3.3 REACTOR VESSEL INTERNALS MECHANICAL DESIGN

3.3.1 Power Generation Objective

Reactor vessel internals (exclusive of fuel, control rods and incore flux monitors) are provided to achieve the following objectives:

a. Maintain partitions between regions within the reactor vessel to provide proper coolant distribution, thereby allowing power operation without fuel damage due to inadequate cooling.

b. Provide positioning and support for the fuel assemblies, control rods, incore flux monitors, and other vessel internals to assure that control rod movement is not impaired.

c. Provide a source of neutrons to assure meaningful nuclear measurements at reactor low power levels.

3.3.2 Power Generation Design Basis

1. The reactor vessel internals shall be designed to provide proper coolant distribution during all anticipated normal operating conditions to allow power operation of the core without fuel damage.

2. The reactor vessel internals shall be arranged to facilitate refueling operations.

3. The reactor vessel internals shall include devices that permit assessment of the core reactivity condition during periods of low power and subcritical operations.

4. Adequate working space and access shall be provided to permit adequate inspection of reactor vessel internals.

3.3.3 Safety Design Basis

1. The reactor vessel internals shall be arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

2. Deflections and deformation of reactor vessel internals shall be limited to assure that the control rods and the Core Standby Cooling Systems can perform their safety functions during abnormal operational transients and accidents.

3. The reactor vessel internals mechanical design shall assure that safety design bases 1 and 2 are satisfied in accordance with the loading criteria of Appendix C, so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.3.4 Description

The reactor vessel internals are installed inside the reactor vessel to properly distribute the flow of coolant delivered to the vessel, to locate and support the fuel assemblies, and to provide an inner volume containing the core that can be flooded following a break in the nuclear system process barrier external to the reactor vessel. The reactor vessel internals are the following components:

Core shroud
Shroud head and steam separator assembly
Core support (core plate)
Top guide
Fuel support pieces
Control rod guide tubes
Jet pump assemblies
Steam dryers
Feedwater spargers
Core spray lines and spargers
Vessel head cooling spray nozzle
Differential pressure and liquid control line
Incore flux monitor guide tubes
Startup neutron sources
Surveillance sample holders

The overall arrangement of the internals within the reactor vessel is shown in Figure 3.3-1. Table 3.3-1 gives detailed design data for the various reactor vessel internals.

Although not mandatory, the design of the reactor vessel internals is in accordance with the intent of Section III of the ASME Boiler and Pressure Vessel Code. The material used for most of the fabrication of the reactor vessel internals is solution heat-treated, unstabilized type 304 austenitic stainless steel conforming to ASTM specifications. Weld procedures and welders are qualified in accordance with the intent of Section IX of the ASME Boiler and Pressure Vessel Code. The floodable inner volume of the reactor vessel is shown on Figure 4.3-4. It is the volume inside the core shroud up to the level of the jet pump nozzles. The boundary of the floodable inner volume consists of the following (see Figure 3.3-1):

a. The jet pumps from the jet pump nozzles down to the shroud support.

b. The shroud support, which forms a barrier between the outside of the shroud and the inside of the reactor vessel.

c. The reactor vessel wall below the shroud support.

d. The core shroud up to the level of the jet pump nozzles.

3.3.4.1 Core Structure

The core structure surrounds the active core of the reactor and consists of the core shroud, shroud head and steam separator assembly, core support, and top guide. This structure is used to form partitions within the reactor vessel to sustain pressure differentials across the partitions, to direct the flow of the coolant water, and to laterally locate and support the fuel assemblies, control rod guide tubes, and steam separators. Figure 3.3-2 shows the reactor vessel internal flow paths. The core structure is designed in accordance with the structural loading criteria of Appendix C.

3.3.4.1.1 Core Shroud

The core shroud is a stainless steel cylindrical assembly which provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus thus providing a floodable region following a recirculation line break. The volume enclosed by the core shroud is characterized by three regions each with a different shroud diameter. The upper shroud has the largest diameter and surrounds the core discharge plenum which is bounded by the shroud head on top and the top guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has the intermediate diameter and is bounded at the bottom by the core support assembly. The lower shroud, surrounding part of the lower plenum, has the smallest diameter and at the bottom is welded to the reactor vessel shroud support (see Subsection 4.2 "Reactor Vessel and Appurtenances Mechanical Design").

3.3.4.1.2 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the upper shroud to form the top of the core discharge plenum. This plenum provides a mixing chamber for the steam-water mixture before it enters the steam separators. Long holddown bolts are used for easy access during removal. The individual stainless steel axial flow steam separators shown in Figure 3.3-3 are attached to the top of standpipes which are welded into the shroud head.

The centrifugal type steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes which impart a spin to establish a vortex separating the water from the steam. The steam exits from the top of the separator and rises up to the dryers. The separated water exits from under the separator cap and flows out between the standpipes, draining into the recirculation flow downcomer annulus.

3.3.4.1.3 Core Support (Core Plate)

The core support consists of a circular stainless steel plate stiffened with a rim and beam structure. Perforations in the plate provide lateral support and guidance for the control rod guide tubes, peripheral fuel support pieces, incore flux monitor guide tubes, and startup neutron sources. The entire assembly is bolted to a support ledge between the central and lower portions of the core shroud after proper positioning has been assured by alignment pins which fit into slots in the ledge.

3.3.4.1.4 Top Guide

The top guide is formed by a series of stainless steel beams joined at right angles to form square openings. Each opening provides lateral support and guidance for four fuel assemblies. Detent sockets are provided beneath the top guide to anchor dry tubes, power range monitor incore detectors, and neutron sources. The top guide is positioned by alignment pins which fit into radial slots in the top of the shroud.

3.3.4.2 Fuel Support Pieces

The fuel support pieces, shown in Figure 3.3-4, are of two basic types--peripheral and four-lobed. The peripheral fuel support pieces, which are welded to the core support, are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support piece will support one fuel assembly and contains a replaceable orifice assembly designed to assure proper coolant flow to the fuel assembly. The four-lobed fuel support pieces will each support four fuel assemblies and are provided with orifice plates to assure proper coolant flow distribution to each fuel assembly. The four-lobed fuel support pieces rest in the top of the control rod guide tubes and are supported laterally by the core support. The control rod blades pass through slots in the center of the four-lobed fuel support pieces. A control rod and the four fuel assemblies which immediately surround it represent a core cell (see Subsection 3.2, "Fuel Mechanical Design").

3.3.4.3 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, (see Figure 3.3-1) extend from the top of the control rod drive housings up through holes in the core support. Each tube is designed as the lateral guide for a control rod and as the vertical support for a four-lobed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing (see Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design") which in turn transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod drive housing from below and is rotated to lock the control rod drive tube in place. A key is inserted into a locking slot in the bottom of the control rod drive housing to hold the thermal sleeve in position.

3.3.4.4 Jet Pump Assemblies

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor vessel wall. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser (Figure 3.3-5). The driving nozzle, suction inlet, and throat are joined together as a removable unit and the diffuser is permanently installed. High pressure water from the recirculation pumps (see Subsection 4.3, "Reactor Recirculation System") is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace is welded to cantilever beams extending from pads on the reactor vessel wall.

The jet pump diffuser is a gradual conical section changing to a straight cylindrical section at the lower end. The diffuser is supported vertically by the shroud support, a flat ring which is welded to the reactor vessel wall and to which is welded the shroud support cylinder. The joint between the throat and the diffuser is a slip fit. A metal-to-metal spherical to conical seal joint is used between the nozzle entry section and riser with firm contact maintained by a clamp arrangement which fits under posts on the riser and utilizes a bolt to provide a downward force on a pad on top of the nozzle entry section. The throat section is supported laterally by a bracket attached to the riser. The jet pump diffuser section is welded to the shroud support and provides a positive seal. This permits reflooding the core to the top of the jet pump inlet following a design basis loss-of-coolant accident.[1]

1 "Design and Performance of GE BWR Jet Pumps," General Electric Co., Atomic Power Equipment Department, July 1968. (APED-5460).

3.3.4.5 Steam Dryers

The steam dryer removes moisture from the wet steam which exits from the steam separator. The wet steam leaving the steam separator flows across the dryer vanes and the moisture flows down through collecting troughs and tubes to the water above the downcomer annulus (see Figure 3.3-6). A skirt extends down into the water to form a seal between the wet steam plenum and the dry steam flowing out the top of the dryer to the steam outlet nozzles. Vertical guide rods facilitate positioning the dryer and shroud head in the vessel. Replacement steam dryers (RSD) have been designed to support Extended Power Uprate (EPU) operation. The RSDs hare curved hood six-bank dryers constructed mainly of type 304L stainless steel. The RSD design incorporates design features that were developed to accommodate flow induced vibration (FIV) acoustic loads that lead to steam dryer failures at a BWR-3 plant. The dryer rests on steam dryer support brackets attached to the reactor vessel wall. The original steam dryers are restricted from lifting by steam dryer holddown brackets which are attached to the reactor vessel closure head over the top of the steam dryer lifting lugs when the head is in place. RSDs are restricted from vertical lifting by latch assemblies which hook underneath two of the steam dryer support brackets.

3.3.4.6 Feedwater Spargers

As a result of cracks discovered in the feedwater nozzle blend radius, nozzle bore regions, and around the sparger flow holes, the General Electric Company (GE) developed an improved nozzle/sparger design which would reduce this cracking. This new design has been installed in all three units. A separate sparger is fitted to each feedwater nozzle (6) with a double piston ring thermal sleeve. Each sparger is shaped to conform to the curve of the vessel wall and is attached to the wall with two end brackets. These end brackets are bolted to the vessel wall brackets which support the weight of the spargers and position the sparger away from the vessel wall. Flow nozzles are welded to the inner radii of the sparger. Feedwater flow enters the center of the sparger and is discharged radially inward and downward through the nozzles to mix the cooler feedwater with the downcomer flow from the steam separators before it contacts the vessel wall. The feedwater also serves to collapse the steam voids and to subcool the water flowing to the jet pumps and recirculation pumps.

This improved nozzle/sparger design, in conjunction with an ultrasonic testing inspection program, will preclude the possibility of any crack growing to a depth which would endanger the pressure vessel integrity.

3.3.4.7 Core Spray Lines

The two 100-percent capacity core spray lines separately enter the reactor vessel through the two core spray nozzles as shown in Figures 4.2-1 and 4.2-3 (see Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"). The lines divide immediately inside the reactor vessel. The two halves are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The header halves are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger ring which is routed halfway around the inside of the upper shroud. The ends of the two sparger rings for each line are supported by slip-fit brackets designed to accommodate thermal expansion of the rings. The header routings and supports are designed to accommodate differential movement between the shroud and the vessel. The lower portion of Core Spray Downcomer “C” has been replaced with a sectional replacement (Unit 3 only). The other core spray line is identical except that the header enters the opposite side of the vessel and the sparger rings are at a slightly different elevation in the shroud. The proper spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the sparger rings (see Section 6, "Core Standby Cooling Systems").

3.3.4.8 Vessel Head Cooling Spray Nozzle

The vessel head cooling spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange above a reactor vessel head nozzle. The vessel head cooling spray nozzles are still installed but are not functional since blind flanges are permanently installed at the mating flange.

3.3.4.9 Differential Pressure and Liquid Control Line

The differential pressure and liquid control line serves a dual function within the reactor vessel to inject liquid control solution into the coolant stream (discussed in Subsection 3.8, "Standby Liquid Control System") and to sense the differential pressure across the core support assembly (see Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"). The line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support assembly. It is used to sense the pressure below the core support during normal operation and to inject liquid control solution when required. This location assures that good mixing and dispersion are facilitated. The use of the inner pipe also reduces the thermal shock to the vessel nozzle should the Standby Liquid Control System ever be used. The outer pipe terminates immediately above the core support assembly and senses the pressure in the region outside the fuel assembly channels.

3.3.4.10 Incore Flux Monitor Guide Tubes

The incore flux monitor guide tubes are welded to the top of the incore flux monitor housings (see Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design") in the lower plenum and extend up to the top of the core support. The power range detectors for the power range monitoring units and the dry tubes for the source range monitoring/intermediate range monitoring (SRM/IRM) detectors are inserted through the guide tubes and are held in place below the top guide by spring tension. A lattice work of clamps, tie bars, and spacers is bolted around the guide tubes at the approximate level of the reactor vessel shroud support to give lateral support and rigidity to the guide tubes. The bolts and clamps are welded after assembly to prevent loosening during reactor operation.

3.3.4.11 Startup Neutron Sources

Startup neutron sources are used to provide a sufficient neutron population to assure that the core neutron flux is detectable by installed neutron monitors and to assure that significant changes in core reactivity are readily detectable by installed neutron flux instrumentation. Antimony-beryllium neutron sources were used for cycle 1. Antimony-beryllium or californium neutron sources may be used in later cycles if spent fuel alone cannot provide the required neutron population. (See Subsection 7.5, "Neutron Monitoring System").

3.3.4.12 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimens capsules (see Subsection 4.2, "Reactor Vessel and Appurtenances Mechanical Design"). The baskets hang from brackets on the inside diameter of the reactor vessel at the mid height of the active core and at radial positions of 30°, 120°, and 300°. These locations are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself while at the same time avoiding jet pump removal interference or damage.

3.3.5 Safety Evaluation

3.3.5.1 Evaluation Methods

To determine that the safety design basis is satisfied, the responses of the reactor vessel internals to loads imposed during normal operation, abnormal operational transients, and accidents were examined. Determination of these effects on the ability to insert control rods, cool the core, and flood the inner volume of the reactor vessel was made. The various structural loading combinations assumed to be

imposed on the reactor vessel internals are as described in Appendix C for Class I equipment. These loading combinations include upset loads, emergency loads, and faulted loads.

The ASME Boiler and Pressure Code, Section III for Class A vessels, is used as a guide to determine limiting stress intensities and cyclic loadings for the reactor vessel internals. For those components, for which buckling is not a possible failure mode and stresses are within those stated in the ASME Code, it was concluded that the safety design basis is satisfied. For those components, for which either buckling is a possible failure mode or stresses exceed those presented in the ASME Code, then either the elastic stability of the structure or the resulting deformation was examined to determine if the safety design basis was satisfied.

3.3.5.1.1 Specific Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied reveals four significant events as follows:

a. Loss-of-coolant accident. This accident is a break in a recirculation line. The accident results in flow induced loads and acoustic shock loads on some of the reactor vessel internals.

b. Steamline break accident. This accident is a break in one main steamline between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across the reactor vessel internals.

c. Thermal shock. The most severe thermal shocks to the reactor vessel internals occur when low pressure coolant injection or high pressure coolant injection (LPCI or HPCI) operations reflood the reactor vessel inner volume following either a recirculation line break or a main steamline break (see Section 6, "Core Standby Cooling System").

d. Earthquake. This event subjects the reactor vessel internals to significant forces as a result of ground motion.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents showed that the loads affecting the reactor vessel internals are less severe than the four postulated events.

3.3.5.1.2 Reactor Internal Pressure Differentials (RIPD)

The core flow, feedwater temperature, reactor pressure, and power level that result in the maximum loads are used as initial conditions.

For GE analyses, the normal condition is analyzed using a digital computer code[2] for the steady-state thermal-hydraulic analysis of a BWR core (ISCOR). The reactor internal pressure differences (RIPDs) are calculated based on the ISCOR results. Appropriate adders or multipliers, which have been conservatively established based on the GE transient analysis methods on a generic basis for GE BWRs, are applied to the normal condition values to determine the upset condition values. For AREVA reload analyses, core and fuel channel RIPDs are evaluated based on XCOBRA analyses.

The locations at which RIPDs are applicable and calculated are shown in Figure 3.3-7. The bounding RIPD values which result from the range of possible reactor power and flow conditions are then determined.

The RIPD values acting on the major components at normal increased core flow conditions are documented in the following reference:

GEH 002N3430, Revision 0, Task T0304: Reactor Internal Pressure Differences and Fuel Lift Evaluation, March 2015, Table 3.3.1.1.

NOTE: The fuel type utilized in the reference is GE13, which is a bounding design for RIPD analysis. AREVA fuel designs in use at BFN are bounded by GE13 as documented in the following reference:

0000-0166-0876-R0, T304 RIPD Assessment for Browns Ferry Control Blade-Channel Interface SC 11-05 Seismic Analysis, GE Hitachi Nuclear Energy, April 2013.

RIPDs were subsequently evaluated for operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain and found to be bounded by the reference evaluation.

3.3.5.2 Recirculation Line Break

This accident is the same design basis loss-of-coolant accident as described in Section 6, "Core Standby Cooling Systems," and Section 14, "Plant Safety Analysis." It is assumed that an instantaneous, circumferential break occurs in one recirculation

2 EPIC/ISCOR, GE Propriety Code

loop. The reactor is assumed to be operating at design power with rated recirculation flow at the time of the break.

The recirculation line break LOCA results in short term transient loads which affect those components in the vicinity of the recirculation outlet nozzle. The resulting flow-induced and acoustic loads from the recirculation line break LOCA have been determined for a reference BWR plant using a recent three-dimensional thermal-hydraulic transient analysis computer code (TRACG, GE Propriety Code). Browns Ferry specific loads are then determined by scaling the reference plant loads by the effects of the geometric and thermal-hydraulic parameter differences between the reference plant and Browns Ferry. The flow-induced and acoustic loads resulting from the introduction of the TRACG method of calculation are greater than the results determined for past Browns Ferry evaluations.

In order to determine the impact of off-rated power/flow conditions, load multipliers are calculated normalized to a multiplier of 1.0 for the normal power and rated core flow condition. Thus, the effect of off-rated power and flow conditions or increased core flow is illustrated by the normalized multipliers.

The geometric scaling is based on the dimensional differences between the Browns Ferry reactor configuration and the reference BWR plant in the vicinity of the recirculation outlet and shroud annulus region. The thermal-hydraulic scaling is determined by using ISCOR program. The load multipliers for the selected points on the power flow map are also determined by using the ISCOR program.

The acoustic loads are short term transient impulse loads and are conservatively evaluated by determining a static equivalent load. The flow induced loads are more slowly applied transient loads as compared to acoustic loads; therefore, the loads are not added. The higher load governs structural evaluations. Detailed evaluation of these loads indicated that the acoustic loads resulted in higher loads and are the loads used in structural evaluations.

Flow-induced and acoustic loads for a recirculation line break LOCA from operation at minimum recirculation pump speed typically results in the highest acoustic load regardless of the rated power or core flow; however, the probability of a LOCA occurring at this in-frequent operational condition is extremely remote. Therefore, evaluations are based on the more probable LOCA during maximum extended load line limit (MELLL) and MELLLA+ operation.

An analysis has been performed to evaluate the potential leakage from within the floodable inner volume of the reactor vessel during the recirculation line break and subsequent LPCI reflooding. The two possible sources of leakage are:

a. Jet pump throat to diffuser joint.
b. Jet pump nozzle to riser joint.

The jet pump to shroud support joint is welded and therefore is not a possible source of leakage. The throat to diffuser joints for all jet pumps by analysis are shown to leak no more than a total of 225 gpm. The jet pump nozzle to riser joint by analysis is shown to leak no more than 582 gpm for the pumps through which the vessel is being flooded. The summary of maximum leakage is then:

225 gpm Total leakage through all throat-to-diffuser joints
582 gpm Total leakage through all nozzle-to-riser joints
807 gpm Total maximum rate (Units 2 and 3)
157 gpm Additional leakage through bolted design access hole covers (Unit 1 only)
964 gpm Total maximum rate (Unit 1)

LPCI capacity is sized to accommodate 3000 gpm leakage at these locations. It is concluded that the reactor vessel internals retain sufficient integrity during the recirculation line break accident to allow reflooding the inner volume of the reactor vessel.

3.3.5.3 Steamline Break Accident

The analysis of this accident assumes an instantaneous circumferential break of one main steamline between the reactor vessel and the main steamline flow restrictor. This is not the same accident as that described in Chapter 14, "Plant Safety Analysis," (which postulates a break downstream of the flow restrictors) because greater differential pressures across the reactor vessel internals result from this accident. It is noteworthy that this accident results in greater loading of the reactor vessel internals and a higher depressurization rate than does the recirculation line break. This is because the depressurization rate is proportional to the mass flow rate and the excess of fluid escape enthalpy above saturated water enthalpy. However, mass flow rate is inversely proportional to escape enthalpy, h_e; therefore, depressurization rate is proportional to $1-h_f/h_e$. Consequently, depressurization rate decreases as h decreases, that is, depressurization is less for mixture flow than for steam flow.

The main steam line (MSL) break inside containment is postulated for calculating the RIPDs, except for the steam dryer RIPD for which the steamline break outside containment is postulated, as specified in Section 3.6 of the UFSAR. These faulted loading conditions are analyzed using the LAMB thermal-hydraulic analysis code. The bounding RIPD values for MSL break from the low reactor power/high core flow (point “I” shown on FSAR Figure 3.7-1) or full power/high core flow (point “F” shown on FSAR Figure 3.7-1) operation is determined and used for reactor internals structural evaluations.

The maximum RIPD values for this faulted condition are identified in the following reference:

GEH 002N3430, Revision 0, Task T0304: Reactor Internal Pressure Differences and Fuel Lift Evaluation, March 2015, Table 3.3.1.1.

Note: The fuel type utilized in the reference is GE13, which is a bounding design for RIPD analysis. AREVA fuel designs in use at BFN are bounded by GE13 as documented in the following reference: 0000-0160-0876-R0, T304 RIPD Assessment for Browns Ferry Control Blade Channel Interface Sc 11-05 Seismic Analysis, GE Hitachi Nuclear Energy, April 2013.

These maximum differential pressures are used, in combination with other assumed structural loads as described in Appendix C, to determine the total loading on the various reactor vessel internals. The various internals are then examined to assess the extent of deformation and collapse, if any. Of particular interest are the responses of the fuel bundle, the core support, the guide tubes, and the metal channels around the fuel bundles.

3.3.5.3.1 Core Support

The two considerations important to the core support evaluation are sliding of the core support and buckling of the supporting beams. Evaluations have determined that the core support will not slide under the postulated accident conditions with preload on the holddown bolts. Additional resistance to sliding is provided by aligners which further stabilize the core support.

The core plate buckling pressure is evaluated by a computer program[3] that calculates core plate stiffener beam-buckling capability. It uses the Rayleigh-Ritz energy method to determine the applied moment to begin yielding, and then to buckle a given tee beam. The tee beam models a segment of a BWR core plate with a stiffener beam. The pressure differential across the plate that would have created this moment is calculated for the longest core plate beam.

The MSL break pressure of 28.5 psid (3952 MWt) is within the permissible load allowed by the Buckling Stability Criteria of Table C.2-3(b) for the safety factors in accordance with Appendix C: Sections C.2 and C.2.6 required emergency and faulted load combinations.

3.3.5.3.2 Guide Tubes

Because the guide tube experiences external-to-internal pressure differentials, the guide tubes were examined for buckling under these conditions. For a guide tube with minimum wall thickness and maximum allowed ovality, the pressure which causes buckling is 105 psi compared to the main steam line break design pressure of 28.5 psid (3952 MWt). This pressure is well within the permissible load allowed

3 PIPST01, GE Proprietary Code

by the Buckling Stability Criteria of Table C.2-3(b) for the safety factors in accordance with Appendix C: Sections C.2 and C.2.6 required emergency and faulted load combinations.

It is concluded that the guide tube will not fail under the assumed accident conditions.

3.3.5.3.3 Fuel Channels

The description of the testing and analysis performed for initial core channels is provided in this section. The NRC approval of the channels used with the current AREVA reload is provided in EMF-93-177(P)(A) Revision 1 with the plant and cycle specific analyses reported in the reload specific fuel mechanical design report.

The fuel channel load due to an internally applied pressure was examined utilizing a fixed-fixed beam analytical model under a uniform load. Tests were conducted to verify the applicability of the analytical model. The results indicate that the analytical model is conservative. The fuel channels may deform sufficiently outward to cause some interference with movement of the control rod blade. There are about 15 factors such as fuel channel deformation, core support hole tolerance, top guide beam location, etc., that determine the clearance between the control rod blade and fuel channel. If each of these tolerance factors are assumed to be at the worst extreme of the tolerance range, then a slight interference would develop under an 18 psi pressure difference across the channel wall. At the top of the control rod there is a roller or spacer pad to guide the blade as it is inserted. The clearance between channels is 70 mils less than the diameter of the roller or spacer pad, causing it to slide or skid instead of roll. As the rod is inserted about halfway there is a tendency for the control rod sheath to push inward on the channel. This is a control rod surface to channel surface contact. A "worst case" study indicates a possibility of a 50-mil interference.

The possibility of a worst case developing is extremely remote. A statistical analysis utilizing a normal distribution for each of the 15 variables indicates that no interference occurs within 3 sigma limits, where sigma is the standard deviation in a point distribution of events. Three sigma lies in the 0.995 percentile of probability of nonoccurrence. However, even if interference occurs, the result is negligible. About 1 pound of lateral force is required to deflect the channel inboard 1 mil. The friction force developed is an extremely small percentage of the total force available to the control rod drives.

The above discussion presupposes the control rod has not moved when the fuel channel experiences the largest magnitude of pressure drop. Analysis indicates that the rod is about 70 percent to 90 percent inserted. If the rod is beyond 70 percent inserted, then no interference is likely to develop because all the channel deformation is in the lower portion of the fuel channel, whereas the roller or spacer

pad is at the top of the rod. It is concluded that the main steamline break accident can pose no significant interference to the movement of control rods. Also, the calculated maximum pressure differential across the core is approximately 12 psi below the 42 psi required to lift a fuel bundle.

The AREVA methodology for evaluating the fuel channel deformation due to the internal pressure and irradiation growth makes use of a Monte Carlo analysis to determine the probability of having a stuck control blade condition. A 95/95 statistical statement is made, taking into account variations in core tolerances, fuel channel tolerances, bulge and bow, to demonstrate acceptable interface with the surfaces of the control blades.

Fuel lift is evaluated for AREVA fuel assuming maximum differential pressure conditions. A substantial margin is calculated to exist prior to lift off.

3.3.5.4 Thermal Shock

The most severe thermal shock effects for the reactor vessel internals result from the reflooding of the reactor vessel inner volume. For some vessel internals, the limiting thermal shock occurs from LPCI operation and for others HPCI operation is controlling, dependent upon the location of the component. These effects occur as a result of any large loss-of-coolant accident, such as the recirculation line break and the steamline break accidents previously described.

Three specific locations are of particular interest, as shown in Figure 3.3-9. The locations are as follows:

a. Shroud support plate,

b. Shroud-to-shroud support plate discontinuity, and

c. Shroud inner surface at highest irradiation zone.

The peak strain resulting in the shroud support plate is about 6.5 percent. This strain is higher than the 5.0 percent strain permitted by the ASME Code, Section III, for ten cycles, but for one cycle, peak strain corresponds to about six allowable cycles of an extended ASME Code as applied to less than ten cycles.

Figure 3.3-10 illustrates both the ASME Code curve and the basic material curves from which it was established (with the safety factor of 2 on strain or 20 on cycles, whichever is more conservative). The extension of the ASME Code curve represents a similar criteria to that used in the ASME Code, Section III, but applied to fewer than ten cycles of loading. For this type 304 stainless steel material, a 10 percent peak strain corresponds to one allowable cycle of loading. Even a 10 percent strain for a single-cycle loading represents a very conservative suggested

limit because this has a large safety margin below the point at which even minor cracking is expected to begin. Because the conditions which lead to the calculated peak strain of 6.5 percent are not expected to occur even once during the entire reactor lifetime, the peak strain is considered tolerable.

The results of the analysis of the shroud-to-shroud support plate discontinuity region are as follows:

Amplitude of Alternating Stress ..... 180,000 psi
Peak Strain ..... 1.34 percent

The ASME Code, Section III, allows 220 cycles of this loading, thus no significant deformations result. The most irradiated point on the inner surface of the shroud is subjected to a total integrated neutron flux of 2.7 x 10^{20} nvt (>1 MeV) by the end of plant life. The peak thermal shock stress is 155,700 psi, corresponding to a peak strain of 0.57 percent. The shroud material is type 304 stainless steel, which is not significantly affected by irradiation. The material does experience some hardening and an apparent loss in uniform elongation, but it does not experience a loss in reduction of area. Because reduction of area is the property which determines tolerable local strain, irradiation effects can be neglected. The peak strain resulting from thermal shock at the inside of the shroud represents no loss of integrity of the reactor vessel inner volume.

3.3.5.5 Earthquake

The seismic loads on the RPV and RPV internals are determined from dynamic earthquake analysis described in Section 12.2 using the mathematical models of the RPV and internals shown in Figures 12.2-27B and 12.2-27C. The design of the RPV and internals are described in Section C.4 of Appendix C and Appendices J, K, and L.

RPV Internals and Structural Integrity Evaluations were performed with GE fuel per the following Reference:

002N4782, Revision 0, Task T0303: RPV Internals Structural Integrity Evaluation, GE Hitachi Nuclear Energy, April 2015.

For AREVA fuels, the evaluations were performed per the following Reference:

0000-0166-4147, Revision 0, RPV Internals Structural Integrity Evaluations for AREVA ATRIUM-10 and ATRIUM-10XM Fuels, GE Hitachi Nuclear Energy, January 2014.

3.3.5.6 Conclusions

The analyses of the responses of the reactor vessel internals to situations imposing various loading combinations on the internals show that deformations are sufficiently limited to allow both adequate control rod insertion and proper operation of the Core Standby Cooling Systems. Sufficient integrity of the internals is retained in such situations to allow successful reflooding of the reactor vessel inner volume. The analyses considered various loading combinations, including loads imposed by external forces. Thus, safety design bases 1, 2, and 3 are satisfied.

3.3.6 Inspection and Testing

Quality control methods were used during the fabrication and assembly of reactor vessel internals to assure that the design specifications were met.

The reactor coolant system, which includes the reactor vessel internals, was thoroughly cleaned and flushed before fuel was loaded initially.

During the preoperational test program, operational readiness tests are performed on various systems. In the course of these tests such reactor vessel internals as the feedwater spargers, the core spray lines, the vessel head cooling spray nozzle, and the Standby Liquid Control System line are functionally tested.

Steam separator-dryer performance tests were run to determine carryunder and carryover characteristics on the first 1098 MWe boiling water reactor plant to go into operation. Samples were taken from the inlet and outlet of the steam dryers and from the inlet to the main steamlines at various reactor power levels, water levels, and recirculation flow rates. Moisture carryover was determined from sodium-24 activity in these samples and in reactor water samples. Carryunder was determined from measured flows and temperatures determined by heat balances.

Vibration analysis of reactor vessel internals is included in the design to reduce failures due to vibration. When necessary, vibration measurements are made during startup tests to determine the vibration characteristics of the reactor vessel internals and the recirculation loops under forced recirculation flow. Vibratory responses are recorded at various recirculation flow rates using strain gauges on fuel channels and control rod guide tubes, accelerometers on the shroud support plate and recirculation loops, and linear differential transducers on the upper shroud and shroud head-steam separator assembly. The vibration analyses and tests are designed to determine any potential, hydraulically induced equipment vibrations and to check that the structures will not fail due to fatigue. The structures were analyzed for natural frequencies, mode shapes, and vibrational magnitudes that could lead to fatigue at these frequencies. With this analysis as a guide, the reactor internals were instrumented and tested to ascertain that there were no gross instabilities. The

cyclic loadings are evaluated using as a guide the cyclic stress criteria of the ASME Code, Section III. These field tests were performed only on reactor vessel internals that represented a significant departure from design configurations previously tested and found to be acceptable. Field test data were correlated with the analyses to ensure validity of the analytical techniques on a continuing basis[4].

The reactor vessel and internals are designed to assure adequate working space and access for inspection of selected components and locations[5.] The criteria for selecting the components and locations to be inspected are based on the probability of a defect occurring or enlarging at a given location and includes areas of known stress concentrations and locations where cyclic strain or thermal stress might occur. The reactor vessel internals inspection program is detailed in Subsection 4.12, "Inservice Inspection and Testing."

After installation, the RSDs are evaluated during initial power ascension to assess their structural performance. The RSD for the lead unit is assessed for structural integrity through measured on-dryer strains and acceleration. The RSDs for the follow on units are assessed through the analysis of dryer loads projected from main steam line strain gauge measurements. Reactor parameters that have been indicative of past dryer structural failures are also monitored during initial power ascension.

4 Quad-Cities Station Units 1 and 2 Docket No.'s 50-254 and 265, Amendment 19
5 Brandt, F. A., "Design Provision for In-Service Inspection," General Electric Company. Atomic Power Equipment Department, April 1967 (APED-5450)