure, as shown on Figure 5.1-1 and Plant Drawings 9321-F-25013, -25023, -25033, -25063, -25073, and -25083 [Formerly Figures 5.1-2 through 5.1-7], consists of side walls measuring 148 feet from the liner on the base to the springline of the dome, and has an inside diameter of 135 feet. The side walls of the cylinder and the dome are 4'-6" and 3'-6" thick, respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The flat concrete base mat is 9 feet thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3 feet of concrete, the top of which forms the floor of the Containment.

Where uplift from pressure occurs at the outer areas of the mat, the 9-ft thick mat has sufficient flexural capacity to resist the uplift until it is dissipated.

No hydraulic uplift exists since the bottom elevation of the mat is considerably higher than that of the high water level.

The large mass of the Containment including interior concrete and equipment makes the structure inherently stable from overturning due to seismic motion or tornado.

In addition, keying action from the reactor pit and sumps, plus friction between the concrete and rock, prevents sliding of the structure from horizontal ground motion.

The basic structural elements that were considered in the design of the containment structure are the base slab, side walls and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors. The

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reinforcing in the structure exhibits a total elastic response to all primary loads. The lower portion of the cylindrical liner is insulated to avoid thermal deformation of the liner under accident conditions.

The containment structure is inherently safe with regard to common hazards such as fire, flood and electrical storm. Internal structures consist of equipment supports shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are supported on the 2'-8" thick floor slab.

A 3-foot thick concrete ring wall serving as a missile and partial radiation shield surrounds the Reactor Coolant System components and supports the polar-type reactor containment crane. A 2-foot thick reinforced concrete floor covers the Reactor Coolant System with removable gratings in the floor provided for crane access to the Reactor Coolant Pumps. The four steam generators, pressurizer and various pipings penetrate the floor. Spiral and scissor stairs provide access to the areas below the floor. There is a reinforced concrete missile shield wall around the pressurizer above the operating floor. The original design is to protect the containment steel liner from postulated valve piece or instrument missiles connected to the pressurizer. Currently these missiles have been shown not to be credible.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete, with wall and shielding water providing the equivalent of 6 feet of concrete. The floor is 4-feet thick. The concrete walls and floor are lined with ¼-inch thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

A sub-surface drainage system is provided around the Containment Building where the mat is below grade as shown in Figure 5.1-11. Since the containment is above the water table, no hydrostatic seepage will occur.

The detailed structural design and analysis of the Containment System is presented in Appendix 5A. See also Sections 16.1 and 16.4 for seismic analysis.

5.1.2.2 General Design Criteria

The following loads were considered to act upon the containment structure creating stresses within the component parts:

a) Dead load consisted of the weight of the concrete wall, dome, liner insulation, base slab and the internal concrete.

b) Live load consisted of snow and construction loads on the dome and major components of equipment in the containment. Snow and ice loads were assumed to be applied uniformly to the top surface of the dome. A construction live load of 50 pounds per square foot was used on the dome, but was not considered to act concurrently with the snow load. Equipment loads were considered as specified on the drawings supplied by the manufacturers of the various pieces of equipment.

c) The internal pressure transient used for the containment design and its variation with time is shown on the pressure-temperature transient curve, Figure 5.1-8. For the free volume of 2,610,000 cubic feet within the containment, the design pressure is 47

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psig. This pressure transient is more severe than those calculated for various Loss-of-Coolant Accidents which are presented in Chapter 14.

d) Thermal expansion stresses due to internal temperature increase caused by a Loss-of-Coolant Accident were considered. This temperature and its variation with time are shown on the pressure-temperature transient curve, Figure 5.1-8. The maximum temperature at the uninsulated section of the liner under accident conditions is 247°F. For the 1.25 times and 1.50 times design pressure loading conditions, the corresponding liner temperatures will be 285°F and 306°F respectively. The pressure temperature transient curves for these loading conditions are shown in Figures 5.1-9 and 5.1-10, respectively. The maximum operating temperature is 130°F. The design 24-hour mean-low ambient temperature is -5°F.

e) The ground acceleration for the operational basis earthquake was determined to be 0.1g applied horizontally and 0.05g applied vertically. These values were resolved as conservative numbers based upon recommendations from Dr. Lynch, then Director of Seismic Observatory, Fordham University.

   A dynamic analysis was used to arrive at equivalent design loads. Additionally, a design basis earthquake ground acceleration of 0.15 horizontal and 0.10 vertical was used to analyze for the no-loss of function. This is discussed in Section 5.1.3.5, Seismic Design Summary.

f) The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designated the site as being in a 25 psf zone. In this code, for height zones between 100 and 499 feet, the recommended wind pressure on a flat surface was 40 psf. Correcting for the shape of the containment by using a shape factor of 0.60, the recommended pressure becomes 24 psf. The State Building and Construction Code for the State of New York stipulated a wind pressure up to 30 psf on a flat surface for heights up to 600 feet. For design, a 30 psf basic wind load was used from ground level up.

g) Internal pressure was applied to test the structural integrity of the vessel up to 115 percent of the design pressure. For this structure, the test pressure was 54 psig.

h) Tornado loads consisted of 300 mph tangential wind traveling with a forward velocity of 60 mph. Also considered as a separate and as a combined loading combination was a 3.0 psi pressure drop external to the structure. In addition, horizontal and vertical missile loads were considered as specified in Section 2.1.5 of Appendix 5A.

5.1.2.3 Material Specifications

Basically, four materials were used for the construction of the containment vessel. These are:

a) Concrete
b) Reinforcing Steel
c) Plate Steel Liner
d) Insulation

Details of material properties, fabrication and erection requirements and material test results are presented in Appendix 5A, Section 5.0. Basic specifications for these materials were as follows:

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a) Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement and water. Aggregates conformed to American Society for Testing Materials Specification C-33 "Standard Specification for Concrete Aggregates." Aggregates consisted of inert materials that were clean, hard, durable, free from organic matter and uncoated with clay or dirt. Fine aggregate consisted of natural sand and the coarse aggregate of crushed stone. Fine aggregate tests performed include gradation, fineness modulus, specific gravity, unit weight, organic impurities, soundness (5 cycles Na_2SO_4) silt content and structural strength of sand in relation to Ottawa sand.

Coarse aggregate tests included gradation, fineness modulus, specific gravity, unit weight, soundness (5 cycles Na_2SO_4), soft particles and Los Angeles abrasion test.

Portland cement conformed to American Society for Testing and Material Specification C-150-65. Manufacturer's mill test reports were required covering each silo of cement drawn for the project.

Water was free from any injurious amounts of chloride, acid, alkali, salts, oil, sediment or organic matter, and fit for drinking.

A testing laboratory tested materials for concrete, designed mixes, tested check mix at batch plant and jobsite, and tested job concrete test cylinders. Design mixes were checked by the laboratory, including adjustments to obtain a workable mix based on specification requirements, and verified by trial batches and laboratory test.

The only admixtures in the concrete were PLACEWELL and AIRECON, both products of Union Carbide. PLACEWELL is a liquid, water-reducing admixture and AIRECON is a liquid air-entraining admixture, both of which enhance the properties of plastic and hardened concrete. PLACEWELL conformed to all the requirements of ASTM C494 for a Type A water reducing admixture and contained no calcium chloride. AIRECON conformed to the requirements of ASTM C260 for Air Entraining Admixtures and also contained no calcium chloride. The mixing of concrete was done with a batch mixer of approved AGC type, or in ready-mix equipment conforming to ASTM Specification C94.

b) Reinforcing steel for the dome, cylindrical walls and base mat was high-strength deformed billet steel bars conforming to ASTM Designation A432-65 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength" (Revised ASTM A615-68, Grade 60). This steel had a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7 percent in an 8-inch specimen. Reinforcing bars No. 11 and smaller in diameter were lapped spliced in the mat for flexural loadings and spliced by the Cadweld process in the walls and dome for tension loading. Bars No. 14S and 18S were spliced by the Cadweld process only. A certification of physical properties and chemical content of each heat of reinforcing steel delivered to the job site was issued from the steel supplier. The splices used to join reinforcing bars were sample tested to assure that they will develop at least 125% of the minimum yield point stress of the bar. The test program required cutting out, at random, approximately 2 percent of the completed splices and testing to determine their breaking strength, thus confirming the strength of both the bars and the splice.

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In the Containment, vertical rebar splices were staggered a minimum of 1'-2". Seismic Diagonal Bar splices were staggered 1'-2" vertically in each direction. In the dome a 2'-0" stagger pattern was used throughout for the Cadweld splices as well as the reinforcing splice plates, except for final closure pieces at the apex of the dome. Horizontal rebar splices were splices were spliced in elevation and in cross-section (bars or bar pairs) with 2'-4" nominal and 2'-0" minimum stagger.

The above requirements were generally satisfied during construction except in special cases where physical or layout problems occurred in isolated areas in the containment.

For all other seismic Class I structures other than the Containment, rebars were lap spliced in accordance with the requirements of ACI-318-63 "Building Code Requirements for Reinforced Concrete." No specific stagger requirements were specified. In the Containment, mechanical splices were included because of the biaxial tensile stress conditions in the concrete, which eliminate bond and require continuous rebar, and the ACI-318 requirement that lapped splices in tension cannot be used for bars greater than No. 11.

c) The plate steel liner is carbon steel conforming to ASTM Designation A442-65 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel had a minimum yield strength of 32,000 psi and minimum tensile strength of 60,000 psi with an elongation of 22 percent in an 8-inch gauge length at failure.

   The liner is ¼-inch thick at the bottom, ½-inch thick in the first three courses and 3/8-inch thick for remaining portion of cylindrical walls except ¾-inch thick at penetrations and ½-inch thick in the dome. The liner material has been tested to assure an NDT temperature more than 30 F lower than the minimum operating temperature of the liner material.

   Impact testing was done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code. A 100 percent visual inspection of the liner anchors was made prior to pouring concrete.

d) The material for insulating the liner plate is urethane foam covered with gypsum board and a stainless steel jacket and backed with asbestos paper on the unexposed side. This insulation was selected to withstand the calculated temperature and pressure conditions associated with Figures 5.1-8, 5.1-9, 5.1-10.

e) Quality of both materials and construction of the containment vessel was assured by a continuous program of quality control and inspection. These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC. The Quality Assurance Program (described in Entergy's Quality Assurance Program Manual) covers modification and maintenance activities.

5.1.2.4 <u>Structural Design Criteria</u>

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The design was based upon limiting load factors which were used as the ratio by which loads will be multiplied for design purposes to assure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior.

The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permitted the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure, therefore, this approach placed minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are presented in Appendix 5A, Section 2.1.0.

The load factors utilized in this design were based upon the load factor concept employed in Part IV-B, "Structural Analysis and Proportioning of Members Ultimate Strength Design" of ACI 318-63. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions. Specific combined load equations used in design are presented in Section 2.1.14 of Appendix 5A. The load factors were chosen in a conservative manner to insure that the structure as designed would respond to design loads with elastic strain behavior.

The primary load in the containment structure analysis is the accident pressure load. Since the formulas for analysis were based on thin shell considerations, resulting in tensile loads resisted by reinforcing, the same results would be obtained regardless of whether a load factor (ultimate) or reduced allowable stress (working) approach was used in design.

The load factors utilized plus the conservative assumptions used in analysis insured that the design for this portion of the structure was conservative. Conservative analysis assumptions included use of full concrete section for determining flexural rigidity thus drawing moment and shear to the stiffer base and use of only hoop steel as the spring constant which controls cylinder growth in the unrestrained area above the discontinuity.

Earthquake and wind loads were based on analysis with the Containment modeled as a cantilever beam. The loads are resisted by tension in rebar thus the same results would be obtained by working strength or ultimate strength design. Secondary, thermal loads were also carried by tension in the rebar and are thus independent of ultimate or working strength design.

Equilibrium checks can be easily made for the containment shell since thin shell membrane analysis was used in the design and only the rebar was assumed to carry the load. In addition, overturning moment for earthquake and tension caused by temperature were assumed carried by rebar only and thus easily checked for equilibrium.

In addition to the above analysis, non-membrane portions of the cylinder such as the Equipment Hatch opening were analyzed by a Finite Element Computer analysis. The results were checked to insure that internal stresses times resisting rebar area gave resultant forces equal to the applied external forces.

In areas of the Containment where tensile stresses in more than one direction occur, all stresses are carried by continuous mechanically spliced rebar. Therefore, stress limits of ACI-318-63 were applicable since concrete strength and rebar bond provided by concrete were not

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considered in design. The f'_c dependent factors were specified in the ACI-318-63 Code used only in the design of the base mat where a uni-axial stress condition exists and in the base of the Containment cylinder wall where hoop stresses are minimal. In the cylinder, radial shear forces are resisted only by rebar. Seismic shear forces are also resisted by rebar with no account taken of resistance offered by concrete.

All structural components were designed to have a capacity required by the most severe loading combination.

Thus, the design included the consideration of both primary and secondary stresses, and the load capacity in structural members was based on the ultimate strength values presented in Part IVB of ACI-318, as reduced by the capacity reduction factor "ϕ" which provided for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under capacity. For tension members, the factor "ϕ" was established as 0.95. The factor "ϕ" was 0.90 for flexure and 0.85 for diagonal tension, bond and anchorage.

For the liner steel the factor "ϕ" was 0.95 for tension. For compression and shear, the primary membrane liner stress was maintained below 0.95 yield and elastic stability was assured as a function of liner anchorage requirements.

The liner was designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads except in regions of local stress concentrations or stresses due to secondary load effects, in which case the liner strain was limited to 0.5 percent. Sufficient anchorage was provided to assure elastic stability of the liner. The basic design concept utilized stud anchorage of the liner plate to the concrete structure which assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. See References 1 and 2. The studs in the ½ inch plate were installed on 24" horizontal and 28" vertical grid and in the 3/8-inch plate on a 24" horizontal and 14" vertical grid. The design considered the possibility of daily stress reversals due to ambient temperature changes for the life of the plant, and fatigue limit of the studs exceeds the design requirements.

5.1.2.5 Missile Protection

High pressure Reactor Coolant System equipment is surrounded by the 3'-0" concrete shield wall enclosing the reactor coolant loop and pressurizer and by the 2'-0" concrete operating floor.

A structure is provided over the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

Systems containing hot pressurized fluids that might affect the engineered safeguards components were carefully checked against the possibility of being sources of missiles. The general criterion adopted was to make provision, when necessary, against the generation of missiles rather than allow missile formation and try to contain their effects.

Once the design requirement that the above systems were not to be sources of missiles had been set forth, identification of potential deficiencies and generation of adequate fixes took place through the quality assurance program.

The following examples illustrate how this approach was implemented:

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Valves

Valves installed in the Nuclear Steam Supply System were evaluated for the probability of their stem becoming missiles. Valve stems are not considered credible missile since at least one feature (in addition to the steam threads) is included in their design that will prevent stem ejection. Valve steams with backseats are prevented from becoming missiles by the backseat feature. Also, valve plugs are secured and locked to the valve stems to prevent loosening in service. In addition, valve stems of valves with power actuators, such as air-operated or motor-operated valves are effectively restrained by the valve actuator. Valve stems of rotary motion valves, such as plug valves, ball valves, and butterfly valves, as well as diaphragm-type valves are not considered as credible missiles. This is because these valves do not have a large reservoir of pressurized fluid acting on the valve stem; therefore there is insufficient stored energy available to produce a missile.

Valves with nominal diameter larger than 2" were designed against bonnet-body connection failure and subsequent bonnet ejection by means of: (a) using the design practice of ASME Section VIII, ASME Section III and USAS B16.5 and (b) by controlling the load during the bonnet body connection stud tightening process.

Stud and nut material is ASTM A193-B7 and A194-2H, or approved equal. The proper stud torquing procedures and/or the use of a torque wrench with indication of the applied torque, control the stress of the stud to acceptable limits established by industry standards. The complete valves were hydro tested per USAS B16.5 (1500 lbs. USAS valves are hydro to 5400 psi). The cast stainless steel bodies and bonnets were radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of 2" or smaller were forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads were designed to withstand the hydrostatic end forces. The pressure containing parts were designed per criteria established by USAS B16.5 specification.

Valves with nominal diameter of 2" or smaller may be supplied with a screwed bonnet and canopy seal. The canopy seal is the pressure boundary while the bonnet threads were designed to withstand the hydrostatic end force. The pressure containing parts were designed per criteria established by the USAS B16.5 specification.

Reactor Coolant Pump Flywheel

The reactor coolant pump flywheel was not considered to be a credible source of missiles because of conservative design and care in manufacture and inspection. The flywheel material is ASTM A-533 having an NDTT less than 10 F. The design results in a primary stress less than 50% of the material yield strength at operating speed. The flywheel was subjected to 100% volumetric ultrasonic inspections which are repeated at intervals during plant life. The finished machined bore was subjected to examination by approved method. The design overspeed of the pump is 125%. The maximum pump overspeed on loss of external load is 112%.

5.1.2.6 Protection From Long-Term Corrosion

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Steel members embedded in reinforced concrete structures are protected from corrosion by concrete in accordance with normal code requirements and hence are not exposed to the atmosphere.

All other steel appurtenances are not main structural load carrying members and are visible and accessible for regular maintenance. These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC. They are generally shop painted with red lead and finish painted with a standard finish paint.

5.1.3 Stress Analysis

5.1.3.1 General

The structural design of the Containment met the requirements established by 1961 edition of "The State Building and Construction Code for the State of New York" so far as these provisions were applicable. All concrete structures were designed, detailed and constructed in accordance with the provisions of "Building Code Requirements for Reinforced Concrete" (ACI-318-63) so far as these provisions were applicable. A detailed description of containment component design is presented in Appendix 5A, Section 4.0.

5.1.3.2 Method of Analysis

Basically three separate structural components were analyzed, each in equilibrium with loads applied to it and with constraints occurring at the juncture of the structures. The three components were:

a) The 135-ft ID hemispherical dome
b) The 135-Ft ID Cylinder
c) The base slab.

Mathematically, the dome and cylinder were treated as thin-walled shell structures, which resulted in a membrane analysis. Since the thickness of the dome and cylinder is small in comparison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridional curves, the stresses due to pressure, tornado wind or earthquake were calculated by assuming that they are uniformly distributed across the shell thickness.

Since the concrete was not assumed to resist any tensile or shear forces, radial shear reinforcing was introduced in the lower portion of the wall in the form of hooked diagonal stirrups and diagonally bent bars as shown in Figure 5.1-1. Likewise, diagonal shear reinforcing in the circumferential direction was included to resist earthquake shears for the full height of the wall and a distance above the springline into the dome until a point was reached where the dome liner and meridional and hoop reinforcing can resist the total shear. The base slab was treated as a flat circular plate supported on a rigid non-yielding foundation.

5.1.3.3 Dome Analysis

The analysis of the hemispherical dome was performed by the superposition of membrane forces resulting from gravity, accident pressure and accident thermal loads. In addition, tornado, earthquake or wind loading create both direct and shear stresses in the dome and operating temperature of the liner creates tension and compression. All of the combined direct

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stresses are developed in the reinforcing steel encased in the concrete. The liner of the dome above a certain point can resist shear load and the anchorages were designed to assure composite action. The dome reinforcing was spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder was realized.

Discontinuity effects at the springline are very slight due to the small difference in radial growth between the dome and cylinder. Since the circumferential reinforcing in the dome and cylinder vary, stresses and, therefore, deformations are essentially equal.

5.1.3.4 Cylinder Analysis

The analysis of the cylinder was done by superposition of membrane forces resulting from gravity, pressure and thermal loads, over-turning due to tornado, earthquake or wind and shears due to tornado, earthquake or wind. The concrete was reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars were placed to resist the horizontal and vertical shears due to tornado, earthquake or wind. The required capacity of the diagonal bars was designed so that the horizontal component per foot of the diagonals equaled the maximum value of shear flow.

A check was made to insure that no net compressive force results in the diagonal bars because of the combination of seismic shear load and internal pressure load. Although, in the cylinder, the liner has some capacity available to resist the seismic shears, no credit was taken for this capacity.

Only in the upper area of the dome (beyond about 30 degrees above the springline), where the seismic shears are small, does the liner help to resist shear. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing was designed to accommodate all seismic shears. No credit was taken for the dowel action of the vertical and horizontal bars in resisting seismic shear. The maximum stress in the rebar beyond about 30 degrees above the springline due to an earthquake was determined by resolution of the principal tensile stress into components parallel to the rebar. This rebar provides an adequate mechanism to resist shear. (See Section 16.1).

A detailed description of the methods of analysis used for the containment concrete structures is presented in Appendix 5A, Section 4.0.

5.1.3.5 Seismic Design Summary

The design of the Containment which is a seismic Class I structure (see Chapter 16) was based on a "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design took into account the acceleration response spectrum curves developed by G. Housner. Seismic accelerations were computed as outlined in the AEC TID-7024[3] and Portland Cement Publication[4].

As indicated in Chapter 16, ground accelerations used for Operational Basis Earthquake are 0.1g horizontally and 0.05g vertically and for the Design Basis Earthquake are 0.15g horizontally and 0.10g vertically. The natural period of vibration was computed by a dynamic analysis; in this method, the containment structure was analyzed a simple cantilever, consisting of lumped masses and weightless elastic columns acting as spring restraints. Both bending and shear deformations were considered. The natural frequencies and mode shapes were computed from the equations of motion of the lumped masses. These equations were solved

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by iteration techniques by a fully tested digital computer program. Based on an uncracked concrete section, the period was determined to be 0.241 Sec.

The response of each mode of vibration to the earthquake ground motion was computed by the response spectrum technique. The participation of each mode was computed and the relative acceleration of each mass was determined using the response spectrum curves for 2% and 5% critical damping. The total response was computed as the square root sum of the squares of the individual modes.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit was taken for the reinforcing in compression.

From 30 degrees above the springline, the shear is resisted by the liner and rebar. The shear is transmitted to the liner by means of tees welded to the liner.

To facilitate construction, a retaining wall was built to carry the roadway at Elevation +95.0 to the northeast quadrant of the Containment Building. There is no backfill between the retaining wall and the Containment, consequently, there was no lateral earth pressure factored into the seismic design.

5.1.3.6 Tornado Design Summary

The design of the Containment, which is a seismic Class I structure, considered the effects resulting from tornado loads.

Tornado wind loading was taken as 300 mph tangential wind traveling with a forward velocity of 60 mph. Also considered as a separate and as a combined loading condition was a 3.0 psi pressure drop external to the structure.

The wind load was considered for three tornado conditions. One included a tangential velocity of 300 mph and a translational velocity of 60 mph. This load superposition depicts a tornado condition where the funnel coincides with the center of the Containment. Load pressure distribution patterns that resulted due to various locations of the funnel were considered. The structure was designed for a triangular and a rectangular wind distribution of 360 mph.

The above wind loading and pressure drop design criteria were consistent with the generally accepted tornado design criteria utilized on nuclear power plants in the eastern United States.

The forces from wind loadings were computed based on ASCE Paper 3269-"Transactions of the ASCE Vol. 126 Part II 1961."

The forces were converted to a shear per lineal foot around the circumference of the Containment by distributing the shear over the circumference of the seismic reinforcing.

The resulting stresses were limited to yield strength or its equivalent as defined in ACI-318, Part IV B and modified as required by the capacity reduction factor Ø.

The seismic bars provide a more than adequate mechanism to withstand the torsional effect from tornado winds, therefore, tornado winds were not a controlling factor in the design of the containment structure.

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