3.1 INTRODUCTION

The reactor is of the pressurized water type, using two primary coolant loops. The core contains 204 fuel bundles and 45 cruciform control blades. The local power distribution is controlled by enrichment zoning within the assemblies and by the use of burnable absorber pins. The overall core is configured in a multiple batch loading pattern to achieve the desired cycle length and control the power distribution.

The fuel is slightly enriched uranium dioxide in the form of ceramic pellets contained in a hermetically sealed Zircaloy or M5® rod. The fuel rods are captured in an open Zircaloy and stainless steel cage and are held laterally by spacer grids made of Inconel and/or Zircaloy. The fuel bundles are supported in the core, with lateral motion constrained, by the core support assembly. The tops of the fuel bundles are constrained laterally by the upper guide structure, which also guides the control blades into the core and the coolant flow from the core. Starting with the Batch Y reload, the fuel rod cladding, instrument tube, and end cap material was M5®. Starting with Batch DD reload, the end cap material was changed back to Zircaloy. The fuel rod cladding and instrument tube remain M5®.

Long-term reactivity control is maintained by dissolving a neutron absorber, boric acid, into the reactor coolant. The beginning of cycle boric acid concentration is reduced by using neutron absorbers mechanically fixed in the fuel bundles. These neutron absorbers may also be used to reduce the nuclear power peaking.

Short-term reactivity control is maintained by 41 of the 45 control blades. The neutron absorber used is a mixture of silver, indium and cadmium that is encapsulated in stainless steel. The control blades are actuated by rack and pinion drive mechanisms that are mounted on the reactor vessel head. The remaining four control blades have neutron absorber of reduced length. They were originally intended for axial power distribution control; but they are not allowed in the core while the reactor is critical because of the possibility of undesirable nuclear power peaking. All of the control blades except the four part-length blades drop into the core on a reactor trip.

3.2 DESIGN BASES

3.2.1 PERFORMANCE OBJECTIVES

The ultimate design of the Plant was a reactor thermal power of 2,650 MW. The primary, secondary and all safety systems were designed for this power level. However, the initial license application was for 2,200 MWt. This has since been upgraded to 2,565.4 MWt. Currently all analysis and safety evaluations of the primary and safety systems are based on a reactor power level of 2,565.4 MWt.

3.2.2 DESIGN OBJECTIVES

The reactor core, together with its control systems and the reactor protection system, is designed to function over its lifetime without exceeding fuel damage limits of excessive fuel temperature, cladding strain and cladding stress as specified in Subsection 3.2.3 during normal operating conditions and anticipated transients.

The combined response of all reactivity feed-back mechanisms to an increase in reactor thermal power at normal power conditions is a net decrease in reactivity. The combined effect of all reactivity coefficients in conjunction with the reactor control system provides stable operation. If power oscillations do occur, their magnitude will be such that the fuel damage limits are not exceeded.

Reactivity control is provided by two independent systems: the Control Rod Drive System and the Chemical and Volume Control System. The Control Rod Drive System controls short-term reactivity changes and is used for rapid shutdown. The Chemical and Volume Control System is used to compensate for long-term reactivity changes and can make the reactor subcritical without the benefit of the Control Rod Drive System. The design of the core and the Reactor Protective System prevents exceeding fuel damage limits for any single malfunction in either of the reactivity control systems.

The maximum reactivity worth of the control rods and the associated reactivity addition rate are limited by core, control rod and Control Rod Drive System design to prevent sudden large reactivity increases that could result in violation of the fuel damage limits, rupture of the reactor coolant pressure boundary, or disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

3.2.3 DESIGN LIMITS

Nuclear Limits

The design of the core meets the following nuclear limits:

1. The combined response of all reactivity coefficients to an increase in reactor thermal power yields a net decrease in reactivity at normal power conditions.

2. Control rods are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The control systems are designed to produce peak-to-average power distributions that are within the acceptable limits on overall nuclear heat flux factor and departure from nucleate boiling ratio (DNBR). The Reactor Protective System, administrative controls and the incore monitoring system ensure that these limits are not exceeded.

3. Axial xenon oscillations, should they occur, would be manually controlled by control rods using information provided by the excore detectors.

Reactivity Control Limits

The control system and operating procedures provide for adequate control of the core reactivity and power distributions such that the following limits are met:

1. Sufficient control rods are withdrawn to provide an adequate shutdown reactivity margin following a reactor trip.

2. The shutdown margin is maintained with the highest worth control rod assumed stuck in its fully withdrawn position.

3. The Chemical and Volume Control System is capable of adding boric acid to the primary coolant at a rate sufficient to maintain the shutdown margin during a primary system cooldown at the design rate following a reactor trip.

Thermal and Hydraulic Limits

Avoidance of thermally induced fuel damage during normal operation and anticipated transients is the principal thermal and hydraulic design basis. It is recognized that there is a small probability of limited fuel damage in certain unlikely situations as discussed in Chapter 14.

The following design bases are established for moderate frequency events:

1. The pressures in reactor coolant and main steam systems should be less than 110% of design values.

2. The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. That is, the minimum calculated departure from nucleate boiling ratio is not less than the applicable limits of the DNBR correlation being used.

3. The radiological consequences should be less than 10 CFR 100 guidelines and/or applicable 10 CFR 50.67 limits.

4. The event should not generate a more serious plant condition without other faults occurring independently.

The Reactor Protective System, the incore monitoring system and the reactor control system provide for automatic reactor trip or corrective actions before these design limits are exceeded.

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

Mechanical Design Limits

The reactor internals are designed to safely perform their functions during steady-state conditions and normal operating transients. The internals safely withstand the forces due to deadweight, handling, system pressure, flow-induced pressure drop, flow impingement, temperature differential, shock and vibration. The design limits deflection where required by function. The structural components satisfy stress values given in Section III of the ASME Boiler and Pressure Vessel Code. Components have been subjected to fatigue analysis where required.

The following limitations on stresses or deformations are employed to assure capability of a safe and orderly shutdown in the event of earthquake and major Loss of Coolant Accident loading conditions. For reactor vessel internal structures, the stress criteria are given in Table 3-1. The intent of the limits in this table is as follows:

1. Under design loadings plus design earthquake forces, the critical reactor vessel internal structures are designed within the stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4.

2. Under normal operating loadings plus hypothetical earthquake forces, the design criteria permit a small amount of local yielding.

3. Under normal operating loadings plus pipe rupture loadings plus hypothetical earthquake forces, permanent deformation is permitted by the design criteria.

The structural adequacy of the reactor internals were further evaluated as part of the Combustion Engineering Owner's Group Asymmetric Loads Program (Reference 3) (FSAR Section 14.17.3). A further evaluation (Reference 4) was performed by Combustion Engineering to show that a flaw in the Primary Coolant System will result in a detectable leak before a large guillotine break would occur. The analysis was reviewed by the NRC in SER dated October 27, 1989 (Reference 32). The SER concluded that, with the exception of concerns regarding seismic grid design, Palisades reactor system would withstand the effects of asymmetric LOCA loads and that the reactor could be brought to a cold shutdown condition safely. A seismic analysis of the High Thermal Performance (HTP) fuel design was performed by the fuel vendor. The analysis was reviewed by the NRC in a Safety Evaluation Report (SER) dated April 6, 1992 (Reference 39). The SER concluded that the HTP fuel assemblies will maintain their structural integrity and functionality if subjected to a safe shutdown earthquake.

To properly perform their functions, the critical reactor internal structures are designed to satisfy the deflection limits listed below in addition to the stress limits given in Table 3-1.

Deflection Limits

Under normal design loadings plus design earthquake forces or normal operating loadings plus hypothetical earthquake forces, deflections are limited to 2/3 of the tested functional deflection limit, so that the control rods can function and adequate core cooling is maintained.

Under normal operating loadings plus hypothetical earthquake forces plus pipe rupture loadings, the design criteria on deflection depend on the size of the piping break. If the equivalent diameter of the pipe break is no larger than the largest line connected to the main primary coolant lines, deflections are limited so that: (1) the core will be held in place, (2) the control rods can function normally and (3) adequate core cooling will be maintained. Those deflections which would influence control rod movement are limited to less than 2/3 of the deflections required to prevent control rod function. For pipe breaks larger than the above, the criteria are that the fuel will be held in place in a manner permitting core cooling, and adequate coolant flow passages will be maintained. For the latter case, critical components which meet the stress criteria of Table 3-1 are restrained from buckling by further limiting the stress levels to 2/3 of the stress level calculated to produce buckling.

Fuel Bundles

The fuel bundles are designed to maintain their structural integrity under steady-state and transient operating conditions, as well as for normal handling, shipping and refueling loads. The design takes into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and guide bars, irradiation effects and wear of all components. Mechanical tolerances and clearances have been established on the basis of the functional requirements of the components. All components including welds are highly resistant to the corrosive action of the reactor environment.

The fuel rod design accounts for internal and external pressure, differential expansion of fuel and clad, neutron fluence-induced growth of materials, fuel swelling, fuel densification, clad creep, fission and other gas release, thermal stress, pressure and temperature cycling and flow-induced vibration. The fuel assembly will meet the following design criteria for the expected conditions and postulated accidents to the design assembly burnup:

1. The maximum steady-state cladding and assembly component stresses are within the ASME boiler and Pressure Vessel Code limits.

2. The maximum steady-state cladding strain is below the design limit.

3. The cladding and assembly component fatigue usage factors are below the design limit.

4. Fretting wear of the spacers and fuel rods is precluded.

5. Corrosion of the fuel rod and the fuel assembly structural components is below the design limit.

6. Fuel rod bowing will be limited so that it has no impact on thermal margins.

7. Axial growth of the fuel rods and fuel assembly is accommodated within the design clearances.

8. The fuel rod internal pressure remains below the criteria limit of reactor system pressure plus 800 psi throughout life (Reference 33, 1).

9. Fuel assembly liftoff will not occur during normal operation.

The fuel rod will operate without failure during normal operation and anticipated transients, meeting the following design criteria:

1. Internal hydriding is precluded.

2. Cladding creep collapse will not occur.

3. Adequate cooling exists to prevent overheating of the cladding.

4. Fuel melting will not occur during normal operation and Anticipated Operational Occurrences (AOOs).

5. The transient circumferential strain is within the design limit.

Fuel Bundle Reconstitution

Fuel bundle reconstitution can occur when fuel rods in an assembly are found to have defects. The defective fuel rod is removed and the fuel assembly is reassembled prior to being reinserted in the reactor. The defective fuel rod is normally replaced with either a fuel rod or an inert rod. To ensure that an unreviewed safety question is not created by fuel reconstitution, the NRC issued Supplement 1 to Generic Letter 90-02 (Reference 41) to ensure licensees properly evaluate the changes resulting from fuel reconstitution. A summary of the issues the NRC believes should be evaluated are listed below:

1. Evaluate the applicability of the test data used to derive the correlations and limits for DNBR;

2. Consider the effect on mechanical design such as differential thermal expansion on the seating of the rod or on relaxation of the spacer spring that might cause fretting;

3. Evaluate the effect on the grid strength, or mass, stiffness and fundamental frequency of the fuel assembly as used in the seismic, LOCA, and control rod insertion analyses;

4. Determine if the reconstitution is extensive enough to have core wide effects that could effect the accident analysis.

Control Rods

The control rods are designed to maintain their structural integrity under all steady-state and transient operating conditions, and under handling, shipping and refueling loads. Thermal distortion, mechanical tolerances, vibration and wear are all accounted for in the control rod design. Control rod clearances and corresponding fuel bundle alignment are established so that possible stack up of mechanical tolerances and thermal distortion will not result in frictional forces that prevent reliable operation of the control system. The structural criteria for control rods are based on limiting the maximum stress intensity to those values specified in Section III of the ASME Boiler and Pressure Vessel Code.

The control rod drive mechanism (CRDM) is capable of performing its actuating functions on the control rod under steady-state and transient operating conditions and during hypothetical seismic occurrences. For pipe rupture accident loads, the CRDM is designed to support and maintain the position of the control rod in the core and to be capable of actuating the control rod when these loads have diminished.

The speed at which the control rod is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CRDM clutch releases to allow the control rod and the connecting CRDM components to drop by gravity into the core. The reactivity is reduced during such a rod drop at a rate sufficient to prevent violation of fuel damage limits.

The pressure housing of the CRDM is an extension of the reactor vessel, providing a part of the primary containment for the primary coolant, and is therefore designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. Pressure and thermal transients as well as steady-state loading were considered in this analysis.

3.3 REACTOR DESIGN

3.3.1 GENERAL SUMMARY

A general perspective view of the reactor is shown in Figure 3-1. The reactor core is composed of 204 fuel bundles and 45 control rods.

The fuel is low enriched UO_2 (<5 w/o) encapsulated in zircaloy or M5® fuel rods. The local power distribution is controlled by enrichment zoning within the assemblies and by the use of burnable absorber pins. The overall core is configured in a multiple batch loading pattern to achieve the desired cycle length and control the power distribution.

In order to help control the power distribution and the moderator temperature coefficient, burnable absorber pins are provided in selected fuel assemblies. These burnable absorber pins use Gadolina (Gd_2O_3) as the neutron absorber material. The Gadolina (Gd_2O_3) is mixed into the uranium dioxide and is fabricated into fuel pellets identical to standard uranium dioxide pellets.

In order to reduce the neutron flux at the reactor vessel wall, low power assemblies are placed on the core peripheral flats. This requires third and fourth cycle assemblies with the highest exposures to be placed along the core periphery. In addition, eight (8) neutron shield assemblies are used along the core periphery to reduce vessel fluence. These shield assemblies may be used for up to six cycles.

In all fuel assemblies, the center fuel rod location is replaced by a captured Zircaloy or M5® instrumentation tube which provides an opening in the fuel lattice for the insertion of incore instrumentation. The cage assembly is made up of Zircaloy guide bars, Inconel and/or Zircaloy spacer grids and stainless steel upper and lower tie plates. This structure axially captures and laterally positions and supports the fuel rods and other assembly components. The outer surface of the guide bars also provides an envelope surrounding the control rod channels in the core.

The 45 control rods are made of rectangular stainless steel tubes containing a silver-indium-cadmium alloy that is hermetically sealed within the tube. The tubes are electron beam-welded into a cruciform structure with stainless steel end fittings. Four of the control blades, known as part length control blades, have neutron absorber (silver-indium-cadmium) only in their lower section. They were originally intended for axial power distribution control, particularly for axial xenon oscillations. However, experience at other CE plants indicates that power distribution peaking factors may be violated by using the part-length rods. If it ever becomes desirable to use the part-length control rods again, their use must be justified with further analysis.

3.3.2 NUCLEAR DESIGN AND EVALUATION

This section discusses the design parameters which are of significance to the performance of the core in normal transient and steady-state operational conditions. A discussion of the nuclear design methods employed and comparisons to experiment which support the use of these methods is included.

3.3.2.1 Reactivity and Control Requirements

The maximum excess reactivity is at beginning of life for the core at cold, clean (ie, zero fission product poison concentrations for fresh fuel), unborated conditions. The excess reactivity is reduced as the reactor is taken from Mode 5 to Mode 1. The major effect reducing reactivity is from the Doppler broadening of the fuel absorption cross section. There is also some effect from the moderator temperature increase, but that depends mainly on the boron concentration in the moderator.

Control of the change in the reactivity of the reactor is accomplished both by control rods and by boric acid dissolved in the Primary Coolant System. The control rods provide rapid changes in reactivity such as a reactor trip. They are used to compensate for moderator and fuel temperature changes, and void formation associated with changes in power level. There are 41 standard control rods and 4 part-length control rods. The standard rods are used for two functions: shutdown and regulation. The shutdown rods are combined into two groups and the regulating rods are combined into four groups. During power operation, the shutdown groups are fully withdrawn while the position of the regulating groups is adjusted to meet reactivity and power distribution requirements. All control rods except the part-length rods drop to a fully inserted position upon reactor trip.

Adjustment of the boric acid concentration is used to control the relatively slow reactivity changes associated with Plant heatup and cooldown, fuel burnup and certain xenon variations. Also, additional boric acid is used to provide a large shutdown margin for refueling operations. The use of dissolved boric acid in the water typically makes it possible to maintain control rods in a withdrawn position during full-power operation, thus minimizing distortions in power distribution.

The boron concentration established for refueling is at least 1,720 ppm and must provide at least 5% $\Delta\rho$ shutdown margin with all control rods fully withdrawn. Administrative controls employed in the placement and movement of fuel within the refueling cavity ensure that the 5% $\Delta\rho$ subcriticality margin is maintained during refueling operations. The refueling concentration is approximately equivalent to 1 wt% boric acid (H_3BO_3) in the coolant which is approximately 10% of the solubility limit at refueling temperatures. After a normal shutdown or reactor trip, boric acid is injected into the primary system to compensate for reactivity increases due to normal cooldown and xenon decay. Although the boric acid system reduces reactivity relatively slowly, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of normal cooldown and xenon decay.

Sufficient worth is available in the regulating rods to compensate for the rapid changes in reactivity associated with power level changes. In addition, these rods may be used for partial control of xenon changes and minor variations in moderator temperature and boron concentration. Control rod reactivity allowances are calculated for each reload cycle as part of the safety analysis for that cycle. The total worth of all control rods, including shutdown rods, covers these requirements and also provides adequate shutdown with the most reactive rod stuck in the fully withdrawn position.

Fuel Temperature Variation

The increase in reactivity occurring when the fuel temperature decreases from the full-power value to the zero-power value is due primarily to the Doppler effect in U-238. The total reactivity difference is compensated by control rod movement and soluble boron changes.

Moderator Temperature Variation

The average coolant temperature in the reactor increases with increasing power level and the associated changes in reactivity are controlled by the control rods. The largest increase in reactivity from full power to zero power occurs at the end of the burnup cycle when the least amount of dissolved boron is present. At hot zero power, beginning of life, when the moderator temperature coefficient is near zero, the change in reactivity with moderator temperature is also near zero.

Moderator Voids

A change in reactivity results from the formation of voids due to local boiling in going from zero to full power. The average void content in the core is very small and is estimated to be 1/4% at full power. As with the moderator temperature effect, the maximum increase in reactivity from full to zero power occurs at end of life when the dissolved boron is absent.

Control Rod Bite

The control rod bite is the minimum reactivity worth in control rods which can be in the core and still accomplish the reactivity ramp rates associated with load changes.

Maneuvering Band

An allowance is made in the reactivity worth of the control rods to compensate for variations in xenon, dissolved boron concentration and moderator temperature. When the control rods reach the limits imposed on control rod motion (ie, the power dependent insertion limits), additional reactivity changes will be made by changing the boron concentration.

Shutdown Margin

An allowance of 2% $\Delta\rho$ has been made for the shutdown margin at hot, zero-power conditions with the most reactive rod stuck in the withdrawn position. At least two percent shutdown margin is required by the Technical Specifications.

3.3.2.2 Reactivity Coefficients

Certain factors which contribute to the reactivity of a reactor, such as the thermal utilization, resonance escape probability, and nonleakage probabilities, are dependent upon reactor parameters, such as moderator pressure and temperature and fuel temperature. Reactivity coefficients, denoted by α, relate changes in the core reactivity with variations in these parameters. The utility of these coefficients lies in linking core reactivity to externally imposed conditions in the analysis concerned with determining the response of the reactor core to normal and abnormal plant conditions.

Cycle lifetime effects will change some reactivity coefficients appreciably; therefore, the range of coefficients expected throughout the cycle must be determined to provide adequate control and protection system setpoints.

The Plant transient analysis is summarized in Chapter 14. The reactivity coefficients used in these analyses are listed or referenced in the appropriate sections.

Moderator Temperature Coefficient

The reactivity worths of control rods and boron vary with moderator temperature in opposite directions. The total worth of the control blades decreases with decreasing moderator temperature while the reactivity of a given amount of dissolved boron increases. The interaction of these temperature effects (along with the temperature coefficient of the unborated core) results in a net moderator temperature coefficient of reactivity at operating temperature which ranges from strongly negative to slightly positive, depending upon the moderator temperature, the soluble boron content, the degree of control rod insertion and the fuel burnup.

In a core partially controlled by chemical shim dissolved in the moderator, the moderator coefficient is more positive than that of a similar core controlled entirely by rods. There are two primary reasons for this. First, an increase in moderator temperature decreases neutron absorption in the boron because of both a decrease in moderator density and a hardening of the thermal neutron spectrum. This results in a positive rise in reactivity with temperature. Secondly, the control rods represent a negative contribution to the coefficient, due to the fact that the rod worth increases as the moderator temperature increases, and since there are less rods in the chemically shimmed core than in the unshimmed, (ie, rodded) core, the chemically shimmed core has a more positive coefficient. If, in addition to the soluble shim, neutron absorber rods are employed to control excess reactivity, the moderator temperature coefficient will be made more negative again. This is because less soluble boron will be needed, and because the mechanically fixed neutron absorber rods have the same negative effect on the coefficient as do the control rods.

The allowed range of the moderator temperature coefficient is from +0.00005 $\Delta\rho$/°F to -0.00035 $\Delta\rho$/°F. The upper limit is a limit from the Technical Specifications. In general, the upper limit on the moderator temperature coefficient is used to limit power increases in transients where the primary system is heating up. The lower limit on the moderator temperature coefficient is set by the Plant transient analysis. It is used to limit the return to power after a severe Plant cooldown.

Moderator Pressure Coefficient

The moderator pressure coefficient is the change in reactivity per unit change in primary system pressure. An increase in pressure slightly increases the water density; therefore, the pressure coefficient is usually opposite in sign to the temperature coefficient. The reactivity effect of increasing the pressure is reduced in the presence of a large amount of dissolved boron because an increase in water density adds significant boron to the core.

Fuel Temperature Coefficient

The fuel temperature coefficient, α_{fuel} (commonly called the Doppler coefficient), reflects the change of core reactivity with fuel temperature. The effect may be broken into two parts, namely, thermal and epithermal (Doppler) contributions. The thermal contribution is due to hardening of the spectrum as the temperature increases. The epithermal contribution is the temperature dependence of the resonance escape probability, which in turn is physically due to Doppler broadening of the absorption resonances in U-238.

Power Defect

The power defect is the integrated reactivity difference between zero power and some higher power level. The reactivity difference is caused by both the moderator temperature effect and by the fuel temperature effect. The value is always negative; that is, reactivity must be added to the core to increase the power level. The curve in the Start-Up and Operations Report is computed with no control rods in the core.

3.3.2.3 Control Blade Worths

Figure 3-2 identifies the core locations and the groupings of the control blades. The total worth available in the 41 full-length, scrammable control rods must be enough to shut the reactor down by at least 2% $\Delta\rho$. The shutdown margin is evaluated at BOC and EOC for HFP and HZP conditions and is defined as the difference between the total control rod worth, less the worth of the most reactive rod (N-1), and the total shutdown requirements.

The worth of all control rods is calculated at HZP. Then the worth is reduced to account for full-power equilibrium xenon as the starting point for the Plant transient analysis of the steam line break which sets the shutdown margin requirement. The N-1 worth is the worth of all banks minus the most reactive rod which is assumed to be stuck out of the core. Reference 46 verifies that 225 ppm boron is a bounding value for the worth of a stuck control rod. This value will also be verified for each future cycle. To ensure that there is sufficient shutdown margin in the core, a 10% reduction is made in the prediction of the N-1 worth.

Shutdown requirements include allowances for power defect, flux redistribution, power dependent insertion limit (PDIL) Group 4 rod insertion and void effects. The power defect (moderator and Doppler) is separated from the flux redistribution effect by the method of performing the calculation. The flux redistribution and void effects are bounding values derived from a calculation performed for a typical PWR at EOC conditions for a severe xenon distribution (Reference 12). The reactivity allowance for HZP and HFP Group 4 insertion is calculated as the worth of the bank inserted to its respective PDIL limits. The PDIL is based both on shutdown margin requirements and on power distribution peaking factor limits.

Excess shutdown margin is defined as the shutdown margin minus the required shutdown margin. The value used for the required shutdown margin is 2.0% $\Delta\rho$ at both the BOC and EOC.

3.3.2.4 Reactivity Insertion Rates

Reactivity insertion from control rod withdrawal, either a single blade or group of blades, has been analyzed (Chapter 14) to show that there are no unsafe consequences resulting from the transient. See Section 14.2 for bank and single rod withdrawal reactivity insertion rates.

The maximum rate of reactivity insertion due to boron removal by operation of the Chemical and Volume Control System is about 1/17 of the rate available from the withdrawal of rods. Adequate time is available to take corrective measures as described in the analysis of the boron dilution incident (Section 14.3). Section 14.3 also shows that the reactor operator has sufficient time to recognize and to take corrective action to compensate for the maximum reactivity addition due to xenon decay and cooldown.

3.3.2.5 Power Distribution and Power Escalation Rates

The power distribution in the core, especially the peak power density, is of major importance in determining core thermal margin. Enrichment zoning within fuel bundles is used to reduce local power peaking.

Since dissolved boron is used to control long-term reactivity changes such as burnup, the control rods do not need to be used to a great extent. Regulating rod insertion is limited by the PDIL graph in the Palisades Plant Core Operating Limits Report.

Several power distribution limits have been established to protect against fuel failures. A limit on the linear heat generation rate that is a function of the axial location of the peak power in the pin protects against departure from nucleate boiling and from overheating during a LOCA. The LHGR limits are given in the Palisades Plant Core Operating Limits Report (COLR).

There is an additional limit on the axially collapsed radial peaking factor that also protects against fuel failures. This limit ensures that the margin to DNB and the linear heat generation rate is not violated during normal or transient conditions and that the thermal margin/low-pressure trip and the high-power trip set points remain valid during normal operations. The peaking factor is given in the COLR. The peaking factor definition is:

Total Radial Peaking Factor - F_r^T

The total radial peaking factor is the maximum ratio of the individual fuel pin power to the core average fuel pin power integrated over the total core height, including tilt.

The linear heat generation rate (LHGR) and Peaking Factor limit shown in the COLR must be reduced by several factors before all the necessary conservatisms are taken into account. To account for the change of dimensions from densification (due to resintering) and thermal expansion, the LHGR limits are reduced by dividing them by 1.03. To account for the uncertainty in the reactor thermal power, the LHGR limits are reduced by dividing them by 1.02. To account for the calculational uncertainties of the incore monitoring system (Reference 35, 36 and 37), the limits are reduced by dividing them by the appropriate measurement uncertainties in COLR Table 2.4-2. The NRC Safety Evaluation and Technical Specification Amendment, dated April 3, 1992, which approved these uncertainties, and the subsequent NRC Safety Evaluations, dated May 6, 1997 and January 31, 2001, govern the use and describe the limitations of the incore monitoring system.

Unrestricted power escalation can cause fuel failures at relatively modest power levels due to differential thermal expansion of fuel pellets and cladding under non-steady state power conditions. Differential thermal expansion results in mechanical interaction between the fuel cladding and the fuel pellets. Mechanical interaction between cladding and pellets can also be caused by pellet relocation. Mechanical interactions such as these are called Pellet-Clad Interactions (PCI) and can result in high localized stress level in the cladding. The concentration of the high stress levels and the fission product environment within the fuel rod may result in cladding failure due to a stress-corrosion cracking (SCC) mechanism.

The fuel manufacturer provided recommendations on allowable power escalation rates. These recommendations are incorporated into plant operating procedures with some additional conservatism included.

3.3.2.6 Neutron Fluence on Pressure Vessel

In May 1988, the NRC issued Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." As a result of this change, the Palisades reactor vessel was projected to be approaching the Pressurized Thermal Shock (PTS) screening criteria in 10 CFR 50.61. Because it was impractical in the time available to submit an analysis justifying operation beyond the PTS screening criteria, a fluence reduction program was initiated in Cycle 8. A low leakage fuel management scheme with partial stainless steel shielding assemblies near the critical axial weld locations was employed to reduce the vessel wall flux. Additional flux reduction was initiated in Cycle 9 using Hafnium poisoned assemblies in place of stainless steel. Cycle 10 was designed using specially fabricated shield assemblies in addition to the Hafnium poisoned assemblies. Cycles 11 and 12 were designed using only stainless steel equipped shield assemblies. The Cycle 13 fluence reduction method used standard Reload N assemblies with three cycles of exposure and Cycle 14 used fourth cycle Reload O assemblies. Cycle 15 used fourth cycle Reload "P" assemblies and neutron shielding (SAN) assemblies. The Cycle 15 loading pattern placed the 8 SAN bundles used in Cycle 10, 11, and 12, directly in front of the six RPV axial welds. In addition, Cycle 15 placed fresh assemblies adjacent to each other along the major core axes (in the interior core region) in order to create an ultra-low leakage core. Reactor vessel fluence evaluations have been performed by Westinghouse. WCAP-15353, Revision 0, included cycle specific evaluations through cycle 14 (References 2 and 7). WCAP-15353, Supplement 1, included cycle specific evaluations for cycles 15 through 21, and assumed a 95% cycle capacity factor for future cycle projections (Reference 53). Using the fluence information, the date when the most limiting reactor vessel material for PTS would reach the 10 CFR 50.61 screening criterion limit was determined. The most limiting material for PTS were the axial welds fabricated from weld material heat W5214. The 10 CFR 50.61 screening criterion limit would not be reached until April 2017 (Reference 54). An updated reactor vessel fluence evaluation that reflected recent actual reactor operation determined that the 10 CFR 50.61 screening criterion would be reached in August 2017 rather than in April 2017 (Reference 56).

In 2014, a license amendment request to implement 10 CFR 50.61a was submitted to the NRC (Reference 109). The alternate PTS evaluation accompanying the submittal concluded that the reactor vessel materials remain below the 10 CFR 50.61a screening criteria through the end of licensed life (42.1 effective full power years) (Reference 110). The NRC approved the license amendment request in 2015 (Reference 111).

A low radial leakage core loading pattern, with 8 shield assemblies, continues to be used to reduce the neutron fluence on critical pressure vessel welds.

A supplemental dosimetry program has been established. Ex-vessel dosimetry has been used to monitor the fluence at various locations during Cycles 8, 9,10, and 11. The Ex-vessel program has been discontinued based on sufficient benchmarking. In-vessel dosimetry has also been employed to monitor Cycle 9 fluence at surveillance capsule location W-290. Irradiated dosimeters have been analyzed and measured flux values have been determined. These flux values have been used for benchmarking the vessel/fluence calculations.

3.3.2.7 Nuclear Evaluation

Nuclear Design Methods

Framatome performs PWR reload design analyses with the SAV95 code system. Siemens Power Corporation submitted the SAV95 Topical Report on May 8, 1996, and the USNRC approved the use of SAV95 in a Safety Evaluation Report (SER) on October 29, 1996. Reference 9 contains a copy of the SER, the Topical Report, and the restrictions on the use of SAV95. In addition, SPC performed benchmark calculations for Palisades Cycles 11 to 14, in Reference 10, to verify the SAV95 methodology against plant specific measurements from Startup Physics Tests and the Incore Monitoring System. Each subsequent Palisades cycle specific SAV95 model encompasses any necessary adjustments to the calculations (ie, Boron Biases). Refer to the current reload Facility Change package for any additional information.

3.3.2.8 Reactor Stability

Xenon stability analyses on the Palisades core indicate that any radial and azimuthal xenon oscillations induced in the core will be damped, and that the core could exhibit instabilities with respect to axial xenon oscillations during certain portions of the burnup cycle, in the absence of appropriate control action. Before discussing the methods of analysis employed to obtain these predictions, it is appropriate to reiterate several important aspects of the xenon oscillation problem.

1. The time scale on which the oscillations occur is long, and any induced oscillations typically exhibit a period of 30 to 50 hours.

2. Xenon oscillations are detectable as discussed below.

3. As long as the initial power peak associated with the perturbation initiating the oscillation is acceptable, the operator has time, on the order of hours to days to decide upon and to take appropriate remedial action. This action is to prevent the allowable peaking factors from being exceeded.

Method of Analysis

The classic method for assessing spatial xenon oscillations is that developed by Randall and St John (Reference 24), which consists of expanding small perturbations of the flux and xenon concentrations about equilibrium values in eigenfunctions of the system with equilibrium xenon present. While the Randall-St John technique is correct only for a uniform unreflected system, its use of the separations between the eigenvalues of the various excited states of the system and the eigenvalue of the fundamental state is helpful in directing attention to which of the various excited states are the most likely to occur. As indicated in Figure 3-4, the first axial mode, which has the minimum eigenvalue separation from fundamental mode, is the most likely to occur, and the higher modes would have, on the basis of this simple theory, the indicated relative likelihoods of occurrence.

However, it is necessary to extend this simpler linear analysis to treat cores which are nonuniform because of fuel zoning, depletion and control rod patterns, for example. Such extensions have been worked out and are reported in References 25 and 26. In this extension, the eigenvalue separations between the excited state of interest and the fundamental are computed numerically for symmetrical flux shapes. For nonsymmetrical flux shapes, the eigenvalue separation can usually be obtained indirectly from the dominance ratio, computed during the iteration cycle of the machine spatial calculation.

In making the analysis, numerical space-time calculations are performed in the required number of spatial dimensions for the various modes as checkpoints for the predictions of the extended Randall-St John treatment described above.

Radial Mode Oscillations

From the remote position of the first radial excited eigenvalue in Figure 3-4 (over 4% in λ), it is expected that such oscillations would be rapidly damped even in a core whose power was flattened by; eg, enrichment zoning. To confirm that this mode is extremely stable, a space-time calculation was run for a reflected, zoned core 11 feet in diameter without including the damping effects of the negative power coefficient. The initial perturbation was a poison worth 0.4% in reactivity placed in the central 20% of the core for one hour. Following removal of the perturbation, the resulting oscillation was followed in 4-hour time steps for a period of 80 hours. As shown in Figure 3-5, the resulting oscillation died out very rapidly with a damping factor of about -0.06 per hour. If this damping coefficient is corrected for a finite time mesh by the formula in Reference 27, it would become even more strongly convergent. On this basis, one is led to conclude that radial oscillations are highly unlikely.

This conclusion is of particular significance because it means that there is no type of oscillation where the inner portions of the core act independently of the peripheral portions of the core whose behavior is most closely followed by the excore flux detectors. As will be noted later, primary reliance is placed on these for the detection of any xenon oscillations.

Azimuthal Mode Oscillations

Azimuthal oscillations in an unreflected uniform reactor are less likely than axial mode oscillations as had been indicated in Figure 3-4. The situation is quite different in a radially power flattened reflected core even at beginning of life, as shown in Figure 3-6. Here, the eigenvalue separations for the actual core are predicted by the modified Randall-St John treatment and include the effects of power flattening. On the basis of this information, it appears that the azimuthal mode is the most easily excited at beginning of life even though the axial mode becomes the most unstable later.

With reference to Figure 3-6, it is indicated that the eigenvalue separation between the first azimuthal harmonic and the fundamental is about 0.7% in λ. Although the axial oscillations were found to be relatively insensitive to the moderator temperature feedback because of the constant power condition, the azimuthal modes should be stabilized appreciably by the negative moderator coefficient. Furthermore, the Doppler coefficient applicable to the Palisades reactor is calculated to be approximately 1.08×10^{-5}, $\Delta\rho/°F$ at HFP, BOC, which is more than enough to ensure stability of all the azimuthal modes.

Axial Mode Oscillations

As checkpoints for the predictions of the modified Randall-St John approach, numerical spatial time calculations have been performed for the axial case at both beginning and end of cycle. The fuel and poison distributions were obtained by depletion with soluble boron control so that, although the power distribution was strongly flattened, it was still symmetric about the core midplane. Spatial Doppler feedback was included in these calculations. In Figure 3-7, the time variation of the thermal neutron flux is shown for two points along the core axis near end of life with Doppler feedback. The initial perturbation used to excite the oscillations was a 20% insertion into the top of the reactor of a 1.5% reactivity rod bank for one hour. As is indicated, the damping factor for this case was about +0.02 per hour. When corrected for finite time mesh by the methods of Reference 27, however, the damping factor became more like +0.05. When this damping factor is plotted on Figure 3-6 at the appropriate eigenvalue separation for this mode at end of cycle, it is apparent that good agreement is obtained with the modified Randall-St John prediction.

At beginning of cycle, the space-time calculations indicated a positive damping coefficient of about +0.04 per hour in the absence of spatial Doppler feedback, and a negative damping coefficient of -0.05 per hour results with a power coefficient of -3.4×10^{-6} $\Delta\rho$/MWt. Again, these space-time results are in excellent agreement with the predictions of the modified Randall-St John technique.

Calculations performed with both Doppler and moderator feedback have resulted in damping factors which were essentially the same as those obtained with Doppler feedback alone. This result suggests that the constant power condition which applies to the axial oscillations results in a very weak moderator feedback since the moderator density is fixed at the top and bottom of the core and only the density distribution in between can change. For the Doppler coefficient of -4.6×10^{-6} $\Delta\rho$/MWt estimated for Palisades, it can be seen from Figure 3-6 that the damping factor toward end of the burnup cycle is about zero; thus, within the uncertainties in predicting power coefficients and uncertainties in the analysis, there is a distinct possibility of unstable axial xenon oscillations.

Detection of Xenon Oscillations

Primary reliance for the detection of any xenon oscillations is placed on the excore flux monitoring instrumentation, one channel of which per quadrant is an axially split ionization detector. As indicated earlier, oscillations in modes such as the radial, which would allow the center of the core to behave independently from the peripheral portions of the core, are highly unlikely and this lends support to reliance on the excore detectors for this purpose. Furthermore, as an example of the ability of the axially split excore detectors to respond to flux tilts in the core, we have included Figure 3-8, which indicates the ratio of the lower half of the axially split detector signal to the signal from the upper half for two different power distributions: one axially symmetric, the other containing a strong contribution from the first axial harmonic and having a peaking factor of about 1.8. In the latter case, the signal seen from the lower half of the detector was 50% higher than that seen from the upper half.

Keeping in mind that the primary response of these detectors will be to the power shapes in the peripheral fuel assemblies, but noting that the lower modes of any induced oscillations will affect the power shapes in these peripheral assemblies, we conclude that any flux tilts can be observed and identified by the use of excore instrumentation to provide data upon which appropriate remedial action can be based.

In addition, the incore instrument detectors provide information which will be used in the early stages of operation to confirm predicted correlations between indications from the excore detectors and the space-dependent flux distribution within the core. Later on, during normal operation, the incore detector system provides information which may be used to supplement that available from the excore detectors.

Operating Experience

The conclusions of the above xenon stability analysis have been confirmed through power testing and many years of operating experience. The Palisades reactor is very stable in the radial and azimuthal directions, and the only significant oscillations observed were deliberately induced during tests. The reactor is less stable in the axial direction, as oscillations can be induced through normal control rod movements and power level changes. However, the axial power shape changes are monitored by the excore detectors through the Thermal Margin Monitor, and are readily controlled by slight insertions of the regulating rods at appropriate times in the oscillation. Even at the operating state of least stability (end of cycle, full power), the damping factor appears to be slightly negative and the power distribution remains stable unless perturbed.

3.3.3 THERMAL-HYDRAULIC DESIGN AND EVALUATION

The thermal-hydraulic design of the reactor has as its primary objective, the assurance that the core can meet normal steady-state and transient performance requirements without exceeding thermal-hydraulic design limits. This subsection, therefore, discusses the thermal-hydraulic characteristics that relate reactor performance to the margin to design limits.

3.3.3.1 Thermal-Hydraulic Design Criteria

The requirements of 10 CFR 50, Appendix A, Criteria 10, 20, 25 and 29 require that the design and operation of the Plant and the Reactor Protective System assure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation including the effects of anticipated operational occurrences (AOOs). As per the definition of AOO in 10 CFR 50, Appendix A, "Anticipated Operational Occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the Plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." The specified acceptable fuel design limits (SAFDLs) are that: (1) the fuel shall not experience center line melt ; and (2) the departure from nucleate boiling ratio (DNBR) shall have a minimum allowable limit such that there is a 95% probability with a 95% confidence interval that departure from nucleate boiling (DNB) has not occurred.

3.3.3.2 Plant Parameter Variations

Normal reactor operation includes both the nominal steady-state design conditions and variations from these conditions during expected operating transients. Instrument and control errors are taken into account in the analysis of transients by setting the initial conditions at the most adverse values within the steady-state operating envelope. Delays between parameter changes, trip signals and initiation of rod movement are made a part of the transient calculations. Values of Plant parameters are given in Section 14 for the nominal and steady-state design conditions.

3.3.3.3 Core Flow Distribution

The XCOBRA-IIIC code (Reference 20) performs steady-state calculations including the effect of cross flow mixing between fuel assemblies and subchannels using boundary conditions produced by approved transient computer codes. Thermal-hydraulic parameters such as DNBR by means of appropriate critical heat flux correlation (References 22, 23, 42, 43 and 45), local quality and void fraction are calculated for each node. The degree of core-wide nodalization and the modeling options available in the code provide calculational flexibility. This code is used for thermal-hydraulic parameter evaluation to generate the TM/LP trip function (Section 3.3.3.4).

The core flow distribution calculation directly models the thermal and hydraulic performance of each fuel assembly as appropriate single hydraulic channels. The thermal performance is evaluated using approved neutronics methods to determine the core and assembly peaking distribution while the hydraulic performance is determined using the results of pressure drop tests. Results of the calculation indicated which fuel assembly will experience the least coolant flow rate and that fuel assembly is selected for the TM/LP trip and LCO calculations.

The limiting assembly calculations model the limiting (highest power) fuel assembly into appropriate subchannels with the assembly flow rate as determined above. The calculation is consistent with the methodology used for the core flow distribution calculations. This calculation determines the limiting subchannel flow rates used in the ensuring MDNBR calculation necessary to establish or verify both TM/LP and LCO setpoints for each fuel cycle.

The calculations include factors to account for manufacturing tolerances and densification effects. Specifically, a 3% engineering factor is applied to the limiting rod power to account for fabrication tolerances on pellet diameter, density, enrichment and cladding diameter. These manufacturing tolerances potentially affect heat flux at the limiting DNBR location in the assembly.

Core Bypass Flow

The core bypass flow is 3% of the total primary coolant flow and consists of the following flow paths:

1. From the vessel entrance region, through clearances between the outlet nozzles and the core support barrel extensions, into the outlet nozzles.

2. From the vessel entrance region, past the alignment keys, into the closure head volume.

3. Through core support plate flow holes into the core-shroud-support barrel annulus, exiting at the underside of the fuel alignment plate.

4. Through vertical gaps at corners of the core shroud structure, into the shroud-support barrel annulus, exiting at the underside of the fuel alignment plate.

5. Entering low in the core, through flow holes into the instrument tubes and exiting above the fuel assembly upper end fittings.

6. Entering low in the core, through flow holes into the empty tubes (e.g. poison cluster guide tubes) and exiting above the upper end fittings.

Core bypass flow reduces the coolant flow through the fuel assemblies and thus affects the minimum DNBR (DNBR References in Section 3.3.3.3).

3.3.3.4 Trip Set Points

A T_{inlet} LCO and thermal margin/low pressure (TM/LP) trip were developed for operation with the modified Reactor Protective System (RPS). Their development is presented in Reference 21. Beginning with Cycle 18, a statistical setpoint analysis method is used (Reference 11). The T_{inlet} LCO provides protection against penetrating DNB during limiting anticipated operational occurrence (AOO) transients. The T_{inlet} LCO is given in Technical Specifications LCO 3.4.1.

The most limiting AOO transient that does not produce a reactor trip is the inadvertent drop of a full-length control assembly. The T_{inlet} LCO must provide DNB protection for this transient assuming a return to full power with enhanced peaking due to the anomalous control assembly insertion pattern.

The modified RPS includes the hardware for a new TM/LP trip which was installed at the Palisades Plant during the 1988 refueling outage. This new TM/LP is an improvement over the previous trip in that it allows monitoring of the core axial shape index.

The function of the TM/LP trip is to protect against slow heat-up and depressurization transient events. In order to perform this function, the TM/LP trip must initiate a scram signal prior to exceeding the specified acceptable fuel design limits (SAFDLs) on departure from nucleate boiling (DNB) or before the average core exit temperature exceeds the saturation temperature. The SAFDL ensures that there is no damage to the fuel rods and the limit on core exit saturation is imposed to assure meaningful thermal power measurements.

The TM/LP trip works in conjunction with the other trips and the limiting conditions of operation (LCO) on control rod group position, radial peaking, and reactor coolant flow. The variable high power (VHP) trip is factored into the TM/LP development by limiting the maximum possible power that can be achieved at a particular radial peaking to 15% (Reference 38) above the power corresponding to that radial peaking. The LCO on the control rod group position is included in the TM/LP through monitoring of the axial shapes, and the LCO on radial peaking is factored in by including its variation with power level in the TM/LP development. Finally, the LCO on reactor coolant flow is built into the TM/LP through the use of conservative flows throughout its development.

3.3.4 MECHANICAL DESIGN AND EVALUATION

The reactor core and internals are shown in perspective in Figure 3-1. A cross section of the reactor core and internals is shown in Figure 3-9. A vertical section of the core and internals is shown in Figure 3-10. Mechanical design features of the reactor internals, the control rod drive mechanisms and the reactor core are described below.

3.3.4.1 Reactor Internals

The reactor internals are designed to support and orient the reactor core fuel bundles and control rods, 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 internals are designed to safely perform their functions during all steady-state conditions and during normal operating transients. The internals are designed to safely withstand the forces due to deadweight, handling, system pressure, flow impingement, temperature differential, shock and vibration. All reactor components are considered Class 1 for seismic design. The reactor internals' design limits deflection where required by function. The structural components satisfy stress values given in Section III of the ASME Boiler and Pressure Vessel Code. Certain components have been subjected to a fatigue analysis. Where appropriate, the effect of neutron irradiation on the materials concerned is included in the design evaluation.

The components of the reactor internals are divided into three major parts consisting of the core support barrel (including the lower core support structure and the core shroud), the upper guide structure (including the control rod shrouds and the incore instrumentation guide tubes) and the flow skirt. These components are shown in Figure 3-10.

Core Support Assembly

The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core support barrel, the core support plate and support columns, the core shrouds, the core support barrel to pressure vessel snubbers and the core support barrel to upper guide structure guide pins. The major material for the assembly is Type 304 stainless steel.

The core support assembly is supported at its upper flange from a ledge in the reactor vessel flange. The lower end is restrained in its lateral movement by six core support barrel to pressure vessel snubbers. Within the core support barrel are axial shroud plates and former plates which are attached to the core support barrel wall and the core support plate and form the enclosure periphery of the assembled core. The core support plate is positioned within the barrel at the lower end and is supported both by a ledge in the core support barrel and by 52 columns. The core support plate provides support and orientation for the fuel bundles. Also within the core support barrel just below the nozzles are four guide pins which align and prevent excessive motion of the lower end of the guide structure relative to the core support barrel during operation.

Core Support Barrel

The core support barrel carries the entire weight of the core and other internals (about 485,000 pounds). It is a right circular cylinder with a nominal inside diameter of 149-3/4 inches and a minimum wall thickness in the weld prep area of 1 inch. It is suspended by a four-inch-thick flange from a ledge on the pressure vessel. The core support barrel in turn supports the core support plate upon which the fuel bundles rest. Press fitted into the flange of the core support barrel are four alignment keys, three measuring 3.25-inch x 4-inch x 12-inch and one 3.25-inch x 5-inch x 12-inch. The keys are located 90 degrees apart. The reactor vessel, closure head and upper guide structure assembly flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

Since the core support barrel is 27 feet long and is supported only at its upper end, it is possible that coolant flow could induce vibrations into the structure. Therefore, amplitude limiting devices, or snubbers, are installed near the bottom outside end of the core support barrel (CSB). The snubbers consist of six equally spaced double lugs around the circumference which are the grooves of the "tongue-and-groove" assembly in which the pressure vessel lugs are the tongues. Minimizing of the clearance between the two mating pieces prevents the barrel from undergoing vibrations of significant amplitude. At assembly, as the internals are lowered into the vessel, the pressure vessel tongues engage the core support grooves in an axial direction. With this design, the internals may be viewed as a beam with supports at the farthest extremities. Radial and axial expansions of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted by this design. The pressure vessel tongues have bolted, lock-welded Inconel shims, and the core support barrel grooves are hard faced with stellite to minimize wear.

Core Support Plate and Support Columns

The core support plate, 1-1/2 inches thick, is a perforated member with flow distribution and pin locating holes for each fuel bundle. The plate is supported by a ledge and by columns. The ledge on the CSB supports the periphery of the plate, and the plate is pinned, bolted and lock welded to the ledge for maintaining accurate location of the plate. A series of columns are placed between the plate and the beams across the bottom of the core support barrel. The columns provide stiffness and transmit the core load to the bottom of the core support barrel.

Core Shroud Plates and Centering Plates

The core shroud follows the perimeter of the core and limits the amounts of coolant bypass flow. The shroud consists of rectangular plates 5/8 inch thick, 145 inches long and of varying widths. The bottom edges of these plates are fastened to the core support plate by use of anchor blocks.

The critical gap between the outside of the peripheral fuel bundles and the shroud plates is maintained by seven tiers of centering plates attached to the shroud plates and centered during initial assembly by adjusting bushings located in the core support barrel. The overall core shroud assembly, including the rectangular plates, the centering plates, and the anchor blocks, is a bolted and lock-welded assembly. In locations where mechanical connections are used, bolts and pins are designed with respect to shear, binding and bearing stresses. The core shroud assembly is designed with some inherent flexibility to minimize internal stresses at fastener locations while maintaining necessary clearances. Because pressure is equalized across inner and outer shroud faces at both the upper and lower ends of the shroud, differential pressure across the shroud during transients will remain relatively low. All bolts and pins are lock welded. In addition, all bolts (bodies and heads) are designed to be captured in the event of fracture. Holes are provided in the core support plate to allow some coolant to flow upward between the core shroud and the core support barrel, thereby minimizing thermal stresses in the shroud plates and eliminating stagnant pockets.

Flow Skirt

The Inconel flow skirt is a perforated (2-1/2 inch diameter holes) right circular cylinder, reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is hung by welded attachments from the core stop lugs near the bottom of the pressure vessel and is not attached to the core support barrel.

Upper Guide Structure Assembly

This assembly (Figure 3-11) consists of a flanged grid structure, 45 control rod shrouds, a fuel bundle alignment plate and a ring shim. The upper guide structure aligns and supports the upper end of the fuel bundles, maintains the control rod channel spacing, prevents fuel bundles from being lifted out of position during a severe accident condition and protects the control rods from the effect of coolant cross flow in the upper plenum. It also supports the incore instrumentation guide tubing. The upper guide structure is handled as one unit during installation and refueling.

The upper end of the assembly is a flanged grid structure consisting of a grid array of 18-inch-deep long beams in one direction with 9-inch-deep short beams at 90 degrees to the deeper beams. The grid is encircled by an 18-inch-deep cylinder with a 3-inch-deep flange welded to the cylinder. The periphery of the flange contains four accurately machined and located alignment keyways, equally spaced at 90-degree intervals which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid aligns and supports the upper end of the control rod shrouds.

The control rod shrouds are of cruciform configuration and extend from about 1 inch above the fuel bundles to about 2 inches above the top of the pressure vessel flange. They enclose the control rods in their fully withdrawn position above the core, thereby protecting them from adverse effects of flow forces. The shrouds consist of 4 formed plates, 0.187 inch thick by approximately 138 inches long, which are welded to 4 end bars to form a cruciform-shaped structure. The shrouds are fitted with support pads at the upper end machined for a bolted and lock-welded attachment to the flanged grid structure. The lower ends of the shrouds are also fitted with support pads machined for a bolted and lock-welded attachment to the fuel bundle alignment plate. The cruciform design provides a stiff section, resulting in low stresses and deflections. In the area of maximum cross flow, the shroud is supported between the flanged grid structure and the fuel bundle alignment plate as a beam with fixed ends.

The fuel bundle alignment plate is designed to align the upper ends of the fuel bundles and to support and align the lower ends of the control rod shrouds. Precision machined and located pins attached to the fuel bundle alignment plate align the fuel bundles. The fuel bundle alignment plate also has four equally spaced slots on its outer edge which engage with stellite hard-faced pins protruding out from the core support barrel to prevent lateral motion of the upper guide structure assembly during operation. Since the weight of a fuel bundle under all normal operating conditions is greater than the flow lifting force, it is not necessary for the upper guide structure assembly to hold down the core. However, the assembly does capture the core and would limit upward movement in the event of an accident condition.

A hold-down device bears on the top of the flange of the upper guide structure to resist axial movement of internals assembly, compensate for axial differential thermal expansions and compensate for closure head rotation considerations during bolt-up and pressurization. The hold-down ring (see Figure 3-12) contains 308 holes, all but one of which contain plungers supported by 22 Belleville washers (each) which are contained within a 304 SS frame. The frame, or ring segments, are bolted to the upper guide structure to provide uniform rigidity within the segments. The design loading of the hold-down device will produce a compression resulting in net hold-down force of nominally 700,000 pounds. In addition, a .290-inch shim is located between the upper guide structure and core support barrel flanges to accommodate fuel growth.

The upper guide structure assembly also supports the incore instrument guide tubes. The tubes are conduits which protect the incore instruments and guide them during removal and insertion operations while refueling.

3.3.4.2 Control Rod Drive Mechanism

The control rod drive mechanism (CRDM) drives the control rod within the reactor core and indicates the position of the control rod with respect to the core. The speed at which the control rod is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CRDM drive releases to allow the control rod and the supporting CRDM components to drop by gravity into the core. The reactivity is reduced during such a rod drop at a rate sufficient to control the core under any operating transient or accident condition.

The control rod is decelerated at the end of the rod drop insertion by the CRDM which supports the control rod in the fully inserted position.

There are 45 CRDMs mounted on flanged nozzles on top of the reactor vessel closure head, located directly over the control rods in the reactor core. Each CRDM is connected to a control rod by a locked coupling. The weight of the CRDMs is carried by the vessel head. In order to provide lateral stability, particularly in resisting horizontal earthquake forces, the CRDMs are supported in the horizontal direction by interconnection. The interconnecting structure permits limited vertical movement due to thermal expansion, but restricts bending deflection so as to limit stresses to allowable values in the lower housing and nozzle areas.

The CRDM is designed to handle a control rod weighing 215 pounds (dry). The total stroke of the drive is 132 inches. The speed of the drive is 46 inches per minute. For a reactor trip, the time from receiving a trip signal to 90% of the full-in position of the rod is less than 2-1/2 seconds. The rod is allowed to accelerate to about 11 ft/s and is decelerated to a stop at the end of the stroke.

The CRDM is of the vertical rack-and-pinion type with the drive shaft running parallel to the rack and driving the pinion gear through a set of bevel gears. The design of the drive is shown in Figure 3-13. The rack is driven by an electric motor operating through a gear reducer and a magnetic clutch. By de-energizing the magnetic clutch, the control rod drops into the reactor under the influence of gravity. The drive assembly is equipped with a magnetic brake and an antireversing clutch which maintain the position of the rod with the drive in the holding condition and prevent upward movement of the rod when in the scrammed condition. For actuating partial length control rods which maintain their position during a reactor trip, the CRDM is modified by replacing the magnetic clutch with a solid shaft assembly which eliminates the trip function. Otherwise, this CRDM is the same as those attached to the full-length control rods. The drive shaft penetration through the pressure housing is closed by means of a face-type rotating seal. The rack is connected to the control rod blade by means of a tie bolt which extends through the rack to a connecting shaft engaged with the upper end of the control rod. The rack is connected to the control rod by means of a rack extension containing a bayonet-type coupling. The rack extension is connected to the rack through a tie rod by means of a nut and locking device at the upper end of the rack. The tie bolt is fixed to the rack by means of a nut and locking device at the upper end of the rack. A small diameter closure is provided at the top of the pressure housing for access to this nut for releasing the control rod from the CRDM. The rack is guided at its upper end by a section having an enlarged diameter which operates in a tube extending the full length of the rod travel. The final cushioning at the end of a rod drop is provided by the dashpot action of the guiding section of the rack entering a reduced diameter in the guide tube.

Pressure Housing

The pressure housing consists of a lower and an upper section joined near the top of the drive by means of a threaded autoclave-type closure. The pressure housing design and fabrication conform to the requirements of the ASME Pressure Vessel Code, Section III, for Class A vessels (Class 1 vessel for replacement housing). The housing is designed for steady-state conditions, as well as all anticipated pressure and thermal transients.

The lower housing section has an integral bottom head, which consists of the eccentric reducer and the lower flange which is machined from a single piece bar stock. This flange fits the nozzle flange provided on the reactor vessel closure head and is seal welded to it by an omega-type seal. Once seal welded and bolted into place, the lower pressure housing need not be removed since all servicing of the drive is performed from the top of this housing. The upper part of the lower housing is machined to form the closure and is provided with a recessed gasket surface for a spirally wound gasket.

The upper part of the pressure housing has a flange which mates with the lower housing closure, a cavity which contains the drive rotating seal, and a tubular housing extension with a small flange closure which provides access for attaching and detaching the control rod.

The shaft seals are hydraulically balanced face seals utilizing stationary O-rings for the shaft and pressure housing seals. The rotating, axially movable member has a carbon-graphite seating surface which in the original design mated to a stationary member made of a carbide alloy. The carbide alloy was replaced with chromium oxide applied directly to the stainless steel body with no bond coat. The carbide alloy was found to present problems because a nickel binder was preferentially leaching out onto the seating surface.

The two parts of the seal are fitted with O-rings to prevent leakage around the seal. The O-rings are static seals. A cooling jacket surrounds the seal area to maintain the temperature of the seal and O-rings below 250°F. This cooling water is from the Component Cooling System and is under low pressure and not connected to the primary water system. A seal leakage collection cup is provided with a thermocouple in the seal leak-off line to monitor for cooling water or seal failure. Seal leakage is drained to the containment sump.

Rack-and-Pinion Assembly

The rack-and-pinion assembly is an integrated unit which fits into the lower pressure housing and couples to the motor drive package through the upper pressure housing. This unit carries the bevel gears which transmit torque from the vertical drive shaft to the pinion gear. The vertical drive shaft has splined couplings at both ends and may be lifted out when the upper pressure housing is removed. Ball bearings are provided for supporting the bevel gears and the pinion gear. The rack engages the pinion, and is held in proper engagement with the pinion by the backup rollers which carry the load due to gear tooth reactions. The gear assembly is attached to a stainless steel tube supported by the upper part of the pressure housing. This tube also carries and positions the guide tube which surrounds the rack. The rack is a tube with gear teeth on one side of its outer surface and flats on the opposite side which form a contact surface for guide rollers. Flats are cut on two opposite sides of the rack tube for forming the rack teeth and for a contact surface for the backup rollers. The upper end of the rack is fitted with an enlarged section which runs in the guide tube and provides lateral support for the upper end of the rack. It also acts as a piston in controlling water flow in the lower guide tube dashpot. The top section also carries a permanent magnet which is used to operate a rod position indicator outside the pressure housing. The load on the guide tube is transferred through a connection at its upper end to the support tube, then to the pressure housing. The support for the guide tube contains an energy absorber at the top end of the tube which deforms to limit the stresses on the tie rod, connector shaft and control rod in case the mechanism is scrammed without water in the dashpot. If such a "dry scram" should occur, the mechanism and control rod would not be damaged; however, it would be necessary to disassemble the drive and replace the guide energy absorber.

Motor Drive Package

Power to operate the drive is supplied by a synchronous, fractional horse-power, 120-volt, single-phase, 60-hertz motor. Since system frequency varies by less than 0.05%, the motor speed changes during operation are considered insignificant. The output is coupled to the vertical drive shaft through a magnetic clutch and an antireverse clutch operating in parallel. When the magnetic clutch is energized, the drive motor is connected to the main shaft and can drive the rod either up or down. With the magnetic clutch de-energized, the rod will drop due to its own weight. The motor shaft is fitted with an electrically operated brake which is connected to release the brake when the motor is energized. When the motor is de-energized, the brake is set by means of springs. This brake prevents driving except by means of the motor and thus holds the drive and control rod in position. The magnetic clutch, when de-energized, separates the drive between the pinion gear and the brake, thus permitting the rod to drop. The antireverse clutch and the brake prevent rotation of the drive in the up direction, and hold the control rod in position against upward forces on the control rod. This action is completely mechanical and does not rely on any outside source of power. The motor, brake, clutches, position indicator and limit switches are all mounted on a common frame for maintaining position and alignment. This entire drive package is assembled and checked as a unit and can be removed and replaced without disturbing the other parts of the mechanism. The frame for the drive package is provided with a flange which is bolted to a flange on the pressure housing for positioning the drive assembly. The electrical connections are located at the top of the drive package and are readily accessible.

The control rod drive mechanism clutch assemblies experienced many early operational problems due to excessive internal friction. A modification was necessary to reduce this friction and improve reliability. The lower jaw face of the clutch assemblies were chrome plated and the sliding spline replaced with a convoluted bellows.

Position Readout Equipment

Two independent position readout systems are provided for indicating the position of the control rod. One (primary system) is a synchrotransmitter geared to the main drive shaft with readout provided by synchroreceivers connected to the transmitter. The other (secondary system) position indicator consists of a series of accurately located reed switches built into a subassembly which is fastened to the outside of the CRDM along the pressure housing. The permanent magnet built into the top of the rack actuates the reed switches one at a time as it passes by them. An appropriate resistor network and above-mentioned servo actuate the readouts to position indication. Limit switches located in the motor drive package are gear driven from the shaft and are used to provide indication of rod position at certain predetermined points. Two of these switches are used as limit switches on the drive system and indicate the fully withdrawn and inserted positions. Other switches are provided which may be adjusted to actuate at intermediate points in the travel. The functions of these switches are described in Chapter 7.

Control Rod Disconnect

The control rod is connected to the drive mechanism by means of an extension shaft with a bayonet-type coupling at its lower end. A tie rod connects the extension shaft to the rack. In order to disengage the rod from the drive, it is necessary to remove the flange closure at the extreme upper end of the drive. A tool is then inserted through this opening and, with the drive in the full down position, the tool is used to release the nut locking device and to unscrew the nut from the tie rod. By turning another handle on the tool, the tie rod and bayonet coupling are rotated about a quarter turn to disengage the CRDM extension from the control rod.

CRDM Evaluation

The pressure containing members of the CRDM are considered to be extensions of the reactor vessel with the same operating and accident load capabilities. They are designed and fabricated in accordance with the ASME Pressure Vessel Code, Section III, Class A (Class 1 for the replacement components).

Additionally, each CRDM pressure housing is hydrostatically tested in accordance with this code to verify its structural integrity.

Development models of internal and external drive components, subassemblies of the CRDM, as well as a complete model CRDM have undergone accelerated life tests under reactor conditions and have demonstrated that the CRDM fulfills all drive, trip and endurance requirements.

In addition to these development tests, a prototype CRDM with a simulated reactor core module was accelerated life tested in an autoclave under reactor conditions to prove the overall adequacy of the CRDM during its design life. Each CRDM manufactured will be tested at design pressure to prove its functional adequacy.

3.3.4.3 Core Mechanical Design

The core approximates a right circular cylinder with an equivalent diameter of 136.7 inches and an active height of approximately 132 inches. It is made up of 204 fuel bundles with each bundle typically carrying 216 fuel rods. The core contains approximately 85 metric tons of slightly enriched uranium in the form of sintered uranium dioxide pellets encapsulated in Zircaloy or M5® fuel rods. The fuel is managed in a three- or four-batch mixed-zone refueling pattern with 52-76 fuel bundles in each new batch. Shield assemblies may be used for up to 6 cycles. A fuel loading pattern is chosen so as to minimize the fast neutron flux on the reactor vessel beltline materials.

Short-term reactivity control is provided by 41 cruciform control rods, 1 for every 4 nonperipheral fuel bundles. Four other control rods contain short-length poison modules on the lower end of the blade. The control rods, which have no followers, are guided within the core by a system of guide bars that are integral parts of the fuel bundles. Each fuel bundle has two guide bars along each side.

Fuel Bundle

Figures 3-14 and 3-15 show a typical reload fuel bundle which consists of a square (15 by 15) array of 225 positions: 216 fuel rods, 8 Zircaloy-4 guide bars, and 1 Zircaloy-4 or M5® instrument tube. For a gadolinia assembly, typically 3 to 16 of the fuel rods would contain gadolinia mixed with the fuel. Fuel rods may be replaced with solid stainless steel rods to allow the bundle to serve a reactor vessel fast neutron shielding function when placed on the core periphery. Fluence reduction methods have been used since cycle 8. Fuel rods adjacent to control blade positions may have a longer upper end cap and a shorter actual rod. Table 3-2 provides further fuel bundle component descriptions.

Analyses in Reference 1 have shown that reload High Thermal Performance fuel assemblies are designed for irradiation to assembly discharge exposure levels of 52,500 MWD/MTU. Reference 14 confirms this design while incorporating debris resistant design features. The large break LOCA analysis in Reference 50 provides justification to support operation up to a peak rod average exposure of 62,000 MWD/MTU. The exposure limits allow the Plant to operate approximately 18-month cycles with a 3-batch, 1/3 core refueling plan.

Reload Q and later assemblies are allowed to reach 58,900 MWD/MTU. However, Reload Q assemblies burned past 55,200, R and later assemblies burned past 52,000 MWD/MTU, may require the upper tie plate to be replaced prior to their last cycle of operation due to assembly irradiation growth (Reference 6). Based on a revised irradiation growth indicator, Reload Q and later assemblies can be irradiated to a burnup of 58,900 MWD/MTU.

The guide bars are solid Zircaloy-4 rods with threaded ends. They are located on the perimeter of the fuel bundle and serve three main functions. First, they serve a structural function. The zircaloy spacer grids are welded to the guide bars at equally spaced intervals and the end fittings are joined to the threaded end of the guide bars with cap screws. Second, they provide a guiding surface for the control rods. The guide bars protrude beyond both the fuel rods and perimeter strip of the spacer grids so that a control rod contacts only the guide bars. Third, they provide guiding surfaces which facilitate refueling and protect fuel rods from damage.

The tie plates and guide bars are connected with Inconel cap screws. The cap screws are torqued during cage assembly. This results in an initial tensile stress that depends upon the initial torque value and coefficient of friction. The minimum value is above the maximum load which could be exerted on the joint due to differential thermal expansion between the fuel rods and guide bars.

The upper and lower tie plates position the fuel bundle between the core support plate and the upper alignment plate. Both tie plates are of cast CF3 stainless steel and contain flow slots and the upper tie plate has a hole for the incore guide tube. In addition, the upper tie plate serves as a lifting fixture. The lower tie plate contains two locating pins which fit into corresponding holes in the core plate. The upper end of the fuel bundle is aligned relative to the core plate by two pins in the upper alignment plate which engage corresponding precision bored holes in the upper tie plate. Positive positioning of the bundle in this manner prevents the bundle from twisting, thereby maintaining the control rod channel opening. It also maintains the proper positioning of the bundle under DBA loading. The outer edges of the lower tie plate serve as a guiding surface during installation or removal of a fuel bundle from the core. Beginning with Reload N, the lower tie plates have been reduced in height to incorporate debris resistant design features, yet keep the overall fuel assembly height unchanged. In addition starting with Reload R, the assemblies will have a FUELGUARD™ grid attached to the underside of the lower tie plate as the debris resistant design feature.

The grid spacers (see Figure 3-17) maintain the fuel rod pitch over the full length of the bundle. The grids are fabricated in two different designs from Zircaloy-4 strips joined in an "eggcrate" fashion and welded together. The fuel rods are supported at ten axial locations using arched flow channels. With the arched flow channel design, four-channel sides with elongated contact areas provide symmetric lateral support to the fuel rod. The axial spacing at the grids prevents excessive lateral bowing of the rod span between grids. The channels have been designed to be flexible enough to elastically accommodate manufacturing tolerances and imposed deflection during assembly and operation. The flow channel design grid provides additional strength and improves thermal performance. The original spring-rigid dimple design was last used in Cycle 11.

The adequacy of the grid spacers has been established by an extensive test program. Fretting characteristics of fuel rods and spring-rigid dimple spacers were evaluated from a flow test made at maximum reactor flow conditions with no sign of any fretting corrosion. In addition, a production spacer was welded to eight fixed guide bars and prototypic cyclic and steady-state loads were applied to the grid cells through coil springs. The high thermal performance spacers have also undergone extensive flow testing, including levitation tests, a 500-hour fretting test, and pressure drop tests. These tests were conducted using a full-scale model of the fuel assembly in the hydraulic test facility of the fuel vendor. Tests indicate superior fretting resistance compared to the spring-rigid dimple design.

One demonstration assembly of Batch R design incorporates all-zircaloy HTP spacers in the top and bottom locations. All other Batch R and all S assemblies use inconel high thermal performance spacers in the bottom location and bimetallic spacers in the top location. The remainder of the assemblies in the core (Batch Q and earlier) have bimetallic spacers in the top and bottom locations. Batch T and beyond use zircaloy spacers at all axial locations.

Tests of the spacer side plate guide bar welds indicated ultimate strengths of the spacer side plate. This strength far exceeds the requirements of the spacer guide bar joint.

Fuel Rod

Figures 3-18 and 3-19 show typical fuel rods which consist of a stack of UO_2 pellets approximately 132 inches in length with a compression spring at the top end all clad within Zircaloy-4 or M5® tubing and sealed by welding end caps to each end (refer to Table 3-2 for dimensional characteristics). The atmosphere within the rods is pressurized helium. This pressure will assure that the fuel rod cladding will be free-standing under all anticipated reactor operating conditions. A plenum is provided at the top of the fuel column to accommodate the gaseous products released from the fuel and to accommodate the axial expansion of the fuel column. The compression spring is located within the plenum to maintain a compact fuel column. For batches “N” through “Q” assemblies, the fuel rods have been modified to accommodate debris resistant design features that included a longer solid lower end cap. Stress analysis conducted on this configuration resulted in lower stresses than with the prior design (Reference 14). The solid end cap, combined with a lowered bottom spacer grid, is designed to trap debris at a location in the bottom of the assembly where fretting would not affect the fission product barrier integrity. Exterior dimensions and active fuel zones of the assemblies are not affected by the changes.

One Batch R demonstration assembly had 19 rods that were modified to demonstrate acceptability of using an HTP spacer in the bottom and top locations. An elongated solid lower end cap with a reduced diameter was used to provide an initial gap between the spacer spring and the end cap. The purpose of the gap is to maximize the potential for fretting wear without affecting the integrity of the fuel rod. In addition, beginning with Batch R assemblies, the active fuel length was increased by 0.8 inch to 132.6 inches and do not use the longer solid lower end cap because of the FUELGUARD™ grid.

One demonstration assembly of Batch S design has been provided with 14 high density fuel rods to demonstrate the acceptability of pellets manufactured with a nominal density of 97% theoretical density. Beginning with Batch T, all spacers will be the zircaloy HTP design. The theoretical density for Batch T fuel is 95.85%. The theoretical density for Batch Y and beyond is 96%.

For Batch W and beyond, each non-shield assembly incorporates 21 rods which contain a long upper end cap (LUEC). This long upper end cap replaces the top 10.6 inches of active fuel. This reduces the active fuel height to 122.0 inches in these rods. Batch W and AA shield assemblies incorporate 18 such rods.

Reactor Vessel Weld Shield Assemblies

A combination of third and fourth burned fuel assemblies and shield fuel assemblies are used on the peripheral flats of the core to shield the welds of the reactor vessel.

Inert Rods

Inert rods consist of a stack of solid M5®, Zircaloy-2, Zircaloy-4 or stainless steel round bar stock sections and a compression spring at the top, all clad with M5® or Zircaloy-4 tubing and sealed by welding end caps to each end (refer to Table 3-2 for dimensional characteristics). The atmosphere within the rods is pressurized helium. Inert rods may also include solid stainless steel rods.

Inert rods are used to repair assemblies that contain failed fuel rods in order to reduce reactor coolant activity levels in subsequent cycles. The criteria for inserting inert rods in burned fuel assemblies is that the inert rod can not cause an increase in the assembly peaking factor. To accomplish this, several rod shuffles may be needed within the assembly, rather than a one-for-one exchange.

Control Rod Design

The control rod shown in Figure 3-20 consists of 32 stainless steel clad poison modules and a hanger section. The modules and hanger section are electron beam welded together to form a cruciform blade with a 12.250-inch span in the absorber segment and a total length of 151 inches, including the hanger section.

Each module contains a 131-inch length of absorber material of 80 wt% silver, 15 wt% indium, 5 wt% cadmium and is clad with 0.020-inch-thick 304 stainless steel. The module cross section is 0.750 inch wide by 0.180 inch thick. End caps are welded to the ends of each module and inspected to ensure integrity.

The hanger section provides a means for handling the blade and for coupling the blade to the CRDM extension shaft. A hanger section is a welded assembly fabricated from a 0.180-inch-thick 304 stainless steel lower section and a 0.312-inch-thick 348 stainless steel upper section.

Four of the 45 control rods contain Ag-In-Cd modules reduced in length to 31 inches. The length of the lower hanger section has been increased proportionately so that the overall length is the same as the rods containing full-length poison modules. Since the stainless steel in the lower hanger section will also act as a neutron absorber, its span is reduced from 12.25 inches to 5.8 inches so that the Ag-In-Cd section will have a higher worth relative to the lower hanger section. The lower hanger section extends

beyond the first guide bar of each fuel bundle making up the control rod channel.
The control rod assembly can accept a 15,000 pound tensile load in the event it is subjected to a dry scram. Under normal operation, the control rod buffering device located in the CRDM reduces the maximum load at the control rod coupling to less than 4,000 pounds for a rod-scram condition.
Control Rod Evaluation

Physical tests have been performed on poison modules, poison modules to hanger sections, and the hanger section at room temperature and operating temperature. In all cases, the tensile test results show the actual components to have higher strength values than the calculated values. Waterlogging experiments on poison modules with simulated clad defects show that no clad swelling occurred under normal depressurization conditions. Under a rapid depressurization transient, only minor clad swelling occurred which would not influence scram times. The thermal distortion tests indicated the poison modules are dimensionally stable.

Further bending, torsion, compression, tension and thermal bowing tests were performed on a prototype control rod to verify the design calculations. A destructive pull test of the control rod coupling connection and a nonbuffered control rod drop was made.

The guidance system for a followerless control rod has been adequately demonstrated in a series of tests in which a control rod was dropped within a four fuel bundle arrangement under flow conditions. The tests were performed in a cold loop with various water velocities along the blade. The guidance system was misaligned in excess of twice the permissible misalignment without impairing rod drop time. The control rod channel was reduced 0.025 inch below the nominal control rod blade thickness again without affecting rod drop time. This test clearly demonstrated the ability of the control rod to drop readily at any elevation even in a channel whose width is 0.147 inch less than the minimum permissible channel width. The fuel bundles used in this phase of the test program were about 1-1/8 inches less in width and about 2 feet shorter in length than a prototype fuel bundle. The test control rod was 0.020 inch thicker, 1-1/4 inches less in span and 2 feet shorter than the prototype control rod. The dimensional difference between the test components and reactor components results in a conservative test since the overall guidance system is less flexible. The above control rod drop tests were repeated at reactor operating conditions with prototype components and under adverse flow location, tolerance and thermal bowing conditions.

Source Design

Prior to Cycle 12, up to four neutron source assemblies were installed in the reactor to serve as sustainer sources for future start-up service. During the 1993 refueling outage, two of the four sources were transferred to the spent fuel pool. The final two were moved to the spent fuel pool during the 1995 refueling outage. Sustainer sources will no longer be used in the reactor. The sustainer source material is antimony-beryllium. The source pins are stored in the instrument guide tubes of the selected assemblies in the spent fuel pool.

The neutron source rods employ Type 304 stainless steel cladding material with a 0.34-inch OD and a 0.024-inch wall thickness. The sustainer sources contain 72 inches of Sb-Be pellets.

The cladding is of a freestanding design. The internal pressure is always less than reactor operating pressure. Internal gaps and clearances are provided to allow for differential expansion between the source material and cladding.