4.2 MECHANICAL DESIGN

The plant conditions for design are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public: Condition I - Normal Operation; Condition II - Incidents of Moderate Frequency; Condition III - Infrequent Incidents; Condition IV - Limiting Faults.

The reactor is designed so that its components meet the following performance and safety criteria:

1. The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the Reactor Control, Protection, and Emergency Cooling Systems (when applicable) assure that:

   a. Fuel damage* is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the Plant Cleanup System and are consistent with the plant design bases.

   b. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged* although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

   c. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

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* Fuel damage as used here is defined as penetration of the fission product barrier (i.e., the fuel rod clad).

2. The fuel assemblies are designed to accommodate expected conditions for design for handling during assembly inspection and refueling operations and shipping loads.

3. The fuel assemblies are designed to accept control rod insertions in order to provide the required reactivity control for power operations and reactivity shutdown conditions.

4. All fuel assemblies have provisions for the insertion of in-core instrumentation necessary for plant operation.

5. The reactor internals, in conjunction with the fuel assemblies, direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements can be met for all modes of operation. In addition, the internals provide core support and distribute coolant flow to the pressure vessel head so that the temperature differences between the vessel flange and head do not result in leakage from the flange during the Condition I and II modes of operation. Required inservice inspection can be carried out as the internals are removable and provide access to the inside of the pressure vessel.

4.2.1 Fuel

4.2.1.1 Design Bases

The fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 4.2 and specific criteria noted below. The same design bases apply to the 17 x 17 standard (STD), 17 x 17 Vantage 5H, 17 x 17 Vantage+ and 17 x 17 Standard Robust Fuel Assembly (RFA) designs.

4.2.1.1.1 Fuel Rods

The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas releases, and excessive cladding stresses and strains. This is achieved by designing the fuel rods so that the following conservative design bases are satisfied during Condition I and Condition II events over the fuel lifetime:

1. Fuel Pellet Temperatures - The center temperature of the hottest pellet is to be below the melting temperature of the UO_2 (melting point of 5080°F(1) unirradiated and reducing by 58°F per 10,000 MWD/MTU). While a limited amount of center melting can be tolerated, the design conservatively precludes center melting. A calculated centerline fuel temperature of 4700°F has been selected as an overpower limit to assure no fuel melting. This provides sufficient margin for uncertainties, as described in Sections 4.4.1.2 and 4.4.2.10.1.

2. Internal Gas Pressure - The internal gas pressure of the lead rod in the reactor will be limited to a value below that which would cause (a) the diametral gap to increase due to outward clad creep during steady-state operation, and (b) extensive departure from nucleate boiling (DNB) propagation to occur.

3. Clad Stress - The effective clad stresses are less than that which would cause general yield of the clad. While the clad has some capability for accommodating plastic strain, the yield strength has been accepted as a conservative design basis.

4. Clad Tensile Strain - The clad tangential strain range is less than one percent. The clad strain design basis addresses slow transient strain rate mechanisms where the clad effective stress never reaches the yield

strength due to stress relaxation. The 1 percent strain limit has been established based upon tensile and burst test data from irradiated clad. Irradiated clad properties are appropriate due to irradiation effects on clad ductility occurring before strain-limiting fuel clad interaction during a transient event can occur.

5. Strain Fatigue - The cumulative strain fatigue cycles are less than the design strain fatigue life. This basis is consistent with proven practice.

Radial, tangential, and axial stress components due to pressure differential and fuel clad contact pressure are combined into an effective stress using the maximum-distortion-energy theory. The von Mises' criterion is used to evaluate if the yield strength has been exceeded. The von Mises' criterion states that an isotropic material under multiaxial stress will begin to yield plastically when the effective stress (i.e., combined stress using maximum-distortion-energy theory) becomes equal to the material yield stress in simple tension as determined by an uniaxial tensile test. Since general yielding is to be prohibited, the volume average effective stress determined by integrating across the clad thickness increased by an allowance for local nonuniformity effects before it is compared to the yield strength. The yield strength correlation is that appropriate for irradiated clad since the irradiated properties are attained at low exposure whereas the fuel/clad interaction conditions which can lead to minimum margin to the design basis limit always occurs at much higher exposure.

The detailed fuel rod design established such parameters as pellet size and density, clad-pellet diametral gap, gas plenum size,

and helium pressure. The design also considers effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup.

Irradiation testing and fuel operational experience has verified the adequacy of the fuel performance and design bases. This experience and testing are discussed in References 2 and 3. Fuel experience and testing results, as they become available, are used to improve fuel rod design and manufacturing processes and assure that the design bases and safety criteria are satisfied.

The safety evaluation of the fuel rod internal pressure design basis is presented in Reference 4.

4.2.1.1.2 Fuel Assembly Structure

Structural integrity of the fuel assemblies is assured by setting limits on stresses and deformations due to various loads and by determining that the assemblies do not interfere with the functioning of other components. Three types of loads are considered.

1. Nonoperational loads such as those due to shipping and handling

2. Normal and abnormal loads which are defined for Conditions I and II

3. Abnormal loads which are defined for Conditions III and IV.

These criteria are applied to the design and evaluation of the top and bottom nozzles, the guide thimbles, the grids, and the thimble joints.

The design bases for evaluating the structural integrity of the fuel assemblies are:

1. Nonoperational 4g axial and 6g lateral loading with dimensional stability.

2. Normal Operation (Condition I) and Incidents Moderate Frequency (Condition II).

For the normal operating (Condition I) and upset conditions (Condition II), the fuel assembly component structural design criteria are classified into two material categories, namely, austenitic steels and Zircaloy. The stress categories and strength theory presented in the ASME Boiler and Pressure Vessel Code, Section III, are used as a general guide. The maximum shear-theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic steel components. The stress intensity is defined as the numerically largest difference between the various principal stresses in a three-dimensional field. The allowable stress intensity value for austenitic steels, such as nickel-chromium-iron alloys, is given by the lowest of the following:

a. 1/3 of the specific minimum tensile strength or 2/3 of the specified minimum yield strength at room temperature

b. 1/3 of the tensile strength or 90 percent of the yield strength at temperature but not to exceed 2/3 of the specified minimum yield strength at room temperature.

The stress limits for the austenitic steel components follow:

Stress Intensity Limits

Categories	Limit
General Primary Membrane Stress Intensity	Sm
Local Primary Membrane Stress Intensity	1.5 Sm
Primary Membrane plus Bending Stress Intensity	1.5 Sm
Total Primary plus Secondary Stress Intensity	3.0 Sm

The Zircaloy and ZIRLO™ structural components which consist of guide thimble and fuel tubes are in turn subdivided into two categories because of material differences and functional requirements. The fuel tube design criteria are covered separately in Section 4.2.1.1.1. The maximum stress theory is used to evaluate the guide thimble design. The maximum stress theory assumes that yielding due to combined stresses occur where one of the principal stresses are equal to the simple tensile or compressive yield stress. The Zircaloy and ZIRLO™ unirradiated properties are used to define the stress limits.

Abnormal loads during Conditions III and IV - worst case represented by combined seismic and blowdown loads.

1. Deflections of components cannot interfere with the reactor shutdown or emergency cooling of the fuel rods.

2. The fuel assembly component stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Pressure Vessel Code Section 3. Since the current analytical methods utilize elastic analysis, the stress allowables are defined as the smaller value of 2.4 Sm or 0.70 Su for primary membrane and 3.6 Sm or 1.05 Su for primary membrane plus primary bending. For the austenitic steel fuel assembly components, the stress intensity is defined in accordance with the rules described in the previous section for normal operating conditions. For the Zircaloy and ZIRLO™ components the stress limits are set at two-thirds of the material yield strength, Sy, at reactor operating temperature. This results in Zircaloy stress intensity limits being the smaller of 1.6 Sy or 0.70 Su for primary membrane and 2.4 Sy or 1.05 Su for primary membrane plus bending. For conservative purposes, the Zircaloy and ZIRLO™ unirradiated properties are used to define the stress limits. The grid component strength criteria are based on experimental tests. The grid component strength criterion is based on the lower 95 percent confidence level on the true mean from distribution of grid crush strength data at temperature.

4.2.1.2 Design Description

Fuel assembly and fuel rod design data are given in Tables 4.1-1 and 4.3-1. Two hundred sixty-four fuel rods, twenty-four guide thimble tubes, and one instrumentation thimble tube are arranged within a supporting structure to form a fuel assembly. The instrumentation thimble is located in the center position and provides a channel for insertion of an in-core neutron detector if the fuel assembly is located in an instrument core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, a neutron source assembly, a burnable absorber assembly or a plugging device (if used), depending on the position of the particular fuel assembly in the core. Figure 4.2-1 shows a cross section of a fuel assembly array, and Figure 4.2-2 shows a standard fuel assembly full length view. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles.

The design changes from the 17 x 17 STD design to the Vantage 5H design include reduced guide thimble and instrumentation tube diameters, and replacement of the six intermediate (mixing vane) Inconel grids with Zircaloy grids. The debris filter bottom nozzle (DFBN) design has been incorporated into the Vantage 5H fuel assembly. The DFBN is similar to the standard bottom nozzle design except that it is thinner and has a new pattern of smaller flow holes. The DFBN helps to minimize passage of debris particles that could cause fretting damage to fuel rod cladding.

The Vantage+ assembly skeleton is identical to that previously described for Vantage 5H except for those modifications necessary to accommodate intended fuel operation to higher burnup levels. The Vantage+ assembly skeleton is made of low cobalt material and the spring height is slightly increased for the reduction in fuel assembly height. The modifications consist of the use of ZIRLO™ guide thimbles as necessary and small skeleton dimensional alterations to provide additional fuel assembly and rod growth space at the extended burnup levels. The Vantage+ fuel assembly is 0.200 inch shorter than the Vantage 5H assembly. The grid centerline elevations of the Vantage+ are identical to those of the Vantage 5H assembly, except for the top grid. The Vantage+ top grid has been moved down by the same 0.200 inch. The RFA assembly provides further design enhancements from the Vantage+ design including modified Low Pressure Drop mid-grids and IFMs, thicker guide tubes, and a protective bottom grid. The RFA continues to utilize the ZIRLO™ fuel rods and skeletons.

However, since the Vantage+ and RFA fuel are intended to replace either the Westinghouse LOPAR or Vantage 5H, their assembly exterior envelope is equivalent in design dimensions, and the functional interface with the reactor internals is also equivalent to those of previous Westinghouse fuel assembly designs. Also, the Vantage+ and RFA are designed to be mechanically and hydraulically compatible with the LOPAR and Vantage 5H, and the same functional requirements and design criteria as previously established for the Westinghouse Vantage 5H fuel assembly remain valid for the Vantage+ and RFA (References 18, 21). The fuel rod clad oxidation and hydriding design criteria for ZIRLO cladding has been updated in Reference 27. The Vantage+ and RFA design parameters are provided in Table 4.1-1. Figure 4.2-2A compares the Vantage+ and the Vantage 5H fuel assembly designs. The RFA design is shown in Figure 4.2-2B.

The feed fuel loaded into Salem Unit 1 (starting with Regions 17A & 17B) and Salem Unit 2 (starting with Regions 14A & 14B) contains further enhancements to the RFA fuel assembly design in order to reduce performance limitations associated with rod internal pressure. The design changes from the 17 x 17 RFA design to the 17 x 17 RFA ZIRLO™+2 design include: an increased fuel rod length, increased fuel assembly length and elimination of the external grip top end plug. The RFA ZIRLO™+2 fuel assembly is 0.2 inches longer than the RFA assembly due to an increase in thimble and guide tube lengths. The small dimensional alterations to the skeleton provide rod growth space for the longer fuel rods. The grid centerline elevations of the RFA ZIRLO™+2 design are identical to those of the RFA fuel assembly, except for the top grid. The RFA ZIRLO™+2 top grid has been moved upward by 0.2 inches. The top nozzle spring height is slightly decreased due to the increase in overall assembly height. The RFA ZIRLO™+2 assembly design is shown in Figure 4.2-2C.

The feed fuel loaded into Salem Unit 1 (starting with Regions 19A and 19B) and Salem Unit 2 (starting with Regions 17A and 17B) contains further enhancements to the RFA mid grid design in order to further reduce grid to rod fretting. The design changes from the RFA mid grid to the RFA-2 mid grid are changes to the spring window cut-outs and the spring and dimple contact areas. Compared to RFA mid grid, RFA-2 mid grid offers improved resistance to fuel rod fretting without significantly affecting any other thermal-hydraulic or mechanical performance. Both RFA and RFA-2 mid grids are made from ZIRLO™ material and both mid grids have the same mass and volume.

Each fuel assembly is installed vertically in the reactor vessel and stands upright on the lower core plate, which is fitted with alignment pins to locate and orient the assembly. After all fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears downward against the fuel assembly top nozzle via the holddown springs to hold the fuel assemblies in place.

4.2.1.2.1 Fuel Rods

The Vantage+ and RFA fuel rod designs represent a modification to the Vantage 5H fuel rod in cladding of ZIRLO™ as compared to Zircaloy-4. ZIRLO™ is a zirconium alloy similar to Zircaloy-4, which has been specifically developed to enhance corrosion resistance. The Vantage 5H, Vantage+, and RFA fuel rods contain enriched uranium dioxide fuel pellets, and Integral Fuel Burnable Absorber (IFBA) coating on some of the fuel pellets. Additionally, the RFA fuel rods contain axial blankets (top and bottom 6 inches) of annular pellets. The annulus represents a 25% void by volume of the pellet. Schematics of the fuel rods are shown in Figures 4.2-3, 4.2-3A, and 4.2-3B.

The Vantage+ and RFA fuel rods have the same wall thickness as the Vantage 5H. The Vantage+ fuel rod length is shorter to provide the required fuel rod growth room. To offset the reduction in plenum length the Vantage+ fuel rod has a variable pitch plenum spring. The variable pitch plenum spring provides the same support as the Vantage 5H plenum spring, but with less spring turns which means less spring volume. The RFA fuel rod is longer than the Vantage+ and Vantage 5H designs but maintains sufficient room for fuel rod growth. This provides additional plenum length to accommodate fission gas release associated with high burnup. In addition, the RFA utilizes the variable pitch plenum spring and axial blankets of annular pellets to provide additional plenum margin. The 6 inches (top and bottom) of annular pellets effectively provide 3 additional inches of plenum volume beyond the current VANTAGE+ design. The bottom end plug has an internal grip feature to facilitate rod loading on the Vantage+, RFA, and Vantage 5H designs and provides appropriate lead-in for the removable top nozzle reconstitution feature. The RFA design also incorporates the external grip top end plug into its fuel rod design. This feature provides additional capability to reposition the fuel during manufacturing and will simplify reconstitution of the fuel rod from the top of the assembly. The Salem Vantage+ and RFA fuel rods also may posses a zirconium dioxide (ZrO_2) coating on the bottom outside surface of the fuel rod. This fuel feature may be incorporated as an option to be determined on a reload basis. The protective oxide coating covers the bottom end plug, the bottom end plug weldment and a portion of the cladding. A metallurgical-bonded layer of ZrO_2 uniformly covers a minimum of 4.5 inches of the bottom end of the fuel rod. The minimum coating length was chosen to ensure that the coating would extend through the top of the current bottom Inconel structural grid, independent of the fuel rod loading position or fuel assembly design. The coating, which is 2 to 6 microns thick, provides a hard, wear resistant, surface layer of ZrO_2 for additional debris damage resistance, thereby improving fuel reliability. This extra layer of oxide coating provides additional rod fretting wear protection. The RFA design incorporates the protective bottom grid and modified fuel rod bottom end plug. These enhanced debris mitigating features, described in Section 4.2.1.2.2, diminish the need for the oxide coating. Therefore, for core designs utilizing the RFA, the oxide coating may be incorporated as an option to be determined on a reload basis.

The RFA ZIRLO™+2 fuel rod design represents a modification to the RFA fuel rod in terms of increased fuel rod length and increased fuel assembly length. The RFA ZIRLO™+2 fuel rod length is 0.2 inches longer than the RFA fuel rod length. Sufficient fuel rod growth margin is accomplished by increasing the length of the instrument thimble and guide thimbles. The RFA ZIRLO™+2 design incorporates a shorter (0.12 inches) external top end plug without a gripper feature. There is no longer any functional purpose for this top end plug gripper feature as fuel rods can still be handled through the top of the assembly for reconstitution. Both the longer fuel rod and shorter external top end plug provide additional plenum length to accommodate fission gas release. The plenum spring free length was increased to accommodate the increase in fuel rod plenum length, while the spring rate was decreased to maintain the same fuel stack hold-down force as in the RFA fuel rod design. A schematic of the RFA ZIRLO™+2 fuel rod is shown in Figure 4.2-3C.

The Salem Vantage 5H, Vantage+, and RFA fuel uses a standardized fuel pellet design which is a refinement to the chamfered pellet design. The standard design helps to improve manufacturability while maintaining or improving performance (e.g., improved pellet chip resistance during manufacturing and handling).

The Vantage 5H, Vantage+, and RFA IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride

coating on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of IFBA rods within an assembly vary depending on specific application. The ends of the coated and uncoated pellets are dished to allow for greater axial expansion at the pellet centerline and void volume for fission gas release. To accommodate a six inch axial blanket length at the top and bottom of the Robust Fuel Assembly (RFA) fuel rods, the annular pellets have a longer length than the solid pellets to obtain an integer number of fuel pellets which will equal a six inch length.

To avoid overstressing of the cladding or seal welds, void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel density changes during burnup. Shifting the fuel within the cladding during handling or shipping prior to core loading is prevented by a stainless steel helical spring which bears on top of the fuel. At assembly the pellets are stacked in the cladding to the required fuel height, the spring is then inserted into the top end of the fuel tube and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process in order to minimize compressive clad stresses and creep due to coolant operating pressures. Fuel rod pressurization is dependent on the planned fuel burnup as well as other fuel design parameters and fuel characteristics (particularly densification potential).

4.2.1.2.2 Fuel Assembly Structure

The fuel assembly structure consists of a bottom nozzle, top nozzle, guide thimbles, and grids, as shown on Figures 4.2-2 and 4.2-2A.

Bottom Nozzle

The bottom nozzle is a box-like structure which serves as a bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The square nozzle is fabricated from Type 304 stainless steel and consists of a perforated plate and four angle legs with bearing plates as shown on Figure 4.2-2. The legs form a plenum for the inlet coolant flow to the fuel assembly. The plate itself acts to prevent a downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide tubes by locked screws which penetrate through the nozzle and mate with an inside fitting in each guide tube.

The debris filter bottom nozzle (DFBN) design was introduced into the Salem fuel assemblies to help reduce the possibility of fuel rod damage due to debris-induced fretting. The Vantage+ and RFA assemblies have a low cobalt

stainless steel DFBN. The DFBN design incorporates a modified flow hole size and pattern, as described below, and a decreased nozzle height and thinner top plate to accommodate the high burnup fuel rods. The DFBN retains the design reconstitution feature which facilitates easy removal of the nozzle from the fuel assembly.

The relatively large flow holes in a conventional bottom nozzle were replaced with a new pattern of smaller flow holes in the DFBN. The holes are sized to minimize passage of debris particles large enough to cause damage. The holes were also sized to provide sufficient flow area, comparable pressure drops, and continued structural integrity of the nozzle. Tests to measure pressure drop and demonstrate structural integrity have been performed to verify that the DFBN is totally compatible with the current design.

Salem Unit 2 Region 23 and Salem Unit 1 Region 26 are the first regions to implement the Standardized Debris Filter Bottom Nozzle (SDFBN) design into Salem RFA fuel assemblies. The SDFBN is designed to have a loss coefficient that is the same, independent of supplier. The SDFBN eliminates the side skirt communication flow holes as a means of improving the debris mitigation performance of the bottom nozzle. This nozzle meets all of the applicable mechanical and thermal-hydraulic design criteria. The RFA fuel assembly with the SDFBN is illustrated in Figure 4.2-2C.

Coolant flow through the fuel assembly is directed from the plenum in the bottom nozzle upward through the penetrations in the plate to the channels between the fuel rods. The penetrations in the plate are positioned between the rows of the fuel rods.

Axial loads (holddown) imposed on the fuel assembly and the weight of the fuel assembly is transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite bearing plates which mate with locating pins in the lower core plate. Any lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

Top Nozzle

The top nozzle assembly functions as the upper structural element of the fuel assembly in addition to providing a partial protective housing for the rod cluster control assembly or other components. It consists of an adapter plate, enclosure, top plate, and pads. The integral welded assembly has holddown springs mounted on the assembly as shown on Figure 4.2-2. The reconstitutable top nozzle (RTN) design contains hold-down springs and screws made of Inconel 718 and Inconel 600, respectively, whereas other components are made of Type 304 stainless steel. The feed fuel loaded into Salem Unit 2 (starting with Regions 14A and 14B) and Salem Unit 1 (starting with Regions 17A and 17B) contains reconstitutable top nozzles with hold-down screws made of Inconel 718, as opposed to Inconel 600, in order to increase resistance to stress corrosion cracking.

Vantage+, RFA, and Vantage 5H fuel assemblies use the reconstitutable top nozzle (RTN). The RTN design for the Vantage 5H fuel assembly differs from the conventional design in two ways: 1) a groove is provided in each thimble thru-

hole in the nozzle plate to facilitate attachment and removal, and; 2) the nozzle plate thickness is reduced to provide additional axial space for fuel rod growth. Additional details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in Section 2.3.2 in Reference 15.

A replacement reconstitutable top nozzle (RRTN) design may be used in a reload cycle to replace the original reconstitutable top nozzle (RTN) on an irradiated fuel assembly. The mechanical features of the RRTN are the same as those for the RTN (see Figure 4.2-2) with some minor dimensional differences in the top nozzle adapter plate thimble hole to facilitate attachment to an irradiated fuel assembly. The RRTN design contains hold-down springs and screws made of Inconel 718, whereas, other components are made of Type 304 stainless steel.

The square adapter plate is provided with round and obround penetrations to permit the flow of coolant upward through the top nozzle. Other round holes are provided to accept sleeves which are welded to the adapter plate and mechanically attached to the thimble tubes. The ligaments in the plate cover the tops of the fuel rods and prevent their upward ejection from the fuel assembly. The enclosure is a sheet metal shroud which sets the distance between the adapter plate and the top plate. The top plate has a large square hole in the center to permit access for the control rods and the control rod spiders. Holddown springs are mounted on the top plate and are fastened in place by screws and clamps located at two diagonally opposite corners. The clamps are attached to the nozzle by a specific arrangement of tack welds or tack weld(s) in combination with a stainless steel clamp screw, depending on the manufacturing process in place at the time a given fuel region was built. The spring screws apply a load directly to the base of the hold-down springs. The clamps do not have any bearing surfaces that load the spring to the nozzle, but primarily provide a stationary location for attachment of lock wires that prevent rotation of the spring screws. On the other two corners, integral pads are positioned which contain alignment holes for locating the upper end of the fuel assembly.

Guide and Instrument Thimbles

The guide thimbles are structural members which also provide channels for the neutron absorber rods, burnable poison rods, or neutron source assemblies. Each one is fabricated from Zircaloy-4 or ZIRLO™ tubing having two different diameters. The larger diameter at the top provides a relatively large annular area to permit rapid insertion of the control rods during a reactor trip as well as to accommodate the flow of coolant during normal operation. Four holes are provided on the thimble tube above the dashpot to reduce the rod drop time. The lower portion of the guide thimbles has a reduced diameter to produce a dashpot action near the end of the control rod travel during normal operation and to accommodate the outflow of water from the dashpot during a reactor trip. The dashpot is closed at the bottom by means of an end plug which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation. The top end of the guide thimble is fastened to a tubular insert by three expansion swages. The insert engages into the top nozzle and is secured into position by the lock tube. The lower end of the guide thimble is fitted with an end plug which is then fastened into the bottom nozzle by a locked screw.

Fuel rod support grids are fastened to the guide thimble assemblies to create an integrated structure. Since welding of the Inconel grid and Zircaloy thimble is not possible, the fastening technique depicted on Figures 4.2-5 and 4.2-9 is used for all but the top and bottom grids in a fuel assembly.

An expanding tool is inserted into the inner diameter of the Zircaloy or Zirlo™ thimble tube to the elevation of the zircaloy sleeves that have been welded to the Zircaloy middle grid assemblies. The four-lobed tool forces the thimble and sleeve outward to a predetermined diameter, thus joining the two components.

The top grid-to-thimble attachment for the Vantage 5H, Vantage+, and RFA design is shown on Figure 4.2-7. The Zircaloy or ZIRLO™ thimbles are fastened to the top nozzle inserts by expanding the members as shown on Figure 4.2-7. The inserts then engage the top nozzle and are secured into position by the insertion of lock tubes.

The bottom grid assembly is joined to the fuel assembly as shown on Figure 4.2-11. The stainless steel insert is spot welded to the bottom grid and later captured between the guide thimble end plug and the bottom nozzle by means of a stainless steel thimble screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of Zircaloy guide thimbles in 1969.

The central instrumentation thimble of each fuel assembly is constrained by seating in counterbores in each nozzle. This tube is a constant diameter and guides the incore neutron detectors. This thimble is expanded at the top and mid grids in the same manner as the previously discussed expansion of the guide thimbles to the grids.

With the exception of an increased length above the dashpot, the Vantage+ guide thimbles are identical to those in the Vantage 5H design. For the RFA, the thimble tube thickness has been increased by 25% (4 Mils) relative to the V5H and Vantage+ designs. The thimble tube outer diameter of both the upper spans (major diameter section above the dashpot) and the dashpot spans have been increased. The inner diameter of the thimble tube in both the upper spans and in the dashpot spans are unchanged from the current designs. With the thicker guide thimble tube, the cross-sectional area is increased by 26% in the major diameter section and 29% in the dashpot section. The Vantage+, RFA and Vantage 5H guide thimble ID provides adequate clearance for the control rods and sufficient diametral clearance for burnable absorber rods and source rods.
The Vantage+, RFA and Vantage 5H instrumentation tube diameter has sufficient diametral clearance for the flux thimble to traverse the tube without binding.

Grid Assemblies

The fuel rods, as shown on Figures 4.2-2, 4.2-2A and 4.2-2B, are supported laterally at intervals along their length by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is afforded lateral support at six contact points within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps interlocked and welded in an "egg-crate" arrangement to join the straps permanently at their points of intersection. The straps contain spring fingers, support dimples, and mixing vanes.

The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

Up to four types of grid types are used in each fuel assembly: Mid-grids (structural grids with flow mixing vanes), Intermediate Flow Mixing (IFM) grids (non-structural grids with flow mixing vanes), top and bottom structural grids without mixing vanes, and the protective bottom grid (P-grid). Table 4.1-1 provides the breakdown of the grid types, number and grid material used in each of the fuel designs. Flow mixing vanes project from the edge of the inner grid strap into the coolant stream to promote mixing of the coolant in the high heat flux regions of the fuel assembly.

The IFMs are positioned at the mid-spans of the four uppermost mid-grids to further increase the flow turbulence in the axial zone where departure from nucleate boiling (DNB) is limiting. Each IFM grid cell contains four dimples, which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel contact with the mixing vanes.

For the RFA, the modified low pressure drop mid-grids and IFM grids are embossed to accept the larger diameter guide thimble tube.

The P-grid is a partial height grid similar in configuration to the mid-grid, but without mixing vanes. It is located between the bottom Inconel grid and the bottom nozzle, nearly on the surface of the bottom nozzle. The intersections of the inner straps of the P-grid align with the flow holes of the DFBN, effectively bisecting the flow path through the flow hole into four quarters. This provides an effective barrier against small debris. In conjunction with the P-grid, the fuel rod bottom end plug is changed to a longer design such that the portion of the fuel rod engaged in the P-grid and extending up past the top of the P-grid is solid end plug material. This provides a protective zone where trapped debris cannot fret through the fuel rod and cause a failure. The hydraulic effects of the P-grid are minimized by positioning the fuel rods 0.085 inches above the bottom nozzle. The combination of the lowered fuel rod position and longer fuel rod end plug results in no change to the axial fuel stack height from the previous Vantage+ fuel region.

For the application of the P-grid, the bottom Inconel grid was welded to 20 of the 24 inserts. The remaining four inserts are spot-welded to the P-grid at the four outermost corners on the grid diagonal.

Salem Unit 2 Region 23 and Salem Unit 1 Region 26 are the first regions to implement the Westinghouse Robust Protective Grid (RPG) which was developed as a result of observed failures of the P-grid in the field during Post Irradiation Exams (PIE) performed at several different plants. It was determined by Westinghouse that observed failures were the result of two primary issues; 1) fatigue failure within the protective grid itself at the top of the end strap and 2) stress corrosion cracking (SCC) primarily within the rod support dimples. The RPG implements design changes such as increasing the maximum nominal height of the grid, increasing the amount of material at the ends of the dimple window cutouts, increasing the radii of the dimple window cutouts, and the welding of four additional inserts for a total of 8 welded inserts out of the 24 total inserts in order to help better support the grid.

The nominal height of the grid was increased to allow "V-notch" window cutouts to be added to help minimize flow-induced vibration caused by vortex shedding at the trailing edge of the inner grid straps. Figure 4.2-2C shows RFA fuel with the increased nominal height due to implementation of the RPG. These design changes incorporated into the RPG design help address the issues of fatigue failures and failures due to stress corrosion cracking. These changes do not impact the thermal hydraulic performance of the RPG as there is no change to the pressure loss coefficient. In addition, the RPG retains the original protective grid function as a debris mitigation feature.

The outside straps on all the grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

During 1989, snag-resistant grids were introduced. These grids contain outer grid straps which are modified to help prevent assembly hangup from grid strap interference during fuel assembly removal. This was accomplished by changing the grid strap corner geometry and the addition of guide tabs on the outer grid strap.

4.2.1.3 Design Evaluation

4.2.1.3.1 Fuel Rods

The fuel rods are designed to assure the design bases are satisfied for Condition I and II events. This assures that the fuel performed, and safety criteria (Section 4.2.1.1) are satisfied.

Materials - Fuel Cladding

The desired fuel rod cladding is a material which has a superior combination of neutron economy (low absorption cross section), high strength (to resist deformation due to differential pressures and mechanical interaction between fuel and clad), high corrosion resistance (to coolant, fuel, and fission products), and high reliability. Zircaloy-4 and ZIRLO™ have this desired combination of cladding properties. There is considerable pressurized water reactor (PWR) operating experience on the capability of Zircaloy as a cladding material (2). Clad hydriding has not been a significant cause of clad perforation since current controls on fuel-contained moisture levels were instituted.

Metallographic examination of irradiated commercial fuel rods have shown occurrences of fuel/clad chemical interaction. Reaction layers of <1 mil in thickness have been observed between fuel and clad at limited points around the circumference. Westinghouse metallographic data indicates that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the layer and eventual clad penetration.

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out-of-reactor tests have shown that in the presence of high clad tensile stress, relatively large concentrations of iodine, or cadmium in solution in liquid cesium can stress corrode zirconium alloy tubing and lead to eventual clad cracking. Extensive post irradiation examination has

produced no conclusive evidence that this mechanism is operative in commercial fuel.

Materials - Fuel Pellets

Sintered, high density uranium dioxide fuel is chemically inert, with respect to the cladding, at core operating temperatures and pressures. In the event of cladding defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration although limited fuel erosion can occur. As has been shown by operating experience and extensive experimental work, the thermal design parameters conservatively account for changes in the thermal performance of the fuel elements due to pellet fracture which may occur during power operation. The consequences of defects in the cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile.

Observations from several operating Westinghouse PWRs (2) have shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent incomplete settling of the fuel pellets results in local and distributed gaps in the fuel rods. Fuel densification has been minimized by improvements in the fuel manufacturing process and by specifying a nominal 95.5 percent initial fuel density.

The effects of fuel densification have been taken into account in the nuclear and thermal-hydraulic design of the reactor described in Sections 4.3 and 4.4, respectively.

Materials - Strength Considerations

One of the most important limiting factors in fuel element duty is the mechanical interaction of fuel and cladding. This fuel- cladding interaction produces cyclic stresses and strains in the cladding, and these in turn consume cladding fatigue life. The

reduction of fuel-cladding interaction is therefore a principal goal of design. In order to achieve this goal and to enhance the cyclic operational capability of the fuel rod, the technology for using prepressurized fuel rods in Westinghouse PWRs has been developed.

Initially the gap between the fuel and cladding is sufficient to prevent hard contact between the two. However, during power operation a gradual compressive creep of the cladding onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Cladding compressive creep eventually results in hard fuel-cladding contact. During this period of fuel-cladding contact, changes in power level could result in significant changes in cladding stresses and strains. By using prepressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of cladding-creep toward the surface of the fuel is reduced. Fuel rod prepressurization delays the time at which substantial fuel-cladding interaction and hard contact occur and hence significantly reduces the number and extent of cyclic stresses and strains experienced by the cladding both before and after fuel-cladding contact. These factors result in an increase in the fatigue life margin of the cladding and lead to greater cladding reliability. If gaps should form in the fuel stacks, clad flattening will be prevented by the rod prepressurization so that the flattening time will be greater than the fuel core life.

Steady-State Performance Evaluation

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

1. Clad creep and elastic deflection

2. Pellet density changes, thermal expansion, gas release, and thermal properties as a function of temperature and fuel burnup

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

These effects are evaluated using an overall fuel rod design model (Reference 17) which include appropriate modifications for time dependent fuel densification. With these interacting factors considered, the model determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and cladding temperatures, and cladding deflections are calculated. The fuel rod is divided lengthwise into several sections and radially into a number of annular zones. Fuel density changes, cladding stresses, strains and deformations, and fission gas releases are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure. The initial rod internal pressure is selected to delay fuel/clad mechanical interaction and to avoid the potential for flattened rod formation. Clad flattening for Salem Nuclear Generating Station (SNGS) fuel is evaluated using the models described in Reference 7.

The gap conductance between the pellet surface and the cladding inner diameter is calculated as a function of the composition, temperature, and pressure of the gas mixture, and the gap size or contact pressure between clad and pellet. After computing the fuel temperature of each pellet annular zone, the fractional fission gas release is assessed using an empirical model derived from experimental data (17). The total amount of gas released is based on the average fractional release within each axial and radial zone and the gas generation rate which, in turn, is a function of burnup. Finally, the gas released is summed over all zones and the pressure is calculated.

The model shows good agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures, and clad deflections (17). Included in this spectrum are variations in power, time, fuel density, and geometry. The in-pile fuel

Temperature measurements' comparisons used are shown in Reference 17.

Typical fuel clad inner diameter and the fuel pellet outer diameter as a function of exposure are presented on Figure 4.2-4. The cycle-to-cycle changes in the pellet outer diameter represent the effects of power changes as the fuel is moved into different positions as a result of refueling. The gap size at any time is merely the difference between clad inner diameter and pellet outer diameter. Total clad-pellet surface contact occurs near the end of Cycle 2. The figure represents hot fuel dimensions for a fuel rod operating at the power level shown on Figure 4.2-6. Figure 4.2-6 illustrates representative fuel rod internal gas pressure and linear power for the lead burnup rod vs. irradiation time. In addition, it outlines the typical operating range of internal gas pressures which is applicable to the total fuel rod population within a region. The "best estimate" fission gas release model was used in determining the internal gas pressures as a function of irradiation time.

The clad stresses at a constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal gas pressure. Because of the prepressurization with helium, the volume average effective stresses are always less than ~10,000 psi at the pressurization level used in this fuel rod design. Stresses due to the temperature gradient are not included in this average effective stress because thermal stresses are, in general, negative at the clad inner diameter and positive at the clad outer diameter and their contribution to the clad volume average stress is small. Furthermore, the thermal stress decreases with time during steady-state operation due to stress relaxation. The stress due to pressure differential is highest in the minimum power rod at the beginning of life (BOL) (due to low internal gas pressure) and the thermal stress is highest in the maximum power rod (due to steep temperature gradient).

Tensile stresses could be created once the clad has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. As shown on Figure 4.2-4, there is very limited clad pushout after pellet-clad contact. Fuel swelling can result in small clad strains (< 1 percent) for expected discharge burnups, but the associated clad stresses are very low because of clad creep (thermal and irradiation-induced creep). Furthermore, the 1 percent strain criterion is extremely conservative for fuel-swelling driven clad strain because the strain rate associated with solid fission products swelling is very slow (~5×10^{-7} hr^{-1}). In-pile experiments (8) have shown that Zircaloy tubing exhibits "superplasticity" at slow strain rates during neutron irradiation. Uniform clad strains of greater than 10 percent have been achieved under these conditions with no sign of plastic instability.

Transient Evaluation Method

Pellet thermal expansion due to power increases is considered the only mechanism by which significant stresses and strains can be imposed on the clad. Power increases in commercial reactors can result from fuel shuffling (e.g., Region 3 positioned near the center of the core for Cycle 2 operation after operating near the periphery during Cycle 1), reactor power escalation following extended reduced power operation, and control rod movement. In

the mechanical design model, lead rods are depleted using best estimate power histories as determined by core physics calculations. During the depletion the amount of diametral gap closure is evaluated based upon the pellet expansion-cracking model, clad creep model, and fuel swelling model. At various times during the depletion, the power is increased locally on the rod to the burnup dependent attainable power density as determined by core physics calculations. The radial, tangential, and axial clad stresses resulting from the power increase are combined into a volume average effective clad stress.

The von Mises' criterion is used to evaluate if the clad yield stress has been exceeded. This criterion states that an isotropic material in multi-axial stress will begin to yield plastically when the effective stress exceeds the yield stress as determined by a uniaxial tensile test. The yield stress correlation is that for irradiated cladding since fuel/clad interaction occurs at high burnup. Furthermore, the effective stress is increased by an allowance, which accounts for stress concentrations in the clad adjacent to radial cracks in the pellet, prior to the comparison with the yield stress. This allowance was evaluated using a two-dimensional (r, e) finite element model.

Slow transient power increases can result in large clad strains without exceeding the clad yield stress because of clad creep and stress relaxation. Therefore, in addition to the yield stress criterion, a criterion on allowable clad positive strain is necessary. Based upon high strain rate burst and tensile test data on irradiated tubing, 1 percent strain was determined to be the lower limit on irradiated clad ductility and thus adopted as a design criterion.

In addition to the mechanical design models and design criteria, Westinghouse relies on performance data accumulated through transient power test programs in experimental and commercial reactors, and through normal operation in commercial reactors.

It is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor which is subjected to daily load follow is the failure of the cladding by low cycle strain fatigue. During their normal residence time in reactor, the fuel rods may be subjected to ~1000 cycles with typical changes in power level from 50 to 100 percent of their steady-state values.

The assessment of the fatigue life of the fuel rod cladding is subjected to a considerable uncertainty due to the difficulty of evaluating the strain range which results from the cyclic interaction of the fuel pellets and claddings. This difficulty arises, for example, from such highly unpredictable phenomena as pellet cracking, fragmentation, and relocation. Nevertheless, since early 1968, Westinghouse has been investigating this particular phenomenon both analytically and experimentally. Strain fatigue tests on irradiated and nonirradiated hydrided Zircaloy-4 claddings were performed which permitted a definition of a conservative fatigue life limit and recommendation of a methodology to treat the strain fatigue evaluation of the Westinghouse reference fuel rod designs.

However, Westinghouse is convinced that the final proof of the adequacy of a given fuel rod design to meet the load follow requirements can only come from in-pile experiments performed on actual reactors. The Westinghouse experience in load follow operation dates back to early 1970 with the load follow operation of the Saxton reactor. Successful load follow operation has been performed on reactor A (300 load follow cycles) and reactor B (150 load follow cycles). In both cases, there was no significant coolant activity increase that could be associated with the load follow mode of operation.

The following paragraphs present briefly the Westinghouse analytical approach to strain fatigue.

A comprehensive review of the available strain-fatigue models was conducted by Westinghouse as early as 1968. This included the

Langer-O'Donnell model (9), the Yao-Munse model, and the Manson-Halford Model.

Upon completion of this review and using the results of the Westinghouse experimental programs discussed below, it was concluded that the approach defined by Langer-O'Donnell would be retained and the empirical factors of their correlation modified in order to conservatively bound the results of the Westinghouse testing program.

The Langer-O'Donnell empirical correlation has the following form:

$$S_a = \frac{E}{4\sqrt{N_f}} \ln \frac{100}{100 - RA} + S_e$$

where:

S_a = 1/2 E $\Delta \varepsilon_t$ = pseudo - stress amplitude which causes failure in N_f cycles (lb/in^2)

$\Delta\varepsilon_t$ = total strain range (in/in)

E = Young's Modulus (lb/in^2)

N_f = number of cycles to failure

RA = reduction in area at fracture in a uniaxial tensile test (percent)

S_e = endurance limit (lb/in^2)

Both RA and S_e are empirical constants which depend on the type of material, the temperature, and the irradiation. The Westinghouse testing program was subdivided in the following subprograms:

1. A rotating bend fatigue experiment on unirradiated Zircaloy-4 specimens at room temperature and at 725°F.

Both hydrided and nonhydrided Zircaloy-4 cladding were tested.

2. A biaxial fatigue experiment in gas autoclave on unirradiated Zircaloy-4 cladding both hydrided and nonhydrided.

3. A fatigue test program on irradiated cladding from the CVTR and Yankee Core V conducted at Battelle Memorial Institute.

The results of these test programs provided information of different cladding conditions including the effect of irradiation, hydrogen level, and temperature.

The Westinghouse design equations followed the concept for the fatigue design criterion according to Section 3 of the ASME Boiler and Pressure Vessel code; namely:

1. The calculated pseudo-stress amplitude (S_a) has to be multiplied by a factor of 2 in order to obtain the allowable number of cycles (N_f).

2. The allowable cycles for a given S_a is 5 percent of N_f, or a safety factor of 20 on cycles.

The lesser of the two allowable number of cycles is selected. The cumulative fatigue life fraction is then computed as:

$$\sum_{1}^{k} \frac{n_k}{N_{fk}} \leq 1$$

where: n_k = number of diurnal cycles of mode k.

The potential effects of operation with waterlogged fuel are discussed in Section 4.4.3.6. Waterlogging is not considered to be a concern during operational transients.

4.2.1.3.2 Fuel Assembly Structure

Stresses and Deflections

The potential sources of high stresses in the assembly are avoided by the design. For example, stresses in the fuel rod due to thermal expansion and Zircaloy or ZIRLO™ irradiation growth are limited by the relative motion of the rod as it slips over the grid spring and dimple surfaces. Clearances between the fuel rod ends and nozzles are provided so that Zircaloy or ZIRLO™ irradiation growth will not result in end interferences. As another example, stresses due to holddown springs in opposition to the hydraulic lift force are limited by the deflection characteristic of the springs. Stresses in the fuel assembly caused by tripping of the rod cluster control assembly have little influence on fatigue because of the small number of events during the life of an assembly. Welded joints in the fuel assembly structure are considered in the structural analysis of the assembly. Appropriate material properties of welds are used to ensure the design bases are met. Assembly components and prototype fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met.

The fuel assembly design loads for shipping have been established at 4 g axial and 6 g lateral. Probes are permanently placed into the shipping cask to monitor and detect fuel assembly displacements that would result from loads in excess of the criteria. Past history and experience have indicated that loads which exceeded the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components such as the grid assembly, sleeves, inserts, and structure joints have been performed to assure that the shipping

design limits do not result in impairment of fuel assembly function.

Dimensional Stability

The Vantage 5H Mechanical Test Program description and results are given in Reference 16 and are considered to be applicable to Vantage+ as the two assemblies are structurally essentially identical.

The development of the RFA design included a comprehensive set of mechanical and hydraulic tests: pressure drop, assembly vibration, fuel rod vibration, bulge joint strength, grid crush. A description of these tests and results is provided in References 19, 20 and 21.

The coolant flow channels are established and maintained by the structure composed of grids and guide thimbles. The lateral spacing between fuel rods is provided and controlled by the support dimples of adjacent grid cells. Contact of the fuel rods on the dimples is assured by the clamping force provided by the grid springs. Lateral motion of the fuel rods is opposed by the spring force and the internal moments generated between the spring and the support dimples. Grid testing is discussed in Reference 10.

No interference with control rod insertion into thimble tubes will occur during a postulated loss-of-coolant accident (LOCA) transient due to fuel rod swelling, thermal expansion, or bowing. In the early phase of the transient following the coolant break, the high axial loads which potentially could be generated by the difference in thermal expansion between fuel clad and thimbles are relieved by slippage of the fuel rods through the grids. The relatively low drag force restraint on the fuel rods will only induce minor thermal bowing not sufficient to close the fuel rod-to-thimble tube gap. This rod-to-grid slip mechanism occurs simultaneously with control rod drop.

Vibration and Wear

The effect of the flow induced vibration on the V5H and Vantage+ fuel assembly and individual fuel rods is minimal. The cyclic stress range associated with deflections of such small magnitude is insignificant and has no effect on the structural integrity of the fuel rod.

The conclusion that the effect of flow induced vibrations on the fuel assembly and fuel rod is minimal is based on test results and analysis documented in Reference 11.

Full flow vibration tests have been performed on the RFA covering both assembly and fuel rod vibration. These tests have shown the Robust Fuel Design is not susceptible to flow induced assembly vibration and provides improved fuel rod vibration performance over the prior designs. In addition, the RFA-2 mid grid design provides further enhancements that improve fuel rod vibration performance over the RFA design.

The reaction on the grid support due to vibration motions is also correspondingly small and much less than the spring preload. Firm contact is therefore maintained. No significant wear of the cladding or grid supports is expected during the life of the fuel assembly, based on out-of-pile flow tests, performance of similarly designed fuel in operating reactors (2), and design analyses.

Clad fretting and fuel rod vibration have been experimentally investigated as shown in Reference 11.

No significant guide thimble tube wear due to flow-induced vibration of the control rods is predicted. Based on a conservative wear analyses, Westinghouse concluded that the integrity of the guide tube is maintained during normal operation, accident conditions, and nonoperational loading condition for at least 250 weeks (> 3 cycles) of fuel assembly operation. The Nuclear Regulatory Commission (NRC) has concluded (12) that the Westinghouse analyses probably accounts for all the major variables in the wear process. However, the NRC requested additional confirmatory information supporting the absence of significant thimble wear (no wear hole formation) for the 17 x 17 fuel assembly design. Examination of 1434 guide thimble tubes in six fuel assemblies examined at Salem Unit 1 shows no wear hole formation. Four of the assemblies had control rods in the parked

position (7 1/2 inches into the guide thimble) for Cycle 1, and two assemblies had control rods parked for Cycles 1 and 2. The parked position of the control rods has the greatest potential for causing guide thimble wear due to flow induced vibrations. The results of this surveillance program satisfy the NRC request to verify the wear analysis conclusion of no wear holes.

Evaluation of the Reactor Core for Limited Displacement RPV Inlet and Outlet Nozzle Breaks

The STD fuel assembly response resulting from the most limiting main coolant pipe break was analyzed using time history numerical techniques. Since the resulting vessel motion induces primarily lateral loads on the reactor core, a finite element model similar to the seismic model described in Reference 10 was used to assess the fuel assembly deflections and impact forces.

The reactor core finite element model which simulates the fuel assembly interaction during lateral excitation consists of fuel assemblies arranged in a planer array with inter-assembly gaps. For the Salem Station, 15 fuel assemblies which correspond to the maximum number of assemblies across the core diameter were used in the mode. The fuel assemblies and the reactor baffle support are represented by single beam elements as shown on Figure 4.2-25. The time history motion for the upper and lower core plates and the barrel at the upper core plate elevation are simultaneously applied to the simulated reactor core model as illustrated on Figure 4.2-25. The three time history motions were obtained from the time history analysis of the reactor vessel and internals finite element model.

The fuel assembly response, namely the displacements and grid impact forces, were obtained from the reactor core model using the core plate and barrel motions resulting from a reactor coolant pump outlet double ended break. The maximum fuel assembly deflection was determined to occur in a peripheral fuel assembly. The fuel assembly stresses resulting from this deflection indicated significant safety margins compared to the allowable values. The grid maximum impact force for both the seismic and lateral blowdown accident conditions occurred at the peripheral fuel assembly locations adjacent to the baffle wall. The grid impact forces were appreciably lower for fuel assembly locations inward from the peripheral fuel. For the lateral blowdown case, only a small (outer) portion of the core experienced significant grid impact forces.

The maximum grid impact force obtained from the limiting rupture break was found to be less than the minimum grid strength (using the 95 x 95 value as determined by tests at reactor operating safe shutdown temperatures). The maximum square-root-of-the-sum-of-the-squares combination of the pipe rupture and safe shutdown earthquake loads for the limiting grid location was found to be less than the minimum grid strength.

The major components that determine the structural integrity of the fuel assembly are the grids. Mechanical testing and analysis of the Vantage 5H Zircaloy grid and fuel assembly have demonstrated that the Vantage 5H structural integrity under seismic/LOCA loads will provide margins comparable to the STD 17 x 17 fuel assembly design and will meet all design bases.

Since the Vantage+ assembly is structurally similar to that of Vantage 5H, the seismic and LOCA analysis for the Vantage 5H assembly are applicable to Vantage+ assembly. The use of ZIRLO™ guide thimbles will not affect the seismic and LOCA loads.

A Salem plant specific seismic and LOCA analysis was performed for the RFA (Reference 22). The results of this analysis demonstrate that the RFA and V5H fuel assembly designs are capable of maintaining a coolable core geometry and control rod insertability under the combined seismic and LOCA loading for both a homogeneous and mixed core of fuel assembly designs.

4.2.1.3.3 Operational Experience

Westinghouse has had considerable experience with Zircaloy-clad fuel since its introduction in the Jose Cabrera plant in June 1968. This experience is extensively described in Reference 2.

4.2.1.3.4 Test Rod and Test Assembly Experience.

This experience is presented in Sections 8 and 23 of Reference 3.

4.2.1.4 Testing and Inspection Plan

4.2.1.4.1 Quality Assurance Program

The quality assurance program plan of the Westinghouse Nuclear Fuel Division for Salem is summarized in Reference 13.

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The program provides for control over all activities affecting product quality, commencing with design and development, and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

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

4.2.1.4.2 Quality Control

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

Fuel System Components and Parts

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

Pellets

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

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

Rod Inspection

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

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

2. Rod welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specifications.

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

4. All fuel rods are inspected by gamma scanning or other approved methods to ensure proper plenum dimensions.

5. All fuel rods are inspected by gamma scanning or other approved methods to ensure that no significant gaps exist between pellets.

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

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

Assemblies

Each fuel assembly is inspected for compliance with drawing and/or specification requirements. Other incore control component inspection and specification requirements are given in Section 4.2.3.4.

Other Inspections

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

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

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

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

Process Control

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

The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and preselected color coding. A Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

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

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by QC. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Unused powder and substandard pellets are returned to storage in the original color-coded containers.

Loading of pellets into the clad is performed in isolated production lines, and again only one enrichment is loaded on a line at a time.

A serialized traceability code is placed on each fuel tube to provide unique identification. The end plugs are inserted and then inert-welded to seal the tube. The fuel tube remains coded and traceability identified until just prior to installation in the fuel assembly.

At the time of installation into an assembly, the traceability codes are removed and a matrix is generated to identify each rod in its positions within a given assembly. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

Similar traceability is provided for burnable poison, source rods, and control rods, as required.

4.2.1.4.3 Onsite Inspection

Surveillance of fuel and reactor performance is routinely conducted on Westinghouse reactors. Power distribution is monitored using the ex-core fixed and in-core movable detectors, and the BEACON (Best Estimate Analyzer for Core Operations Nuclear) on-line core monitoring system. BEACON is also known as the Power Distribution Monitoring System (PDMS) in the Technical Specifications. Coolant activity and chemistry is followed which permits early detection of any fuel clad defects.

Visual fuel inspection is routinely conducted during refueling. Additional fuel inspections are dependent on the results of the operational monitoring and the visual inspections.

4.2.2 Reactor Vessel Internals

4.2.2.1 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

1. The reactor internals, in conjunction with the fuel assemblies, shall direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

2. In addition to neutron shielding provided by the reactor coolant, a separate thermal shield is provided to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

3. Provisions shall be made for installing in-core instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

4. The core internals are designed to withstand mechanical loads arising from operating basis earthquake (OBE), design basis earthquake (DBE), and pipe ruptures and meet the requirement of Item 5 below.

5. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

6. Following the design basis accident (DBA), the plant shall be capable of being shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the DBA are shown in Table 4.2-1. To ensure column loading of rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 4.2-1.

4.2.2.2 Description and Drawings

The reactor vessel internals are described as follows:

The components of the reactor internals consist of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure, and the in-core instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guides for the in-core instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. Flow passages in the lower core plate are sized to provide the desired inlet flow distribution to the core. After passing through the

core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

All the major material for the reactor internals is Type 304 stainless steel. Parts not fabricated from Type 304 stainless steel include bolts and dowel pins which are fabricated from Type 316 stainless steel and the radial support clevis inserts and bolts which are fabricated of Inconel. The only stainless steel materials used in the reactor core support structures which have yield strengths greater than 90,000 pounds are the 403 series used for holddown springs. The use of these materials is compatible with the reactor coolant and is acceptable based on the 1971 ASME Boiler and Pressure Vessel Code, Case Number 1337.

All reactor internals are removable from the vessel for the purpose of their inspection as well as the inspection of the vessel internal surface.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown on Figure 4.2-8. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel and its lower end is restrained from transverse motion by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the core.

The lower core support structure and core barrel serve to provide passageways and direct the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

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

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried by the lower core plate into the lower core plate support flange on the core barrel shell and through the lower support columns to the core support and thence through the core barrel shell to the core barrel flange supported by the vessel flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct

connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

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

Radial and axial expansion of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within the ASME Section III limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy absorbing devices limit the displacement of the core after contacting the vessel bottom head. The load is then transferred through the energy absorbing devices of the lower internals to the vessel.

The energy absorbers are mounted on a base plate which is contoured on its bottom surface to the reactor vessel bottom internal geometry. Their number and design are determined so as to limit the stresses imposed on all components except the energy absorber to less than yield (ASME Code Section III valves). Assuming a downward vertical displacement, potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

Upper Core Support Assembly

The upper core support assembly, shown on Figures 4.2-10 and 4.2-12, consists of the upper support assembly and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the upper support assembly and the upper core plate and are fastened at the top and bottom to these plates. The support columns transmit the mechanical loadings between the upper support and upper core plate.

The guide tube assemblies shield and guide the control rod drive shafts and control rods. They are fastened to the upper support and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support.

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

core support assembly. The spring is compressed when the reactor vessel head is installed on the pressure vessel.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the upper support assembly and then into the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support and upper core plate. The upper support plate is particularly stiff to minimize deflection.

In-Core Instrumentation Support Structures

All bottom-mounted in-core instrumentation support structures consist of a system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7-6 shows the Basic Flux-Mapping System). Specifically, the flux thimbles enter the reactor vessel through the bottom penetration nozzles. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet

during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The in-core instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the in-core instrumentation support structure. Reactor vessel surveillance specimen capsules are covered in Section 4.5.1.

As part of the conversion to an all bottom mounted in-core instrumentation, the support structures for the original top entry core exit thermocouple system have been removed. Specifically, the five thermocouple columns on the upper support plate have been removed and their corresponding reactor vessel head penetrations have been cut and capped. However, the support bases for the columns were left on the upper support plate so as not to create any flow openings across the upper support plate.

The core exit thermocouple system went from a top entry system to a bottom entry system for the following reasons as stated in letter NPE-85-1035:

1. Resistance readings were taken on the thermocouples and many were found to have shorts or were declared inoperable. These would provide erroneous indication at normal operating conditions.

2. Thermocouple columns were bent/damaged several times during refueling operations and not all thermocouples were recoverable. The installation of bottom-mounted thermocouples provided an easier method of reactor disassembly and lower probability of damage.

3. Installation of a bottom entry system would also allow the elimination of the five instrument ports and resolve the numerous problems encountered with the qualification of the reference junction boxes and thermocouples on the top-mounted system.

4.2.2.3 Design Loading Conditions

The design loading conditions that provide the basis for the design of the reactor internals are:

1. Fuel Assembly Weight

2. Fuel Assembly Spring Forces

3. Internals Weight

4. Control Rod Scram (equivalent static load)

5. Differential Pressure

6. Spring Preloads

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7. Coolant Flow Forces (static)

8. Temperature Gradients

9. Differences in Thermal Expansion

   a. Due to temperature differences

   b. Due to expansion of different materials

10. Interference Between Components

11. Vibration (mechanically or hydraulically induced)

12. One or More Loops Out of Service

13. All Operational Transients Listed in Table 4.1-10

14. Pump Overspeed

15. Seismic Loads (OBE and DBE)

16. Blowdown Forces (due to cold and hot leg break)

Combined seismic and blowdown forces are included in the stress analysis as a design loading condition by assuming the maximum amplitude of each force to act concurrently.

The main objectives of the design analysis are to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue

stresses are considered when the allowable amplitude of oscillation is established.

As part of the evaluation of design loading conditions, extensive testing and inspection are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

4.2.2.4 Design Loading Categories

The combination of design loadings fits into either the normal, upset, or faulted conditions as defined in the ASME Section III Code.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions summarized in Table 5.1-10.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the LOCA plus the DBE condition, the deflection criteria of critical internal structures are the limiting values given in Table 4.2-1. The corresponding no loss of function limits are included in Table 4.2-1 for comparison purposes with the allowed criteria.

The criteria for the core drop accident are based upon analyses which have been performed to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately 1/2 inch. An additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1 1/4 inches which is insufficient to permit the grips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This device limits the fall of the core and absorbs the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive step is provided to ensure support.

4.2.2.5 Design Criteria Basis

The basis for the design stress and deflection criteria is identified below.

Allowable Stress

For normal operating conditions, Section III of the ASME Nuclear Power Plant Components Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered. Under Code Case 1618, bolt material Type 316 stainless steel is now covered in ASME Section III and is so treated. It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the DBA used for the core support structures are based on the January 1971 draft of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

4.2.3 Reactivity Control System

4.2.3.1 Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

4.2.3.1.1 Design Stresses

The Reactivity Control System is designed to withstand stresses originating from various operating conditions as summarized in Table 5.2-10.

Allowable Stresses: For normal operating conditions, Section III of the ASME Boiler and Pressure Code is used as a general guide.

Dynamic Analysis: The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the Reactivity Control System.

4.2.3.1.2 Material Compatibility

Materials are selected for compatibility in a PWR environment, for adequate mechanical properties at room and operating temperature, for resistance to adverse property changes in a radioactive environment, and for compatibility with interfacing components.

4.2.3.1.3 Reactivity Control Components

The reactivity control components are subdivided into two categories:

1. Permanent devices used to control or monitor the core

2. Temporary devices used to control or monitor the core.

The permanent type components are the rod cluster control assemblies, control rod drive assemblies, neutron source assemblies, and thimble plug assemblies. Although the thimble plug assembly does not directly contribute to the reactivity control of the reactor, it is presented as a Reactivity Control System component in this document because it can be used to restrict bypass flow through those thimbles not occupied by absorber, source, or burnable absorber rods.

The temporary component is the burnable absorber assembly. The design bases for each of the mentioned components are in the following paragraphs.

Absorber Rods

The following are considered design conditions under Subsections NG and NB of the ASME Boiler and Pressure Vessel Code Section III.

1. The external pressure equal to the Reactor Coolant System operating pressure
2. The wear allowance equivalent to 1,000 reactor trips
3. Bending of the rod due to a misalignment in the guide tube
4. Forces imposed on the rods during rod drop
5. Loads caused by accelerations imposed by the control rod drive mechanism
6. Radiation exposure for maximum core life
7. Temperature effects at operating conditions

The absorber material temperature shall not exceed its melting temperature (1470°F for Ag-In-Cd absorber material)(14).

Burnable Absorber Rods

Two kinds of discrete burnable absorber rods may be used at Salem. The first is the borosilicate glass (PYREX) burnable absorber. The second is the Wet Annular Burnable Absorber (WABA) design. See References 23 and 24 of this section for a more detailed discussion of WABA. Reference 24 extends the allowable lifetime of the WABA rods from 18,000 to 40,000 EFPH, and allows utilization of the WABA rods for up to two eighteen month cycles without requiring additional inspections of the WABA rods. Discrete burnable absorbers are utilized to meet nuclear design requirements of UFSAR Section 4.3. Some comparative data is also shown in UFSAR Table 4.3-1.

The burnable absorber rod clad is designed using Subsections NG and NB of the ASME Boiler and Pressure Vessel Code, Section III, 1973 as a general guide for Conditions I and II. For abnormal loads during Conditions III and IV, Code stresses are not considered limiting. Failures of the burnable absorber rods during these conditions must

not interfere with reactor shutdown or emergency cooling of the fuel rods.

The burnable absorber material is nonstructural. The structural elements of the burnable absorber rods are designed to maintain the absorber geometry even if the absorber material is fractured. The PYREX rods are designed so that the borosilicate absorber material is below its softening temperature (1492°F* for reference 12.5 weight percent boron rods). The WABA rods utilize an aluminum oxide/boron carbide (Al_2O_3-B_4C) absorber material which is a sintered ceramic and has a very high melting temperature. In addition, the structural elements are designed to prevent excessive slumping.

Neutron Source Rods

The neutron source rods are designed to withstand the following:

1. The external pressure equal to the Reactor Coolant System operating pressure

2. An internal pressure equal to the pressure generated by released gases over the source rod life

Thimble Plug Assembly

If used in the core (optional), the thimble plug assemblies satisfy the following:

1. Accommodate the differential thermal expansion between the fuel assembly and the core internals

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* Borosilicate glass is accepted for use in burnable absorber rods if the softening temperature is 1510 + 18°F. The softening temperature is defined in ASTM C 338.

2. Maintain positive contact with the fuel assembly and the core internals

3. Limit the flow through each occupied thimble to acceptable design value

4.2.3.1.4 Control Rod Drive Mechanisms

The mechanisms are Class I components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the control rod drive mechanism (CRDM) when a seismic disturbance has been postulated to confirm the ability of the mechanism to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

The CRDM design used for the 17 x 17 fuel assembly control rod is identical to the 15 x 15 CRDM. The seismic analysis and response of 17 x 17 CRDM will be identical to those of the 15 x 15 mechanisms.

Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the CRDMs are as follows:

1. 5/8-inch step

2. 150-inch travel

3. 360-pound maximum load

4. Step in or out at 45 inches/minute (72 steps/minute)

5. Power interruption shall initiate release of drive rod assembly

6. Trip delay of less than 150 ms - Free fall of drive rod assembly shall begin less than 150 ms after power interruption no matter what holding or stepping action is being executed with any load and coolant temperatures of 100°F to 550°F

7. 40-year design life with normal refurbishment

8. 28,00 complete travel excursions which is 13 x 10^6 steps with normal refurbishment

4.2.3.2 Design Description

Reactivity control is provided by neutron absorbing rods and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as:

1. Fuel depletion and fission product buildup

2. Cold to hot, zero power reactivity change

3. Reactivity change produced by intermediate-term fission products such as xenon and samarium

4. Burnable absorber depletion

The rod cluster control assemblies provide reactivity control for:

1. Shutdown

2. Reactivity changes due to coolant temperature changes in the power range

3. Reactivity changes associated with the power coefficient of reactivity

4. Reactivity changes due to void formation

If soluble boron were the sole means of control, the moderator temperature coefficient could be positive. It is desirable to have a negative moderator temperature coefficient throughout the entire cycle in order to reduce possible deleterious effects caused by a positive coefficient during loss-of-coolant or loss-of-flow accidents. This is accomplished by installation of burnable absorber assemblies.

The neutron source assemblies and spontaneous fission neutron sources associated with the irradiated fuel assemblies provide a means of monitoring the core during periods of low neutron activity.

The most effective reactivity control components are the rod cluster control assemblies and their corresponding drive rod assemblies which are the only kinetic parts in the reactor. Figure 4.2-13 identified the rod cluster control and drive rod assembly, in addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly, guide tubes, and CRDM.

The guidance system for the control rod cluster is provided by the guide tube as shown on Figure 4.2-13. The guide tube provides two regimes of guidance. In the lower section a continuous guidance system provides support immediately above the core. This system protects the rod against excessive deformation and wear due to hydraulic loading. The region above the continuous section provides support and guidance at uniformly spaced intervals.

The envelope of support is determined by the pattern of the control rod cluster as shown on Figure 4.2-13. The guide tube assures alignment and support of the control rods, spider body,

and drive rod while maintaining trip times at or below required limits. In the following paragraphs, each reactivity control component is described in detail.

4.2.3.2.1 Reactivity Control Components

Rod Cluster Control Assembly

The rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature variations. Two criteria have been employed for selection of the control groups. First, the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that some of these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability is met. The control and shutdown groups provide adequate shutdown margin which is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped assuming that the highest worth assembly remains fully withdrawn and assuming no changes in xenon or boron concentration.

A rod cluster control assembly comprises a group of individual neutron absorber rods fastened at the top end to a common spider assembly, as illustrated on Figure 4.2-14.

The absorber material used in the control rods is silver-indium-cadmium single piece absorber rod which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded rods which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant. In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing which is then sealed at the bottom and the top by welded Type 308L stainless steel end plugs as shown on Figure 4.2-15. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions. The cladding surface has been ion-nitrided for hardening and corrosion resistance.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper end plug is threaded for assembly to the spider and is machined with a reduced diameter shank to provide flexibility to the joint for any misalignment condition.

The spider assembly is a one-piece machined casting in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive rod assembly are machined into the upper end of the hub. A coil spring inside the spider body absorbs the impact energy at the end of a trip insertion. A centerpost which holds the spring and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. The spider casting material is CF3M cast 316 stainless steel.

The absorber rods are fastened securely to the spider assembly as shown in Figure 4.2-15 to assure trouble free service. The threaded end of the upper end plug is inserted into the bottom of the spider boss hole. A nut is tightened on and welded to the spider boss to prevent loosening. A lock pin is inserted into the aligned holes of the spider base and upper end plug and welded to prevent the end plug and rod from backing off.

The overall length is such that when the assembly is withdrawn through its full travel the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble.

Burnable Absorber Assembly

Each burnable absorber assembly consists of burnable absorber rods attached to a holddown assembly. Burnable absorber assemblies are shown on Figure 4.2-16.

The PYREX absorber rods consist of borosilicate glass tubes contained within Type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall tubular inner liner of Type 304 stainless steel. The top end of the liner is open to permit the diffused helium to pass into the void volume and the liner overhangs the glass. The liner has an outward flange at the bottom end to maintain the position of the liner with the glass. A PYREX burnable absorber rod is shown in longitudinal and transverse cross sections on Figure 4.2-17.

The WABA consist of aluminum oxide/boron carbide pellets contained within Zircaloy tubular cladding which is plugged and seal welded at the ends to encapsulate the pellets. The pellets are also supported along the length of their insider diameter by a thin wall tubular liner of Type 304 stainless steel. The top and bottom of the inner tube, or liner, is open to allow for coolant flow. There is a void above the Al_2O_3-B_4C pellets between the inner and outer tube to contain diffused helium. A cross section drawing of the WABA rod is shown in Figure 4.2-17A.

The rods are statically suspended and positioned in selected guide thimbles within specified fuel assemblies. The absorber rods in each fuel assembly are grouped and attached together at the top end of the rods to a holddown assembly by a flat, perforated retaining plate which fits within the fuel assembly top nozzle and rests on the adaptor plate. The retaining plate (and the absorber rods) is held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. This arrangement assures that the absorber rods cannot be ejected from the core by flow forces. Each rod is permanently attached to the base plate by a nut which is lock welded into place.

The clad in the rod assemblies is Zircaloy or slightly cold worked Type 304 stainless steel. All other structural materials are Type 304 or 308 stainless steel except for the springs which are Inconel 718. The borosilicate glass tube and /or aluminum oxide/boron carbide pellets provide sufficient boron content to meet the criteria discussed in Section 4.3.1.

Neutron Source Assembly

A neutron source assembly can be used to provide a base neutron level to monitor core multiplication to changes in core reactivity. Since there is very little neutron activity when the core is subcritical, such as refueling and approach to criticality, neutron source assemblies are placed in the reactor if required to provide a count rate greater than 1 cps in Mode 3 prior to starting the approach to critical on the source range monitors. The source range monitors, which receive their signal from the source range detectors, are used primarily when the core is subcritical and during special subcritical modes of operation. In addition to having a greater than 1 cps count rate, the source channels should maintain a signal to noise ratio of at least two in Mode 3 prior to beginning the approach to critical.

The source assembly also supplements detection of changes in the core multiplication factor during core loading, refueling and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Therefore a change in the multiplication factor can be detected during addition of fuel assemblies while loading the core, a change in control rod positions, and changes in boron concentration.

Both primary and secondary neutron source rods are used. The primary source rod, containing a radioactive material, spontaneously emits neutrons during the initial core loading and reactor startup. After the primary source rod decays beyond the desired neutron flux level, neutrons are then supplied by the secondary source rod. The secondary source rod contains a stable material which must be activated by neutron bombardment during reactor operation. The activation results in the subsequent release of neutrons. This becomes a source of neutrons during periods of low neutron flux, such as during refueling and subsequent startups.

The initial reactor core employs four source assemblies, two primary source assemblies, and two secondary source assemblies. Each primary source assembly contains one primary source rod and between 0 and 23 burnable absorber rods. Each secondary source assembly contains a symmetrical grouping of 4 or 6 secondary source rods and between 0 and 20 burnable absorber rods. The 4 rodlet secondary source utilizes a single encapsulated design. The 6 rodlet secondary source utilizes a double encapsulated design which provides additional margin against source material leakage. Locations not filled with a source or burnable absorber rod may contain a thimble plug (optional). Source assemblies are shown on Figures 4.2-18 and 4.2-19. A comparison of the single and double encapsulated secondary source design is provided in Table 4.2-2.

Following the initial operating cycle, the actual source range count rate will depend on the core loading and the outage duration. Normally for subsequent reloads, the primary sources are removed and the secondary sources continue functioning.

The core loading determines the placement of the secondary sources and other inherent neutron sources (i.e., spontaneous fission from the irradiated fuel) relative to the location of the source range detectors. For secondary source assemblies made of antimony-beryllium (Sb-Be, gamma-neutron reaction), ^{124}Sb has a 60.2 day half-life. Thus, extended outages may impact the effectiveness of the secondary sources. For such extended outages, inherent neutron sources are sufficient to provide the required greater than 1 cps count rate in Mode 3 prior to beginning the approach to critical. Likewise, if a reload design has sufficient spontaneous fission neutrons to ensure the minimum required count rate response on the source range channels, then there is no need to install the secondary source assemblies.

The primary and secondary source rods both utilize the same cladding material as the absorber rods. The single encapsulated secondary source rods contain approximately 500 grams of Sb-Be pellets in one rod. The double encapsulated secondary source rods contain approximately 338 grams of Sb-Be pellets in one rod. The primary source rods contain capsules of Californium source material and alumina spacer rods to position the source material within the cladding. The rods in each assembly are permanently fastened at the top end to a holddown assembly, which is identical to that of the burnable absorber assemblies.

The other structural members are constructed of Type 304 stainless steel except for the springs. The springs exposed to the reactor coolant are wound from an age hardened nickel base alloy for corrosion resistance and high strength. The springs, when contained within the rods where corrosion resistance is not necessary, are oil tempered carbon steel.

Thimble Plug Assembly

Thimble plug assemblies may be used in order to limit bypass flow through the rod cluster control guide thimbles in fuel assemblies which do not contain either control rods, source rods, or burnable absorber rods.

The thimble plug assemblies as shown on Figure 4.2-20 consist of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly.

The 24 short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Similar short rods are also used on the source assemblies and burnable absorber assemblies to plug the ends of all vacant fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adaptor plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a small lock-bar welded to the nut.

All components in the thimble plug assembly, except for the springs, are constructed from Type 304 stainless steel. The springs are wound from an age hardened nickel base alloy for corrosion resistance and high strength.

4.2.3.2.2 Control Rod Drive Mechanism

All parts exposed to reactor coolant are made of metals which resist the corrosive action of the water. Three types of metals are used exclusively: stainless steels, Inconel and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used. Cobalt based alloys are used for the pins and latch tips. Inconel is used for the springs of both latch assemblies and Type 304 stainless steel is used for all pressure containing parts. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts.

A position indicator assembly slides over the CRDM rod travel housing. It detects the drive rod assembly position by means of 42 discrete coils that magnetically sense the entry and presence of the rod drive line through its center line over the normal length of the drive rod travel.

Control Rod Drive Mechanism

Control rod drive mechanisms are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods and derive their name from this feature. The CRDM is shown on Figure 4.2-21 and schematically on Figure 4.2-22.

The primary function of the CRDM is to insert or withdraw rod control clusters within the core to control average core temperature and to shut down the reactor.

The CRDM is a magnetically operated jack. A magnetic jack is an arrangement of three electro-magnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod control clusters in the reactor core in discrete steps.

The CRDM consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, the latch assembly, and the drive rod assembly.

1. The pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal welded, maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded plug with a canopy seal weld for pressure integrity.

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

2. The coil stack assembly includes the coil housings, an electrical conduit and connector, and three operating coils: 1) the stationary gripper coil, 2) the moveable gripper coil, and 3) the lift coil.

   The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

   Energizing of the operation coils causes movement of the pole pieces and latches in the latch assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, moveable pole pieces, and two sets of latches: 1) the moveable gripper latch, and 2) the stationary gripper latch.

   The latches engage grooves in the drive rod assembly. The moveable gripper latches are moved up or down in 5/8 inch steps by the lift pole to raise or lower the drive rod assembly while the moveable gripper latches are repositioned for the next 5/8 inch step.

4. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The drive rod has 5/8 inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and produces the means for coupling to the rod control cluster assembly.

   The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod control cluster assembly and permits remote disconnection of the drive rod.

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The CRDM is a trip design. Tripping can occur during any part of the power cycler sequencing if power to the coils is interrupted. The CRDM is threaded and seal welded on an adapter on top of the reactor vessel and is coupled to the rod control cluster assembly directly below.

The mechanism is capable of handling a 360 pound load, including the drive rod weight, at a rate of 45 inches per minute. Withdrawal of the rod control cluster is accomplished by magnetic forces while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F with forced air cooling required to maintain that temperature.

The CRDM shown schematically on Figure 4.2-22 withdraws and inserts its control rod as electrical pulses are received by the operator coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferro-magnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the drive mechanism holds the control rod withdrawn from the core in a static position until the movable gripper coil is energized.

Rod Cluster Control Assembly Withdrawal

The control rod is withdrawn by repetition of the following sequence of events:

1. Movable Gripper Coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 1/16 inch axial clearance exists between the latch teeth and the drive rod.

2. Stationary Gripper Coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

3. Lift Coil (C) - ON

   The 5/8 inch gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 inch).

4. Stationary Gripper Coil (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16 inch vertical drive rod assembly movement transfers the drive rod assembly load from the moveable gripper latches to the stationary gripper latches.

5. Movable Gripper Coil (B) - OFF

The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

6. Lift Coil (C) - OFF

The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

7. Repeat Step 1

The sequence described above (1 through 6) is termed as one step or one cycle. The control rod moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5/8 inch groove pitch) is raised 72 grooves per minute. The control rod is thus withdrawn at a rate up to 45 inches per minute.

Rod Cluster Control Assembly Insertion

The sequence for control rod insertion is similar to that for control rod withdrawal, except the timing of lift coil (C) ON and OFF is changed to permit lowering the control rod.

1. Lift Coil (C) - ON

The 5/8 inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Movable Gripper Coil (B) - ON

The latch locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 1/16 inch axial clearance exists between the latch teeth and the drive rod assembly.

3. Stationary Gripper Coil (A) - OFF

The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 1/16 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift Coil (C) - OFF

The force of gravity separates the movable gripper pole from the lift pole and the drive rod assembly and attached control rod drop down 5/8 inch.

5. Stationary Gripper (A) - ON

The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 1/16 inch. The 1/16 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable Gripper Coil (B) - OFF

The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Repeat Step 1

The sequences are repeated, as for control rod withdrawal, up to 72 times per minute which give a control rod insertion rate of 45 inches per minute.

Holding and Tripping of the Control Rods

During most of the plant operating time, the CRDMs hold the control rods withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached control rod hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the rod cluster control assembly is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight acting upon the latches. After the drive rod assembly is released by the mechanism, it falls freely until the control rods enter the buffer section of their thimble tubes.

4.2.3.3 Design Evaluation

4.2.3.3.1 Reactivity Control Components

The components are analyzed for loads corresponding to normal, upset, emergency, and faulted conditions. The analysis performed

depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control Rod Trip (equivalent static load)

2. Differential Pressure

3. Spring Preloads

4. Coolant Flow Forces (static)

5. Temperature Gradients

6. Differences in thermal expansion

    a. Due to temperature differences

    b. Due to expansion of different materials

7. Interference Between Components

8. Vibration (mechanically or hydraulically induced)

9. All Operational Transients Listed in Table 4.1-10

10. Pump Overspeed

11. Seismic Loads (OBE and DBE)

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the

functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data (11) and experience from operating reactors.

The design of reactivity component rods provides a sufficient cold void volume within the burnable absorber and source rods to limit the internal pressures to a value which satisfies the criteria in Section 4.2.3.1. The void volume for the helium in the burnable absorber rods is obtained through the use of glass in tubular form which provides a central void along the length of the rods. Helium gas is not released by the neutron absorber rod material; thus the absorber rod only sustains an external pressure during operating conditions. The internal pressure of source rods continues to increase from ambient until end-of-life at which time the internal pressure never exceeds that allowed by the criteria in Section 4.2.3.1. The stress analysis of reactivity component rods assumes 100 percent gas release to the rod void volume, considers the initial pressure within the rod, and assumes the pressure external to the component rod is zero.

Based on available data for properties of the borosilicate glass and on nuclear and thermal calculations for the burnable absorber rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube could occur but would continuously until the glass came in contact with the inner liner. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping and to collapse locally before rupture of the exterior cladding if unexpected large volume changes due to swelling or cracking should occur. The top of the inner liner is open to

allow communication to the central void by the helium which diffuses out of the glass.

Sufficient diametral and end clearances have been provided in the neutron absorber, burnable absorber, and source rods to accommodate the relative thermal expansions between the enclosed material and the surrounding clad and end plugs. There is no bending or warping induced in the rods although the clearance offered by the guide thimble would permit a postulated warpage to occur without restraint on the rods. Bending, therefore, is not considered in the analysis of the rods. The radial and axial temperature profiles have been determined by considering gap conductance, thermal expansion, and neutron and/or gamma heating of the contained material as well as gamma heating of the clad. The maximum neutron absorber material temperature was found to be less than 850°F which occurs axially at only the highest flux region. The maximum borosilicate glass temperature was calculated to be about 1200°F and takes place following the initial rise to power. The glass temperature then decreases rapidly for the following reasons: (1) reduction in power generation due to B_{10} depletion; (2) better gap conductance as the helium produced diffuses to the gap; and (3) external gap reduction due to borosilicate glass creep. Rod, guide thimble, and dashpot flow analysis performed indicates that the flow is sufficient to prevent coolant boiling and maintain clad temperatures at which the clad material has adequate strength to resist coolant operating pressures and rod internal pressures.

Analysis on the rod cluster control spider indicates the spider is structurally adequate to withstand the various operating loads including the higher loads which occur during the drive mechanism stepping action and rod drop.

The reactivity control component materials selected are considered to be the best available from the standpoint of resistance to irradiation damage and compatibility to the reactor environment. The materials selected partially dictate the reactor environment

(e.g., Cl^- control in the coolant). The current design type reactivity controls have been in service for more than 10 years with no apparent degradation of construction materials.

With regard to the materials of construction exhibiting satisfactory resistance to adverse property changes in a radioactive environment, it should be noted that on work on breeder reactors in current design, similar materials are being applied. At high fluences the austenitic materials increase in strength with a corresponding decreased ductility (as measured by tensile tests) but energy absorption (as measured by impact tests) remains quite high. Corrosion of the materials exposed to the coolant is quite low and proper control of Cl^- and 0_2 in the coolant will prevent the occurrence of stress corrosion. All of the austenitic stainless steel base materials used are processed and fabricated to preclude sensitization.

Analysis of the rod cluster control assemblies shows that if the drive mechanism housing ruptures, the rod cluster control assembly will be ejected from the core by the pressure differential of the operating pressure and ambient pressure across the drive rod assembly. The ejection is also predicted on the failure of the drive mechanism to retain the drive rod/rod cluster control assembly position. It should be pointed out that a drive mechanism housing rupture will cause the ejection of only one rod cluster control assembly with the other assemblies remaining in the core.

Ejection of a burnable absorber or thimble plug assembly (if used) is conceivable based on the postulation that the holddown bar fails and that the base plate and burnable absorber rods are severely deformed. In the unlikely event that failure of the holddown bar occurs, the upward displacement of the burnable absorber assembly only permits the base plate to contact the upper core plate. Since this displacement is small, the major portion of the burnable absorber material remains positioned within the core. In the case of the thimble plug assembly, the thimble plugs will partially remain in the fuel assembly guide thimbles thus maintaining a majority of the desired flow impedance. Further displacement or complete ejection would necessitate the square base plate and burnable absorber rods be forced, thus plastically deformed, to fit up through a smaller diameter hole. It is expected that this condition requires a substantially higher force or pressure drop than that of the holddown bar failure.

Experience with control rods, burnable absorber rods, and source rods is discussed in Reference 2.

The mechanical design of the reactivity control components provides for the protection of the active elements to prevent the loss of control capability and functional failure of critical components. The components have been reviewed for potential failure and consequences of a functional failure of critical parts. The results of the review are summarized below.

Rod Cluster Control Assembly

1. The basic absorbing material is sealed from contact with the primary coolant and the fuel assembly and guidance surfaces by a high quality stainless steel clad.

Potential loss of absorber mass or reduction in reactivity control material due to mechanical or chemical erosion or wear is therefore reliably prevented.

2. A breach of the cladding for any postulated reason does not result in serious consequences. The absorber material, silver-indium-cadmium, is relatively inert and would still remain remote from high coolant velocity regions. Rapid loss of material resulting in significant loss of reactivity control material would not occur.

3. The individually clad absorber rods are doubly secured to the retaining spider finger by a threaded top end plug secured by a nut welded to the finger and a welded lock pin.

It should also be noted that in several instances of control rod jamming caused by foreign particles, the individual rods at the site of the jam have borne the full capacity of the CRDM and higher impact loads to dislodge the jam without failure. The guide tube card/guide thimble arrangement is such that large loads are required to buckle individual control rods. The conclusion to be drawn from this experience is that this joint is extremely insensitive to potential mechanical damage. A failure of the joint would result in the insertion of the individual rod into the core. This results in reduced reactivity which is a fail safe condition. Further information is given in Reference 2.

4. The spider is a one-piece machined casting and includes the radial vanes and fingers. Reliability is increased by not using brazed joints. Casting allows the rod holes to be drilled to the positional tolerances prior to assembly to ensure the rods will align with the guide cards.

5. The spider hub being of a one-piece machined casting is very rugged and of extremely low potential for damage. It is difficult to postulate any condition to cause failure. Should some unforeseen event cause fracture of the hub above the vanes, the lower portion with the vanes and rods attached would insert by gravity into the core causing reactivity decrease. The rod could then not be removed by the drive line, again a fail safe condition. Fracture below the vanes cannot be postulated since all loads, including scram impact, are taken above the vane elevation.

6. The rod cluster control rods are provided a clear channel for insertion by the guide thimbles of the fuel assemblies. All fuel rod failures are protected against by providing this physical barrier between the fuel rod and the intended insertion channel. Distortion of the fuel rods by bending cannot apply sufficient force to damage or significantly distort the guide thimble. Fuel rod distortion by swelling, though precluded by design, would be terminated by fracture before contact with the guide thimble occurs. If such were not the case, it would be expected that a force reaction at the point of contact would cause a slight deflection of the guide thimble. The radius of curvature of the deflected shape of the guide thimbles would be sufficiently large to have a negligible influence on rod cluster control insertion.

Burnable Absorber Assemblies

The burnable absorber assemblies are static temporary reactivity control elements. The axial position is assured by the holddown assembly which bears against the upper core plate. Their lateral position is maintained by the guide thimbles of the fuel assemblies.

The individual rods are shouldered against the underside of the retainer plate and securely fastened at the top by a threaded nut which is then locked in place. The square dimension of the retainer plate is larger than the diameter of the flow holes through the core plate. Failure of the holddown bar or spring pack therefore does not result in ejection of the burnable absorber rods from the core.

The only incident that could potentially result in ejection of the burnable absorber rods is a multiple fracture of the retainer plate. This is not considered credible because of the light loads borne by this component. During normal operation the loads borne by the plate are approximately 5 lbs per rod, or a total of 100 lbs. distributed at the points of attachment. Even a multiple fracture of the retainer plate would result in jamming of the plate segments against the upper core plate, again preventing ejection. Excessive reactivity increase due to burnable poison ejection is therefore prevented.

The same type of stainless steel clad used on rod cluster control rods is also used on the burnable absorber rods. In this application there is even less susceptibility to mechanical damage since these are static assemblies. The guide thimbles of the fuel assembly afford the same protection from damage due to fuel rod failures as that described for the rod cluster control rods.

The consequences of clad breach are also similarly small. The absorber material is borosilicate glass which is maintained in position by a central hollow tube. In the event of a hole developing in the clad for any postulated reason the expected consequence is only the loss of the helium produced by the absorption process into the primary coolant. The glass is chemically inert and remains remote from high coolant velocities; therefore significant loss of absorber material resulting in reactivity increase is not expected.

Drive Rod Assemblies

All postulated failures of the drive rod assemblies either by fracture or uncoupling, lead to the fail safe condition. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with and is guided by the rod cluster control assembly. This always results in reactivity decrease for the control rods.

4.2.3.3.2 Control Rod Drive Mechanism

Material Selection

All pressure-containing materials comply with Section III of the ASME Pressure Vessel Code, and, with the exception of the needle vent valve, will be fabricated from austenitic (Type 304) stainless steel or CF-8 stainless steel. The vent valve is a modified austenitic stainless steel cap screw.

Magnetic pole pieces are fabricated from Type 410 stainless steel. All nonmagnetic parts, except pins and springs, are fabricated from Type 304 stainless steel. Haynes 25 is used to fabricate link pins. Springs are made from Inconel-X. Latch arm tips are clad with Stellite to provide improved wearability. Hard chrome plate and Stellite are used selectively for bearing and wear surfaces.

At the start of the development program, a survey was made to determine whether a material better than Type 410 stainless steel was available for the magnetic pole pieces. Ideal material requirements are as follows:

1. High magnetic saturation value

2. High permeability

3. Low coercive force

4. High resistivity

5. High Curie temperature

6. Corrosion resistant

7. High impact strength

8. Nonoriented

9. High machinability

10. Radiation damage

After a comprehensive material trade-off study was made it was decided that the Type 410 stainless steel was satisfactory for this application.

The cast coil housings require a magnetic material. Both low-carbon cast steel and ductile iron have been successfully tested for this application. The choice, made on the basis of cost, indicated that ductile iron will be specified on the CRDM. The finished housings are zinc plated to provide corrosion resistance.

Coils are wound on bobbins of molded Dow Corning 302 material, with double glass-insulated copper wire. Coils are then vacuum impregnated with silicon varnish. A wrapping of mica sheet is secured to the coil outer surface. The result is a well-insulated coil capable of sustained operation at 200°C.

The drive shaft assembly utilized a Type 410 stainless steel drive rod. The coupling is machined from Type 403 stainless steel. Other

parts are Type 304 stainless steel with the exception of the springs which are Inconel-X and the locking button which is Haynes 25.

Radiation Damage

As required by the equipment specification, the CRDMs are designed to meet a radiation requirement of 10 Rads/Hr. Materials have been selected to meet this requirement. The above radiation level which amount to 1.753×10^6 Rads in 20 years will not degrade control rod drive mechanism life. Control rod drive mechanisms at Yankee Rowe which have been in operation since 1960 have not experienced problems due to radiation.

Positioning Requirements

The mechanism has a step length of 5/8 inches which determines the positioning capabilities of the control rod drive mechanism. (Note: Positioning requirements are determined by reactor physics.)

Elevation of Materials Adequacy

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Boiler and Pressure Vessel Code, Section III. Internals components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment as confirmed by life tests.

Results of Dimensional and Tolerance Analysis

With respect to the CRDM systems as a whole, critical clearances are present in the following areas:

1. Latch assembly (Diametral clearances)

2. Latch arm-drive rod clearances

3. Coil stack assembly-thermal clearances

4. Coil fit in coil housing

The following write-up defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 650°F minimum clearance is 0.0045 inch, and at the maximum expected operating temperatures of 550°F is 0.0057 inch.

Latch Arm - Drive Rod Clearances

The CRDM incorporates a load transfer action. The movable or stationary gripper latch is not under load during engagement, as previously explained, due to load transfer action.

Figure 4.2-23 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 4.2-24 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearance of the coil stack assembly over the latch housing was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F the inside diameter of the coil stack is 7.308/7.298 inches. The outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism due to operating temperature of the CRDM results in minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above, it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed from four hot CRDMs mounted on 11.035-inch centers on a 550°F test loop, allowed to cool, and then replaced without incident as a test to prove the proceeding.

Coil Fit in Coil Housing

Control rod drive mechanism and coil housing clearances are selected so that coil heatup results in a close or tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

4.2.3.4 Tests, Verification, and Inspections

4.2.3.4.1 Reactivity Control Components

Tests and inspections are performed on each reactivity control component to verify the mechanical characteristics. In the case of the rod cluster control assembly, prototype testing has been conducted, and both manufacturing test/inspections and functional testing at the plant site are performed.

During the component manufacturing phase, the following requirements apply to the reactivity control components to assure the proper functioning during reactor operation:

1. All materials are procured to specifications to attain the desired standard of quality.

2. All clad/end plug welds are checked for integrity by visual inspection and X-ray, and are helium leak checked. All the seal welds in the neutron absorber rods, burnable poison rods, and source rods are checked in this manner.

3. To assure proper fitup with the fuel assembly, the rod cluster control, burnable poison, and source assemblies are installed in the fuel assembly and checked for binding in the dry condition.

   The rod cluster control assemblies (RCCA) are functionally tested following core loading, but prior to criticality to demonstrate reliable operation of the assemblies. Each assembly is operated (and tripped) one time at no flow/cold conditions and one time at full flow/hot conditions. In addition, selected assemblies, amounting to about 15 to 20 percent of the total assemblies are operated at no-flow/operating temperature conditions and full flow/ambient conditions. Also the slowest rod and the fastest rod are tripped 10 times at no-flow/ambient conditions and at full flow/operating temperature conditions. Thus each assembly is tested a

minimum of 2 times or up to 14 times maximum to ensure that the assemblies are properly functioning.

In order to demonstrate continuous free movement of the RCCAs and to ensure acceptable core power distributions during operations, partial movement checks are performed on every RCCA, as required by the Technical Specifications. In addition, periodic drop tests of the RCCAs are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements.

If an RCCA cannot be moved by its mechanism, adjustments in the boron concentration of the coolant ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated but would impose additional demands on the plant operator. Therefore, the number of inoperable RCCAs has been limited to one.

4.2.3.4.2 Control Rod Drive Mechanism

Quality assurance procedures during production of CRDMs include material selection, process control, mechanism component tests during production, and hydrotests.

After all manufacturing procedures had been developed, several prototype CRDMs and drive rod assemblies were life tested with the entire drive line under environmental conditions of temperature, pressure, and flow. All acceptance tests were of duration equal to or greater than service required for the plant operation. All drive rod assemblies tested in this manner have shown minimal wear damage.

These tests include verification that the trip time achieved by the CRDMs met the design requirement of 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry. Trip time requirement will be confirmed for each CRDM at periodic intervals after initial reactor operation. In addition,

a Technical Specification has been set to ensure that the trip time requirement is met.

It is expected that all CRDMs will meet specified operating requirements for the duration of plant life with normal refurbishment. However, a Technical Specification pertaining to an inoperable RCCA has been set.

If an RCCA cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable RCCAs has been limited to one.

In order to demonstrate continuous free movement of the RCCAs and to ensure acceptable core power distributions during operation, partial-movement checks are performed on every RCCA at least every 31 days during reactor critical operation. In addition, periodic drop tests of the full length RCCAs are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical RCCA ejection. During these tests the acceptable drop time of each assembly is not greater than 2.7 seconds, at full flow and operating temperature, from the beginning of decay of stationary gripper coil voltage to dashpot entry.

To confirm the mechanical adequacy of the fuel assembly and RCCA, functional test programs have been conducted on a full scale control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive

line components did not reveal significant fretting. The control rod free fall time against 125 percent of nominal flow was less than 1.5 seconds to the dashpot; about 10 feet of travel.

Actual experience on the Ginna, Mihama No. 1, Point Beach No. 1, and H. B. Robinson plants indicates excellent performance of CRDMs.

All units are production tested prior to shipment to confirm ability of CRDMs to meet design specification-operational requirements. Periodic tests are also conducted during plant operation to confirm brake core operation.

During refueling, tests are also conducted to confirm condition to stator windings.

4.2.4 References for Section 4.2

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