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5.0 Reactor Coolant System and Connected Systems

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5.1 Summary Description

5.1.1 General

5.1.1.1 System

The Reactor Coolant System consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnecting piping. The system is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a reactor coolant pump, to the reactor vessel. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector, and a solvent for the soluble poison (boron in the form of boric acid). The system pressure settings are listed in Table 5-1; the integrity of the reactor coolant pressure boundary is described in Section 5.2; the reactor vessel design is described in Section 5.3; and other major components and subsystems in the reactor coolant pressure boundary (RCPB) are described in Section 5.4. The maximum reactor coolant system volume is 12,085 ft^3 for Unit 3 original steam generators, 12,005 ft^3 for Units 2 and 3 replacement steam generators and 11,848 ft^3 for Unit 1 replacement steam generators. (RCS volumes listed assume 0% tube plugging and MK-B11 fuel.)

The Reactor Coolant System piping diagrams are Figure 5-1 (Oconee 1) and Figure 5-2 (Oconee 2 & 3).

In 1970, the Oconee 1 reactor coolant pumps were replaced with Westinghouse Model 93A pumps. The reactor coolant piping was modified slightly to accommodate the replacement pumps. Both the original pumps and the replacement pumps were bottom suction and side discharge allowing installation of the replacement pumps on the same centerlines as the original pumps. The original motors were utilized with the replacement pumps.

Figure 5-3 and Figure 5-4 show the revised arrangement of the reactor coolant piping for Oconee 1.

5.1.1.2 System Protection

Engineered safety features and associated systems are protected from missiles which might result from a loss of coolant accident. Protection is provided by concrete shielding and/or segregation of redundant components.

The reactor vessel is surrounded by a concrete primary shield wall and the heat transport loops are surrounded by a concrete secondary shield wall. These shielding walls provide missile protection for the Reactor Building liner plate and equipment located outside the secondary shielding.

Removable concrete slabs over the reactor vessel area and the concrete deck over the area outside of the secondary shield wall also provide shielding and missile protection.

The Reactor Coolant System is analyzed for maximum hypothetical earthquake to determine that resultant stresses do not jeopardize the safe shutdown of the Reactor Coolant System and removal of decay heat.

5.1.1.3 System Arrangement

The system arrangement in relation to shielding walls, the Reactor Building and other equipment in the building are described in Chapter 1. Plan and elevation drawings showing principal dimensions of the Reactor Coolant System in relation to the supporting or surrounding concrete structures are provided in Figure 5-3, Figure 5-4 (Oconee 1), Figure 5-5, Figure 5-6 (Oconee 2) and Figure 5-7, Figure 5-8 (Oconee 3).

In 2003, the Unit 1 steam generators were replaced with steam generators manufactured by Babcock and Wilcox Canada. In 2004, the steam generators in Unit 2 and Unit 3 were also replaced with steam generators manufactured by Babcock and Wilcox Canada.

5.1.1.4 System Parameters

5.1.1.4.1 Flow

The Reactor Coolant System is designed on the basis of 176,000 gpm flow rate in each heat transport loop.

5.1.1.4.2 Temperatures

Reactor Coolant System temperatures as a function of power are shown in Figure 5-9. The system is controlled to a constant average temperature throughout the power range from 15 percent to 100 percent full power. The average system temperature is decreased between 15 percent and 0 percent of full power to the saturation temperature at 900 psia.

5.1.1.4.3 Heatup

All Reactor Coolant System components are designed for a continuous heatup rate of 100°F/hr.

5.1.1.4.4 Cooldown

All Reactor Coolant System components are structurally designed for a continuous cooldown rate of 100°F/hr. System cooldown to 250°F is accomplished by use of the steam generators and by bypassing steam to the condenser with the Turbine Bypass System. The Low Pressure Injection System provides the heat removal for system cooldown below 250°F.

5.1.1.4.5 Volume Control

The only coolant removed from the Reactor Coolant System is that which is letdown to the High Pressure Injection System. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice.

To maintain a constant pressurizer water level, total makeup to the Reactor Coolant System must equal that which is letdown from the system. Total makeup consists of the seal injection water through the reactor coolant pump shaft seals and makeup returned to the system through the reactor coolant volume control valve (High Pressure Injection System). The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. Reactor coolant volume changes during plant load changes exceed the capability of the reactor coolant volume control valve, and thus result in variations in pressurizer level. The level is returned to normal as the system returns to steady state conditions.

5.1.1.4.6 Chemical Control

Control of the Reactor Coolant Chemistry is a function of the Chemical Addition and Sampling System. Sampling lines from various points off the RCS and Auxiliary Systems provide samples of the reactor coolant for chemical analysis. During normal reactor operation, all chemical addition is made from the Chemical Addition and Sampling System to the High Pressure Injection System. Chemical additions may also be made directly to the RCS via the Pressurizer or the Low Pressure Injection System when the Unit is not at power. See Chapter 9 for detailed information concerning the Chemical Addition and Sampling System and for the High Pressure Injection System.

5.1.1.4.7 Boron

Boron in the form of boric acid is used as a soluble poison in the reactor coolant. Concentrated boric acid is stored in the Chemical Addition and Sampling System and is transported to the Reactor Coolant System in the same manner as described above for chemical addition. The concentrated boric acid may be stored in the concentrated boric acid storage tank (CBAST) or directly in the boric acid mix tank. The CBAST receives concentrated boric acid from the boric acid mix tank. The CBAST is required to contain a specified concentration of boric acid based on the volume in the tank in order to supply a source of concentrated soluble boric acid to the Reactor Coolant System in addition to the borated water storage tank. The concentrated boric acid is pumped to the High Pressure Injection System which transports it to the Reactor Coolant System. Boron concentrations are reduced by running letdown flow through the deborating demineralizers and/or diluting the reactor coolant with demineralized water. All bleed and feed operations for changing the boric acid concentrations of the reactor coolant are made between the High Pressure Injection System and the Coolant Storage System.

5.1.1.4.8 pH

The pH of the reactor coolant is controlled to minimize corrosion of the Reactor Coolant System surfaces which minimizes coolant activity and radiation levels of the components.

5.1.1.4.9 Water Quality

The reactor coolant water chemistry specifications have been selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces. The solids content of the reactor coolant is maintained below the design level by minimizing corrosion through chemistry control and by continuous purification by the demineralizer of the High Pressure Injection System. Excess hydrogen is maintained in the reactor coolant to chemically combine with the oxygen produced by radiolysis of the water.

5.1.1.4.10 Vents and Drains

Vent and drain lines are located at the high and low points of the system and provide the means for draining, filling, and venting the heat transport loops and pressurizer. The reactor vessel cannot be drained below the top of the reactor outlet nozzle using these drain lines. Each vent and drain line contains two manual valves in series. Vent lines are routed to a header connected to the quench tank gas space and drain lines are routed to a header connected to the suction of the component drain pump.

5.1.2 Performance Objectives

5.1.2.1 Steam Output

The Reactor Coolant System is designed to operate at a core power level of 2,568 MWt and transfer a total of 2,584 MWt (including 16 MWt input from reactor coolant pumps) to the steam generators. The system will produce a total steam flow of 10.8 million lbm/hr (replacement steam generators).

5.1.2.2 Transient Performance

The Reactor Coolant System will follow step or ramp load changes under automatic control without relief valve or turbine bypass valve action as follows:

5.1.2.2.1 Step Load Changes

Increasing or decreasing load steps of 10 percent of full power in the range between 20 percent and 90 percent full power.

5.1.2.2.2 Ramp Load Changes

Increasing load ramps of 1 percent per minute between 2 percent and 15 percent, 5 percent per minute between 15 percent and 20 percent, 9.9 percent between 20 percent and 95 percent and 5 percent per minute between 95 percent and 100 percent full power are acceptable. Decreasing load ramps of 9.9 percent between 100 percent and 20 percent, 5 percent per minute between 20 percent and 15 percent and 1 percent per minute between 15 percent and 2 percent full power are acceptable.

The combined actions of the Control System and the Turbine Bypass System permit a 40 percent load rejection or a turbine trip from 40 percent full power without safety valve action. The combined actions of the Control System, the turbine bypass valves, and the main steam safety valves are designed to accept separation of the generator from the Transmission System without reactor trip.

5.1.2.3 Partial Loop Operation

The Reactor Coolant System will permit operation with less than four reactor coolant pumps in operation. The nominal steady-state operating power levels for combinations of reactor coolant pumps operating are as follows:

Reactor Coolant Pumps Operating	Rated Power, %
4	100
3	75

5.1.2.4 Natural Circulation

Natural circulation provides an acceptable method of energy removal from the core with transfer of energy to the Secondary System through the steam generators. The controlling parameters which determine the magnitude of the natural circulation flow rates, i.e., steam generator liquid level and source of feedwater (emergency or main), produce more than adequate circulation rates under steady conditions. The margins to the limits for acceptable operation are more than adequate for steady-state and expected transients.

Natural circulation cooldown mode of operation is not expected to be undertaken at Oconee Nuclear Station except for SBLOCA events which do not allow continued operation of or restart of reactor coolant pumps. In all other situations, procedures recommend that MODE 3 with average Reactor Coolant temperature ≥525°F be maintained until those systems required for forced circulation are put back into service.

In response to Generic Letter 81-21, Duke has developed a procedure to continuously vent the reactor vessel head to containment during a natural circulation cooldown to Decay Heat Removal System conditions. Venting the upper head area will maintain a cooling water flow through the upper head area and prevent the formation of a steam void in this area. This procedure results in a single steam void in the RCS, i.e, in the pressurizer, and simplifies pressure control during cooldown. NRC Safety Evaluation Report (Reference 1) concurs with Duke that natural circulation cooldown is not a safety concern due to operator training and procedures.

5.1.3 References

1. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated June 5, 1985. Subject: NRC Safety Evaluation Report on Duke Response to Generic Letter 81-21 Natural Circulation Cooldown.

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5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design Conditions

5.2.1.1 Pressure

The Reactor Coolant System components are designed structurally for an internal pressure of 2,500 psig.

5.2.1.2 Temperature

With the exception of the components associated with the pressurizer, the Reactor Coolant System pressure boundary components are designed for a temperature of 650°F. The pressurizer and associated code safety valves, power operated relief valve and piping, surge line, sample and drain lines and associated valves, and a portion of the spray line piping are designed for 670°F.

5.2.1.3 Reactor Loads

Reactor Coolant System components are supported and interconnected so that stresses resulting from combined mechanical and thermal forces are within established code limits. Equipment supports are designed to transmit piping rupture reaction loads to the foundation structures.

The Reactor Coolant System supports are on an eight foot six inch thick, heavily reinforced concrete slab which rests on a solid rock subgrade. The minimum ultimate crushing strength of rock cores tested was 720 kips per square foot and the maximum applied dynamic gross load is 30 kips per square foot. Based on the subgrade, the ratio of applied load to bearing capacity of the subgrade, and the monolithic nature of the base slab, differential settlement of the foundation is not anticipated.

5.2.1.4 Cyclic Loads

All Reactor Coolant System components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Design transient cycles are shown in Table 5-2 and Table 5-23.

Flow-induced vibration analyses have been performed for the fuel assembly, including fuel rods, and for the reactor internals components. The analyses and design criteria for the thermal shield, flow distributor assembly, surveillance holder tubes and shroud tubes, and the "U" baffles are given in B&W Topical Report BAW-10051, Reference 1.

Components subjected to cross flow are checked for response during design, so that the fundamental frequencies associated with cross flow are above the vortex shedding frequencies. It has also been conservatively determined that the flow induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc. Emergency operational modes are covered in B&W Topical Report BAW-10008, Part I, Reference 2.

Oconee Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of UFSAR Section 5.2.1.4 cyclic and transient occurrences to assure that components are maintained within design limits. This requirement is managed by the Oconee Thermal Fatigue Management Program.

For license renewal, continuation of the Oconee Thermal Fatigue Management Program into the period of extended operation will provide reasonable assurance that the thermal fatigue analyses, including applicable flaw growth calculations, will remain valid or that appropriate action is taken in a timely manner to assure continued validity of the design.

References for this Section: Application [Reference 38] and Final SER [Reference 39]

5.2.1.5 Seismic Loads and Loss-of-Coolant Loads

Reactor Coolant System components are designated as Class I equipment and are designed to maintain their functional integrity during an earthquake. Design is in accordance with the seismic design bases shown below. The loading combinations and corresponding design stress criteria for internals and pressure boundaries of vessels and piping are given in the section. A discussion of each of the cases of loading combinations follows:

Protection criteria against dynamic effects associated with pipe breaks is covered in Section 3.6. Large reactor coolant loop pipe ruptures (double-ended guillotine breaks) were eliminated for steam generator replacement by the application of leak-before-break-criteria to the reactor coolant loop piping. This was permitted by the NRC as described in Reference 62.

5.2.1.5.1 Seismic Loads

Case I - Design Loads Plus Operating Basis Earthquake (OBE) Loads - For this combination, the reactor must be capable of continued operation; therefore, all components excluding piping are designed to Section III of the ASME Code for Reactor Vessels. The primary piping was originally designed according to the requirements of USAS B31.1 and B31.7. The S_m values for all components, excluding bolting, are those specified in Table N-421 of the ASME code. The S_m value for bolts are those specified in Table N-422 of the ASME Code. The Class I RCS piping was redesigned to the 1983 ASME (No Addenda) Code during the steam generator replacement project.

CASE II - Design Loads Plus Safe Shutdown Earthquake (SSE) - In establishing stress levels for this case, a “no-loss-of-function” criterion applies, and higher stress values than in Case I can be allowed. The multiplying factor of 1.2 has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e., 304 SST, 316 SST, SA302B, SA212B, and SA106C, an acceptable margin of safety will always exist. A more detailed discussion of the adequacy of these margins of safety is given in BAW-10008, Part 1, “Reactor Internals Stress & Deflection Due to LOCA and Maximum Hypothetical Earthquake (MHE)”. Note that the MHE is equivalent to the SSE. The S_m value for all components are those specified in Table N-421 of the ASME Code.

5.2.1.5.2 Loss-of-Coolant Loads

A loss-of-coolant accident coincident with a seismic disturbance has been analyzed to assure that no loss of function occurs. In this case, primary attention is focused on the ability to initiate and maintain reactor shutdown and emergency core cooling. Two additional cases are considered as follows:

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

Case IV - Design loads plus Safe Shutdown Earthquake (SSE) Loads Plus Pipe Rupture Loads - Two thirds of the ultimate strength has been selected as the stress limit for the simultaneous

occurrence of MHE and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin of safety of 50 percent will always exist.

The design allowable stress of Case IV loads is given in BAW-10008 for 304 stainless steel. This curve is used for all reactor vessel internals including bolts. It is based on adjusting the ultimate strength curves published by U.S. Steel to minimum ultimate strength values by using the ratio of ultimate strength given by Table N-421 of Section III of the ASME code at room temperature to the room temperature strength given by U.S. Steel.

In Cases II, III, and IV, secondary stresses were neglected, since they are self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus, the conditions that caused the stresses are assumed to have been satisfied. BAW-10008, Part 1, contains a more extensive discussion of the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals. Table 5-3 provides the stress limits for seismic, pipe rupture, and combined loads.

5.2.1.6 Service Lifetime

A specific service lifetime is not established for the major reactor coolant system components. Rather, through the license renewal process, a detailed aging management review has assured that programs are in place to manage the impact of aging on these components. The number of cyclic system temperature and pressure changes (Table 5-2 and Table 5-23), is also managed and corrective actions taken when appropriate.

5.2.1.7 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of the Reactor Coolant System surfaces. To ensure the best protection is provided, reactor coolant water quality specifications are based upon the most current revision of the EPRI PWR Primary Water Chemistry Guidelines and vendor recommendations. These are addressed in the Chemistry Section Manual.

5.2.1.8 Vessel Radiation Exposure

The reactor vessel is the only Reactor Coolant System component exposed to a significant level of neutron irradiation and is therefore the only component subject to material radiation damage. The maximum predicted exposure from fast neutrons (E > 1.0 MeV) at the inside vessel surface over a 40 and 60-year life with an 82.5 and 80-percent load factor, respectively, has been computed to be as follows (per References 27, 28, 29, 56, 57, and 58):

	33 EFPY	48 EFPY
Oconee Unit 1	9.56×10^{18} neutrons/cm^2	1.31×10^{19} neutrons/cm^2
Oconee Unit 2	9.25×10^{18} neutrons/cm^2	1.28×10^{18} neutrons/cm^2
Oconee Unit 3	9.13×10^{18} neutrons/cm^2	1.26×10^{18} neutrons/cm^2

5.2.1.9 Leak Before Break

Leak-before-break is used at Oconee in three applications. The first application is to establish Mark - B fuel assembly spacer grid impact loads and displacement time histories. It is also used to eliminate the dynamic effects of large bore breaks, thereby allowing the removal of the RCS piping whip restraints. Leak-before-break was also used to justify deletion of all or portions of

the RCS LOCA restraints during steam generator replacement as discussed in Section 5.4.8.6. See Section 3.9.2.4. The second application supports the flaw growth analysis performed for the cast austenitic steel reactor coolant pump inlet and exit nozzles. The third application (discussed below) is used to eliminate the need to analyze for specific pipe ruptures in the CF/LPI Systems for Units 1, 2 and 3. The second and third applications are discussed in the following paragraphs.

The successful application of Leak-Before-Break (LBB) to the Oconee Reactor Coolant System main coolant piping is described in B&WOG topical report entitled, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," BAW-1847, Revision 1, September 1985. This report provides the technical basis for evaluating postulated flaw growth in the main Reactor Coolant System piping under normal plus faulted loading conditions and was approved by the NRC (Reference 62) for the current term of operation. The time-limited aging analyses in BAW-1847, Revision 1, include fatigue flaw growth and the qualitative assessment of thermal aging of cast austenitic stainless steel reactor coolant pump inlet and exit nozzles.

Subsequent analyses by Babcock and Wilcox Canada (Reference 59) demonstrated the acceptability of the weld and piping materials added to the Reactor Coolant System during replacement of the steam generators. This report also increased the acceptable heat up and cool down cycles from 240 to 360 for the LBB evaluation.

Fatigue flaw growth evaluations are based on transient definitions defined by the Reactor Coolant System design specification and described in Table 5-2 and Table 5-23. The transient cycles that make up the thermal fatigue design basis are being monitored by the Oconee Thermal Fatigue Management Program. If a transient cycle count approaches or exceeds the allowable design limit, corrective actions are taken. The cast austenitic stainless steel reactor coolant pump inlet and outlet nozzles are susceptible to thermal aging. Thermal aging of cast austenitic stainless steel causes a reduction of fracture toughness. Reduction of fracture toughness of the reactor coolant pump nozzles has been determined to be acceptable for the period of extended operation through a flaw stability analysis.

The successful application of Leak-Before-Break (LBB) to the Oconee Core Flood Piping System is described in the FANP report entitled, “Leak-Before-Break analysis of the Core Flood and Low Pressure Injection/Decay Heat Removal Piping Systems of Oconee Units 1, 2 and 3 (Reference 67). This report provides the technical basis for evaluating postulated flaw growth in the Core Flood system piping under normal plus faulted loading conditions and was approved by the NRC [Reference 71].

The LBB evaluation concluded that LBB technology is applicable to the Core Flood system piping inside containment for Oconee Units 1, 2 and 3. This includes both CF/LPI piping trains between the two valves (LP-176 and LP-177), the two Core Flood Tank nozzles and up to but not including the two RV Core Flood Nozzles. The margin of 10 on leakage detection, the margin of 2 on postulated crack size and the margin of 1.0 on loads combined by the absolute sum method required by Standard Review Plan 3.6.3 were demonstrated. Therefore, the use of LBB technology to eliminate the dynamic effects of postulated pipe breaks in these systems is justified.

The Core Flood LBB evaluation [Reference 67] eliminates the need to analyze for the dynamic effects associated with pipe ruptures in the identified piping.

References for this Section: [Reference 40], [Reference 41] and [References 67, 68, 69, 70, 71].

5.2.2 Codes and Classifications

The codes listed in this section and Table 5-4 include the code addenda and case interpretations issued through Summer 1967 unless noted otherwise. Quality control and quality assurance programs relating to the fabrication and erection of system components are summarized in Section 5.2.3.11.

The applicable ASME Boiler and Pressure Vessel Code, Section XI Edition and Addenda used for Inservice Inspection shall comply with 10CFR50.55a.

5.2.2.1 Vessels

The design, fabrication, inspection and testing of the reactor vessel and closure head, steam generator (both reactor coolant side and secondary side), pressurizer and attachment nozzles on the vessels is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels.

5.2.2.2 Piping

The design, fabrication, inspection and testing of the reactor coolant piping excluding the pressurizer surge line and the spray line is in accordance with USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February, 1968, and as corrected for Errata under date of June, 1968. The pressurizer surge and spray lines were fabricated and initially inspected in accordance with USAS B31.7, February 1968 with June, 1968, Errata. However, the surge line, which was analyzed in accordance with the ASME Code, 1977 edition, Summer 1979 Addenda, has been reanalyzed to the 1986 ASME Code in response to NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Straitifcation” concerns. The spray line has been reanalyzed to the 1983 Edition of the ASME Code. The following Reactor Coolant Branch lines were analyzed, up to the first isolation valve from the Reactor Coolant Loop, to Class 1 rules of the 1983 Edition, no addenda, of the ASME code:

1. High Pressure Injection (Emergency Injection)
2. High Pressure Injection (Normal Injection)
3. High Pressure Injection (Letdown)
4. Low Pressure Injection (Decay Heat Removal Drop-line)
5. Low Pressure Injection (Core Flood)
6. Reactor Coolant Drains
7. Pressurizer Relief Valve Nozzles

Deleted paragraph(s) per 2004 update

The main feedwater header and the auxiliary feedwater header for the replacement steam generator are designed, fabricated, inspected and tested in accordance with the requirements of the ASME Code for Class I piping components. These piping components meet the ASME stress and fatigue limits as stipulated in NB-3600. The analysis is documented in the Base Design Condition Report (BWC-006K-SR-01) (Reference 60) and the Transient Analysis Stress Report (BWC-006K-SR-01) (Reference 61).

For the analysis of the reactor coolant loop for the replacement of the steam generators, the design code for the RCS piping was changed from the 1968 Edition of the USA Standard B31.7 to subsection NB of Section III of the 1983 Edition of the ASME Boiler and Pressure Vessel Code (no addenda). The basic material allowable stresses used were the lower of those in the 1968 B31.7 or those in the 1983 ASME Boiler and Pressure Vessel Code. Stress allowable factors were taken from the 1983 ASME Boiler and Pressure Vessel Code.

5.2.2.3 Reactor Coolant Pumps

The reactor coolant pump casings are designed, fabricated, inspected and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, but are not code stamped.

5.2.2.4 Relief Valves

The pressurizer code safety valves and the electromatic relief valve comply with Article 9, Section III, of the ASME Boiler and the Pressure Vessel Code.

5.2.2.5 Welding

Welding qualifications are in accordance with the ASME Boiler and Pressure Vessel Code, Section III and Section IX and Section XI, as applicable.

5.2.3 System Design Evaluation

5.2.3.1 Design Margin

The Reactor Coolant System is designed structurally for 2,500 psig and 650°F. The system will normally operate at 2,155 psig and 604°F.

In the event of a complete loss of power to all reactor coolant pumps, reactor coolant flow, coastdown, and subsequent natural circulation flow is more than adequate for core cooling and decay heat removal as shown by the analysis in Chapter 15.

The number of transient cycles specified in Table 5-2 and Table 5-23 for the fatigue analysis is conservative.

5.2.3.2 Material Selection

For Reactor Coolant System major components, materials of construction are listed in Table 5-5. Each of the materials used in the Reactor Coolant System has been selected for the expected environment and service conditions. Materials were chosen for specific uses at various locations within the system because of their compatibility with the reactor coolant. There are no novel material applications in the Reactor Coolant System. Reactor Coolant System materials normally exposed to the coolant are corrosion-resistant materials consisting of 304 or 316 stainless steel, Inconel, Alloy 600, Alloy 690, 17-4PH (H1100), Zircaloy or weld deposits with corrosion resistant properties equivalent to or better than those of 304 SS.

In some specific locations of the Reactor Coolant System, small areas of Low Alloy Steel (LAS) or Carbon Steel (CS) may be exposed to the reactor coolant as a result of RCS component modifications. In each case, a corrosion evaluation was performed demonstrating the component meets the appropriate design Code design requirements for the duration of its service life. Two examples are the mitigation of the Alloy 600 Pressurizer thermowell and the Alloy 600 Pressurizer 1 inch vent nozzle components. Small areas of CS were exposed in the annulus region between the replacement Stainlees Steel vent nozzle and the Pressurizer head (less than 22 sq. in.), as well as the replacement Alloy 690 thermowell and the Pressurizer shell (less than 30 sq. in.). For each design, a corrosion evaluation was performed as documented in a calculation (Reference 72) and ONS License Amendment Request (References 73 & 74). This was approved by the NRC in their Safety Evaluation Report (Reference 75).

To assure long steam generator tube lifetime, feedwater quality entering the steam generator is maintained as high as practical. The current revision of the SGOG EPRI PWR Secondary

Chemistry Guidelines and vendor recommendations are used to prepare operating specifications which are addressed in the Chemistry Section Manual.

The selection of materials and the manufacturing sequence for the Reactor Coolant System components, is arranged to insure that no pressure boundary material is furnace-sensitized stainless steel. Safe ends are provided on those carbon steel nozzles of the system vessels which connect to stainless steel piping. All dissimilar metal welds, with the exception of Inconel to Stainless Steel pipe welds, will be made in the manufacturer's shops.

Piping systems designed to resist seismic forces have been restrained by steel supports capable of withstanding these seismic forces. The restraints also act as pipe stops restraining the lines against whipping. In systems, where it was necessary to use hydraulic or mechanical snubbers to resist seismic forces, the mechanical action associated with the snubbers makes it possible to consider them as restraints against pipe whipping. A more detailed discussion of the types of snubbers in use at Oconee is provided in Section 3.9.3.4.2.2.

The basic design criteria for pipe whip protection is as follows:

1. All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.
2. All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures.
3. All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity.
4. All secondary lines whose break could damage a primary line and also breach containment are designed to maintain containment integrity.

The pressure boundary of the RCS is fabricated primarily from ferritic materials, while that of the attached systems is fabricated primarily from austenitic material.

Consequently, the RCS components are the only ones that require special protection against nonductile failure and that must comply with the fracture toughness requirements of Appendix G to 10 CFR 50. This protection is ensured by establishing pressure-temperature limitations on the RCS. The margin of safety is controlled by not exceeding the calculated allowable pressure at any given temperature. The following loading conditions require pressure-temperature limits:

1. Normal operations including bolt preloading, heatup, and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic tests.
4. Reactor core operation.

For a better understanding of the required protection against non-ductile failure, typical operational parameters of the RCS are described in the following sections for each of the loading conditions.

5.2.3.2.1 Normal Operation

During bolt preload, the reactor vessel closure studs are tensioned to the specified load. Bolt preloading is not allowed until the reactor coolant temperature or the volumetric average temperature of the closure head region (including the studs) is higher than the specified minimum preload temperature. After the studs are tensioned, system pressure can be increased

by the pressurizer until it is above the net positive suction head (NPSH) required for reactor coolant pump (RCP) operation. The heatup transient begins when the RCP is started.

During heatup, the RCS is brought from MODE 5 to MODE 3 with average Reactor Coolant temperature ≥525°F. The heat sources used to increase the temperature of the system are the RCP and any residual (decay) heat from the core. Normally, when the pumps are started, the temperature of the water in the pressurizer is about 400°F; this corresponds to the pressure in the RCS, which is about 300 psig. The coolant temperature is at or above the minimum specified bolt preload temperature.

Initially, the reactor coolant temperature may be as low as room temperature for initial core loading or as high as 130°F for subsequent refueling. The system pressure is maintained below the maximum allowable pressure of approximately 625 psig (20 percent of preoperational system hydrostatic test pressure) until the reactor coolant temperature is approximately 270°F.

At any given time throughout the heatup transient, the temperature of the reactor coolant is essentially the same throughout the system except, of course, in the pressurizer. The system pressure, as controlled by the pressurizer heaters, is maintained between the minimum required for RCP NPSH and the maximum established to meet the fracture toughness requirements. The heatup rate is maintained below the maximum rate used to establish the maximum allowable pressure-temperature limit curve.

RCS cooldown brings the system from MODE 3 to MODE 5. The cooldown is normally accomplished in two phases: The first phase reduces the fluid temperature from approximately 550°F to below the design temperature of the decay heat removal system (approximately 300°F). This temperature reduction is accomplished using the steam generators but bypassing the turbine and dumping the steam directly to the condenser. Once below its design temperature (and pressure), the Decay Heat Removal System (DHRS) is activated in the second phase to further reduce the reactor coolant temperature to that desired.

Before cooldown, the RCS temperature is maintained constant by balancing the heat removal rate from the steam dump with the heat contributed by the RCP and core decay heat. The system pressure is maintained by the pressurizer. The cooldown is normally initiated by stopping two RCPs in one loop. The two remaining pumps provide coolant circulation through both steam generators, and the turbine steam bypass flow controls the cooldown rate. The primary pressure during cooldown is controlled with the pressurizer heaters and spray. After cooling down below the DHRS design temperature and pressure, the cooling mode is changed from the steam generators to the DHRS. Before the switch, the RCS pressure is below 625 psig (20 percent of preoperational system hydrostatic test pressure) and below the DHRS pressure but above the pressure required for the RCP to operate.

To minimize the thermal shock on the RCPB, the two RCP remain in operation as the water flow of the DHRS is initiated. The DHRS flow rapidly mixes with the reactor coolant; but during this period, the indicated RCS temperature may fluctuate until mixing is complete. After the switch is completed, the RCP are stopped. During this phase, the cooldown rate is controlled by the temperature and flow of the DHRS.

5.2.3.2.2 Preservice System Hydrostatic Test

Prior to initial operation, the RCS is hydrostatically tested in accordance with ASME Code requirements. During this test, the system is brought up to an internal pressure not less than 1.25 times the system design pressure. This minimum test pressure is in accordance with Article NB-6000 of ASME Section III. Since the system design pressure is 2500 psig, the preservice system hydrostatic test pressure is 3125 psig. Initially, the RCS is heated to a

temperature above the calculated minimum test temperature required for adequate fracture toughness. This heatup is accomplished by running the RCP. The pressurizer heaters are used to heat the pressurizer to the required temperature. Before the test temperature is reached, the pressure is maintained above NPSH required for the RCP but below the maximum allowable pressure for adequate fracture toughness. When the test temperature is reached, the RCP are stopped and RCS makeup water is added to fill the pressurizer. The test pressure is then reached using either the pressurizer heaters or the hydrostatic pumps connected to the RCS. The test pressure is held for the minimum specified time, and the examination for leakage follows in accordance with the ASME Code.

5.2.3.2.3 Inservice System Pressure Testing

Class 1, 2, and 3 system pressure testing complies with Section XI, Articles IWA-5000, IWB-5000, IWC-5000, and IWD-5000.

5.2.3.2.4 Reactor Core Operation

The reactor core is not allowed to become critical until the RCS fluid temperature is above 525°F except for brief periods of low-power physics testing. This temperature is much higher than the minimum permissible temperature for the inservice system hydrostatic pressure test, and it is also at least 40°F above the calculated minimum temperature required at normal pressure for operation throughout the service life of the plant.

5.2.3.3 Reactor Vessel

The ability of the reactor pressure vessel to resist fracture is the primary factor in ensuring the safety of the primary system in light water cooled reactors. The beltline region of the reactor vessel is the most critical region of the vessel because it is exposed to neutron irradiation. The general effects of fast neutron irradiation on the mechanical properties of such low-alloy ferritic steels as SA302B, Code Case 1339, used in the fabrication of the Oconee 1 reactor vessel, and SA508, Class 2, used in the fabrication of Oconee 2 and 3 reactor vessels, are well characterized and documented in the literature. The low-alloy ferritic steels used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties with a corresponding decrease in ductility after irradiation. In reactor pressure vessel steels, the most serious mechanical property change is the increase in temperature for the transition from brittle to ductile fracture accompanied by a reduction in the Charpy upper-shelf impact strength.

10 CFR 50, Appendix G, “Fracture Toughness Requirements,” specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of water-cooled power reactors and provides specific guidelines for determining the pressure-temperature limitations on operation of the RCPB. The toughness and operational requirements are specified to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Although the requirements of 10 CFR 50, Appendix G, became effective on August 13, 1973, the requirements are applicable to all boiling and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

10 CFR 50, Appendix H, “Reactor Vessel Materials Surveillance Program Requirements”, defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water-cooled reactors resulting from exposure to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel.

These data will permit determination of the condition under which the vessel can be operated with adequate safety margins against fracture throughout its service life.

A method for guarding against brittle fracture in reactor pressure vessels is described in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. This method utilizes fracture mechanics concepts and the reference nil-ductility temperature, RT_{NDT}, which is defined in ASME Section III, Paragraph NB 2331. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve), which appears in Appendix G of ASME Section III. The K_{IR} curve is a lower bound of dynamic, static, and crack arrest fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits can then be determined using the allowable stress intensity factors.

The RT_{NDT} and, in turn the operating limits of a nuclear power plant, can be adjusted to account for the effects of radiation on the properties of the reactor vessel materials. The radiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a surveillance program in which a surveillance capsule containing prepared specimens of the reactor vessel materials is periodically removed from the operating nuclear reactor and the specimens tested. The increase in the Charpy V-notch 30-ft-lb temperature, is added to the original RT_{NDT} along with a margin value to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} is used to index the material to the K_{IR} curve, which, in turn is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

5.2.3.3.1 Stress Analysis

Original Analysis

A stress evaluation of the reactor vessel was initially performed in accordance with Section III of the ASME Code. The evaluation showed that stress levels are within the Code limits.

Table 5-6 lists the reactor vessel steady-state stresses from the initial stress evaluation at various load points. The results of the initial transient analysis and the determination of the initial fatigue usage factor at the same load points are listed in Table 5-7. Calculation OSC-1815 provides the current stress values and fatigue usage factors for the reactor vessel. As specified in the ASME Code, Section III, Paragraph 415.2(d)(6), the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 5-2. Figure 5-10 illustrates the points of stress analysis for the stresses listed in Table 5-6 and the fatigue usage factors listed in Table 5-7.

The initial stress summaries provided in UFSAR Table 5-6 and Table 5-7 demonstrated that all of the requirements for stress limits and fatigue required by ASME Section III for all of the operational requirements imposed by the design specifications were met (the current stress analysis is presented in calculation OSC-1815). The values tabulated in these summaries were the maximum value obtained in each region. The imposed transients are based on description of the realistic behavior that might be expected for this plant. Transients such as loss of flow and load that cause temperature and pressure variations are included in the reactor vessel specification and Table 5-2. Their effect on accumulated usage factor were included in the initial stress analysis as summarized in Table 5-7. These transients were not the major contributors to the largest usage factor of 0.38 for the stud bolts from the initial fatigue evaluation as given in Table 5-7. The current reactor vessel fatigue evaluation provided in OSC-1815 shows that the largest usage factor for the stud bolts remains less than 1.0."

Replacement Steam Generator Analysis

A load comparison and stress analysis was performed for the reactor vessel nozzles and support skirt. All of the locations are acceptable with the replacement steam generators in service. The comparison showed that stress levels are still within the ASME Code limits. The evaluation of the reactor vessel during the replacement steam generator analysis increases the usage factor of the inlet and outlet nozzles, but does not change the acceptance to Section III of the ASME Code.

Calculation OSC-1815 provides the current stress values and fatigue usage factors for the reactor vessel. As specified in the ASME Code, Section III, Paragraph 415.2(d)(6), the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 5-2.

The imposed transients are based on description of the realistic behavior that might be experienced for this plant. Transients such as loss of flow and load that cause temperature and pressure variations are included in the reactor vessel specification and Table 5-2. Their affect on accumulated usage for the reactor vessel were updated during the replacement steam generator analysis (see the current stress analysis presented in calculation OSC-1815). These transients are the major contributing factor to the largest usage factor of 0.61 for the inlet and outlet nozzles. Calculation OSC-1815 shows that all of the usage factors for the reactor vessel components remain below 1.0 after the replacement steam generator analysis.

5.2.3.3.2 Reference Nil-Ductility Temperature (RT_{NDT})

Throughout the lifetime of a reactor vessel, the impact and tensile properties of the ferritic beltline region materials will change because of neutron irradiation. These changes require periodic adjustment of pressure-temperature relationships for heatup and cooldown during normal, upset, and testing conditions.

To determine the pressure-temperature operating limitations for the RCPB the reference nil-ductility temperature (RT_{NDT}) of the ferritic materials must be established. The RT_{NDT} is needed to calculate the critical stress intensity factor (K_{IR}). In ASME Section III, Appendix G, K_{IR} is related to temperature, T, and to RT_{NDT} by the following equation:

$$K_{IR} = 26.777 + 1.223 \quad \exp[0.0145(T - RT_{NDT} + 160)] ksi\sqrt{in}.$$

This relationship is applicable only to ferritic materials that have a specified minimum yield strength of 50,000 psi or less at room temperature.

Since the impact properties of the beltline region materials of a reactor vessel will change throughout its lifetime, periodic adjustments are required on the pressure-temperature limit curves of the RCPB. The magnitude of these adjustments is proportional to the shift in RT_{NDT} caused by neutron fluence. Therefore, it is essential to determine the radiation-induced ΔRT_{NDT} of the beltline region materials.

The RT_{NDT} of the ferritic materials, which were specified and tested in accordance with the fracture toughness requirements of the ASME Section III Summer 1972 Addenda (to 1971 Edition) or subsequent addenda, are determined as required by that Code. When enough material is available, the RT_{NDT} of those beltline region materials, which were specified and tested in accordance with an edition or addenda of ASME Section III prior to the Summer 1972 Addenda, are obtained by testing specimens oriented normal to the principal working direction. The test procedure is in accordance with ASME Section III, paragraph NB 2300 (Summer 1972 Addenda).

The Oconee pressure boundaries were designed and constructed in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda. Except for the beltline region materials for which sufficient test material is available, the RT_{NDT} of the ferritic materials must be estimated. This is necessary because the test data required for the exact determination of RT_{NDT} were not required by the applicable ASME Code.

Generally, drop weight tests were not performed, and the Charpy V-notch tests were limited to "fixed" energy level requirements for specimens oriented in the longitudinal (principal working) direction at a temperature of 40°F or lower.

To obtain an RT_{NDT} estimate that is appropriately conservative, B&W has collected and evaluated the data from tests conducted on pressure-retaining ferritic materials to which the new fracture toughness requirements were applied. Based on these evaluations, techniques were developed to estimate RT_{NDT}. These techniques as well as the results are described in B&W Topical Report BAW-10046P, Reference 4.

10 CFR 50, Appendix G, requires complete characterization of the unirradiated impact properties of all the beltline region materials of the reactor vessel. The complete characterization includes the determination of RT_{NDT} and Charpy (C_v) test curves for the directions normal to and parallel to the principal working direction (other than the thickness direction). Appendix G also requires a minimum C_v USE of 75 ft-lb for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy provide an adequate margin for deterioration from irradiation.

For the beltline region materials of reactor vessels that were specified in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda, the complete C_v test curves, including C_v USE, is determined when the material forms part of the reactor vessel surveillance program. For the beltline region materials that do not form part of the surveillance program, and when enough material is available, the C_v test curve and USE are determined only in the direction normal to the principal working direction. No minimum Charpy V-notch USE are required, other than the 50 ft-lbs/35 mils of lateral expansion for the beltline region materials of these reactor vessels. When the unirradiated USE of these materials is below 75 ft-lb/, the procedures described in BAW-10046P are applied to predict the end-of-service USE.

The C_v USE must be estimated for reactor vessel beltline region materials that were specified in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda and for which insufficient material is available for testing. All available data from tests conducted on reactor vessel beltline region materials were collected and evaluated in order to obtain an appropriately conservative estimate. Not all the data were obtained in accordance with the methods specified in ASME Section III, Appendix G, since in some cases the absorbed energy was obtained only at one temperature. Based on these evaluations, estimates of C_v USE were developed. The techniques and results are described in BAW-10046P.

5.2.3.3.3 Neutron Flux at Reactor Vessel Wall

The design value for the fast neutron flux greater than 1.0 MeV at the inner surface of the reactor vessel is 3.0 x 10^{10} n/cm^2-sec at a rated power of 2,568 MWt. The most recent corresponding calculated maximum fast neutron flux at the vessel wall is approximately a factor of 3 lower. For 40 years at 80 percent load this corresponds to a fluence of approximately 1 x 10^{19} n/cm^2 for the vessel wall.

A semiempirical method is used to calculate the surveillance capsule and reactor vessel flux. The method employs explicit modeling of the surveillance capsule, reactor vessel, and internals and uses a time-weighted average pin-by-pin core power distribution in the two-dimensional DOT IV, version 4.3, computer code. DOT IV is a two-dimensional code which is used to calculate the energy- and space-dependent neutron flux at all points of interest in the specific reactor system configuration. DOT IV employs the discrete ordinates method of solution of the Boltzmann transport equation and has multigroup and asymmetric scattering capability.

The calculational model is an R-theta geometric representation of a plan view through the reactor core midplane using one-eighth core symmetry. The model includes the core with a time-averaged radial power distribution core liner, coolant regions, core barrel, thermal shield, pressure vessel, and concrete. The DOT calculation is carried out with an S_8 order of angular quadrature, a P_3 expansion of the scattering matrix, and the CASK23E cross-section set. The P_3 order of scattering indicates a third order LeGendre polynomial scattering approximation which adequately describes the predominately forward scattering of neutrons observed in the deep penetration of steel and water media. This calculation provides the neutron flux as a function of energy at the detector position and, in addition to the flux, the DOT IV code calculates the saturated specific activity of the various neutron dosimeters located in the surveillance capsule using the ENDF/B5 dosimeter reaction cross-sections. The saturated activity of each dosimeter is then adjusted by a factor which corrects for fraction of saturation attained during the dosimeter's actual detailed irradiation history. Additional corrections are normally made to account for the effects of the following:

1. photon-induced fissions in the U and Np dosimeters,
2. short half-life of isotopes produced in Fe and Ni dosimeters, and
3. Pu-239 generated in the U-238 dosimeter.

These calculated activities are used for comparison with the measured dosimeter activity values. The basic equation for the calculated activity $(\mu Ci/g)$ is

$$D_i = \frac{N}{A_n 3.7x10^4} f_i \sum_E \sigma_n(E)\phi(E) \sum_{j=1} F_j(1-e^{-\lambda_i t_j})e^{-\lambda_i(T-\tau_j)}$$

where:

N	=	Avagadro's number,
A_n	=	atomic weight of target material n,
f_i	=	either weight fraction of target isotope in nth material or fission yield of desired isotope,
$\sigma_n(E)$	=	group-averaged cross sections for material n
$\phi(E)$	=	group-averaged fluxes calculated by DOT analysis,
F_j	=	fraction of full power during jth time interval t_j ,
λ_i	=	decay constant of ith material,
t_j	=	length of the jth time period,

T	=	sum of total irradiation time, i.e., residual time in reactor and wait time between reactor shutdown and counting,
τ_j	=	Cumulative time from reactor startup to end of jth time period, i.e.,

$$\tau_j = \sum_{k=1}^{j} t_k .$$

The flux normalization factor C_i is then obtained by the following equation:

$$C_i = \frac{D_i\,(\text{measured})}{D_i\,(\text{measured})}$$

With C specified, the neutron fluence greater than 1 MeV can be calculated from

$$\phi t(E > 1.0\text{MeV}) = C \sum_{E=1}^{E=15\text{MeV}} \phi(E) \sum_{j=1}^{j=M} F_j t_j$$

where M is the number of irradiation time intervals; the other values are defined above.

The specific results of these calculations are included in the specific capsule evaluation reports prepared as part of the Reactor Vessel Materials Surveillance Program (FSAR Section 5.2.3.13).

5.2.3.3.4 Radiation Effects

The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT}, the unirradiated RT_{NDT}, and a margin value. The predicted ΔRT_{NDT} is calculated using the respective neutron fluence and copper and nickel contents. The design curves of Regulatory Guide 1.99 were used to predict the radiation-induced ΔRT_{NDT} values as a function of the material's copper and phosphorous content and neutron fluence. With the issuance of Rev. 2 of Regulatory Guide 1.99 in May, 1988, ΔRT_{NDT} values are obtained on the basis of copper and nickel contents.

The effects of radiation on the Charpy USE level of the beltline region material is estimated using the curves shown in Regulatory Guide 1.99, Rev. 2, Figure 2.

Several operating plant reactor vessels were manufactured with “high-copper MnMoNi/Linde 80” submerged-arc weld metal. This class of weld metal is susceptible to relatively large changes in impact properties when exposed to fast neutron irradiation. The Charpy V-notch upper-shelf energy (C_vUSE) of some of these welds may drop below the 50 ft-lb threshold required by federal regulatory requirements (10 CFR 50, Appendix G) during the 40-year reactor design life. Should the C_vUSE drop below 50 ft-lb, certain corrective actions would be required that could severely impact plant availability.

One of the major goals of the B&W Owners Group Program has been to determine the period of time each 177-fuel assembly (FA) reactor vessel can operate without violating the 50 ft-lb C_vUSE threshold. The work that has been completed in this program includes reports entitled “Prediction of Charpy Upper Shelf Energy Drop in Irradiated Weld Metals,” “Pressure Vessel Fluence Analysis for 177-FA Reactors”, “Chemistry of B&W 177-FA Owners Group Reactor Vessel Beltline Welds”.

BAW-1803, Rev. 1, Reference 6, describes the implementation of predictive methodology developed in this program to determine the service life to reach the 50 ft-lb C_vUSE threshold for each of the Owners Group reactor vessels. It was also necessary to establish a means of predicting the pre-service C_vUSE of each of the beltline region reactor vessel welds. The available C_vUSE data obtained from B&W manufactured, early vintage welds (high-Cu MnMoNi/Linde 80 submerged-arc) were analyzed collectively for this purpose.

Based on the developed methods, the limiting Oconee reactor welds are predicted to exhibit a C_vUSE of more than 50 ft-lb for >32 EFPY plant operation for the 40-year design life (BAW-2192PA, Reference 30 and BAW-2178PA, Reference 31). Charpy Upper-Shelf Energy Analsysis for 60-year design life are detailed in Section 5.2.3.3.10 and reported in Table 5-27, Table 5-28, and Table 5-29.

5.2.3.3.5 Fracture Mode Evaluation

An analysis has been made to demonstrate that the reactor vessel can accommodate without failure the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at end of vessel design life. A summary of the evaluation follows:

The state of stress in the reactor vessel during the loss-of-coolant accident was evaluated for an initial vessel temperature of 603°F. The inside of the vessel wall is rapidly subjected to 90°F injection water of the maximum flow rate obtainable. The results of this analysis show that the integrity of the vessel is not violated.

The assumed modes of failure are ductile yielding and brittle fracture, which includes the nil-ductility approach and the fracture mechanics approach. The modes of failure are considered separately in the following paragraphs.

<u>Ductile Yielding</u>

The criterion for this mode of failure is that there shall be no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Code, Section III. The analysis considered the maximum combined thermal and pressure stresses through the vessel wall thickness as a function of time during the safety injection. Comparison of calculated stresses to the material yield stress indicated that local yielding may occur in the inner 8.0 percent of the vessel wall thickness.

<u>Brittle Fracture</u>

Because the reactor vessel wall in the core region is subjected to neutron flux resulting in embrittlement of the steel, this area was analyzed from both a nil-ductility approach and a fracture mechanics approach. The results of the two methods of analysis compare favorably and show that pressure vessel integrity is maintained.

The criterion used in the nil-ductility approach is that a crack cannot propagate beyond any point where the applied stress is below the threshold stress for crack initiation (5-8 ksi), or when the stress is compressive. This approach involves making the very conservative assumption that all of the vessel material could propagate a crack by a low-energy absorption or cleavage mode. End-of-life vessel conditions were assumed. The crack arrest temperature through the thickness of the wall was developed on a stress-temperature coordinate system. The actual quench-induced, stress-temperature condition through the thickness of the wall at several times during the quench was developed and plotted. The maximum depth at which the material in the vessel wall would be in tension or at which the stress in the material would be in excess of the threshold stress for crack initiation (5-8 ksi) was determined by comparison of the plots. The

comparison showed that a crack could propagate only through the inner 35 percent of the wall thickness if a crack initiation threshold of 5-8 ksi is applicable.

The foregoing method of analysis is essentially a stress analysis approach which assumes the worst conceivable material properties and a flaw size large enough to initiate a crack. Actually, the outer 83 percent of the vessel wall is at a temperature above the Ductility Transition Temperature (DTT) (NDTT + 60°F) when credit is taken for the neutron shielding, and for the original DTT profile through the wall thickness. The analysis is conservative in that it does not deny that cracks can be initiated, and in that it assumed a crack from 1 to 2 ft long to exist in the vessel wall at the time of the accident. Therefore, it can be concluded that, if a crack were present in the worst location and orientation (such as a circumferentially oriented crack on the inside of the vessel wall), it could not propagate through the vessel wall.

A fracture mechanics analysis was conducted which assumed a continuous surface flaw to exist on the inside surface of the vessel wall. The criterion used for the analysis is that a crack cannot propagate when the stress intensity at the tip of the crack is below the critical crack stress intensity factor (K_{IC}). Topical Report BAW-10018, Reference 7, provides the details of the analysis. This report includes an evaluation considering the Irwin fracture mechanics method and performs a sensitivity analysis of the effect of varying the conservatism of several major parameters on the result.

5.2.3.3.6 Pressurized Thermal Shock

In response to the TMI Action Plan (Item II.K.2.13 "Thermal-Mechanical Report") the effect of cold high pressure injection water entering the reactor vessel during a small break loss of coolant accident or an overcooling transient was considered. The concern was that the cold injection water could rapidly cool the reactor vessel welds and that the resulting thermal stresses, coupled with the relatively high pressure stress on the vessel, would lead to a loss of vessel integrity. This type of event is a particular concern later in life as the vessel neutron fluence increases and the metal becomes more brittle. Various vendor, utility, and EPRI research performed in response to this action item showed that good mixing of the injection water with the warmer Reactor Coolant System fluid would occur, even under near zero loop flow conditions. In particular, the vent valves in the Oconee plant would provide a source of heated water flowing directly from the vessel upper plenum to the downcomer, thus mitigating the cooling effect of the injection flow. The NRC Staff concluded that there is reasonable assurance that vessel integrity would be maintained during a II.K.2.13 event (Reference 12).

The NRC amended its regulations for light water nuclear power plants, effective July 23, 1985, to establish a screening criterion related to the fracture resistance of PWR vessels during PTS events. Only those plants that exceed the screening criterion are required to perform further analysis using Regulatory Guide 1.154. All Oconee units passed the screening criterion (References 32, 35, and 36) and, therefore, met the regulations regarding the PTS concern. This rule was further amended on June 14, 1989, to make the definition of RT_{PTS} equal to RT_{NDT} in Regulatory Guide 1.99, Rev. 2. Assessment in accordance with the amended rule is complete (BAW-2143, Reference 23). All Oconee units satisfy this revised screening criterion.

Section 50.61(b)(1) provides rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility. For license renewal, RT_{PTS} values are calculated for 48 EFPY for Oconee Units 1, 2, and 3.

Section 50.61(c) provides two methods for determining RT_{PTS}: (Position 1) for material that does not have credible surveillance data available, and (Position 2) for material that does have

credible surveillance data. Availability of surveillance data is not the only measure of whether Position 2[1] may be used; the data must also meet tests of sufficiency and credibility.

RT_{PTS} is the sum of the initial reference temperature (IRT_{NDT}), the shift in reference temperature caused by neutron irradiation (ΔRT_{NDT}), and a margin term (M) to account for uncertainties.

IRT_{NDT} is determined using the method of Section III of the ASME Boiler & Pressure Vessel Code. That is, IRT_{NDT} is the greater of the drop weight nil-ductility transition temperature or the temperature that is 60°F below that at which the material exhibits Charpy test values of 50 ft-lbs and 35 mils lateral expansion. For a material for which test data is unavailable, generic values may be used if there are sufficient test results for that class of material. For Linde 80 weld material with the exception of WF-70, the IRT_{NDT} is taken to be the currently NRC accepted values of -7°F or -5°F. For WF-70, the IRT_{NDT} is similarly taken to be a measured value, -26.5°F, in accordance with the discussion and results presented in BAW-2202[2] [Reference 42]. For forgings and plate material, measured values are used where appropriate data is available. Where not available, the generic value of +3°F is used for forgings and +1°F is used for plate material [Reference 43].

For Position 1 material (surveillance data not available), **ΔRT_{NDT}** is defined as the product of the chemistry factor (**CF**) and the fluence factor (**ff**). **CF** is a function of the material's copper and nickel content expressed as weight percent. "Best estimate" copper and nickel contents are used which is the mean of measured values for the material. For Oconee, best estimate values were obtained from the following FTI reports: BAW-1820, BAW-2121P, BAW-2166, and BAW-2222[3] [References 44, 45, 46, and 47]. The value of **CF** is directly obtained from tables in Section 50.61. **ff** is a calculated value[4] using end-of-license (EOL) peak fluence at the inner surface at the material's location. Fluence values were obtained by extrapolation to 48 EFPY of the current 32 EFPY values for each Oconee unit.

For beltline welds and plate materials for which surveillance data is available, evaluations were performed in accordance with Regulatory Guide 1.99, Revision 2, Position 2. The applicable chemistry factors, margin, and RT_{PTS} at 48 EFPY are summarized in Table 5-24, Table 5-25, and Table 5-26.

For Position 2 material (surveillance data available), the discussion above for Position 1 applies except for determination of **CF**, which in this instance is a material-specific value calculated as follows:

1. Multiply each ΔRT_{NDT} value by its corresponding **ff**.
2. Sum these products.

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[1] The term “Position” is taken from Regulatory Guide 1.99, the methodology of which was incorporated into 10 CFR 50.61.

[2] BAW-2202 is an FTI topical report submitted to the NRC for their acceptance on September 29, 1993. The NRC’s acceptance for use at the Zion plants was published in the Federal Register, Vol. 59, No. 40 Page 9782 – 9785, March 1, 1994.

[3] BAW-1820 and BAW-2121P were provided to the NRC for their information. BAW-2166 and BAW-2222 were provided to the NRC as part of the Generic Letter 92-01 program.

[4] $ff=f^{(0.28-0.1*\log f)}$, where $f=\text{fluence}*10^{-19}$ (n/cm^2, E>1MeV).

3. Divide this sum by the sum of the squares of the **ff**s.

The margin term (M) is generally determined as follows:

$$M = 2(\sigma_I{}^2 + \sigma_\Delta{}^2)^{0.5}$$

where σ_I is the standard deviation for IRT_{NDT}

and σ_Δ is the standard deviation for ΔRT_{NDT}.

For Position 1, $\sigma_I = 0$ if measured values are used. If generic values are used, σ_I is the standard deviation of the set of values used to obtain the mean value. For ΔRT_{NDT}, σ_Δ = 28°F for welds and 17°F for base metal (plate and forgings), except that σ_Δ need not exceed one-half of the mean value of ΔRT_{NDT}. For Position 2, the same method for determining the σ values are used except that the σ_Δ values are halved (14°F for welds and 8.5°F for base metal).

Section 50.61(b)(2) establishes screening criteria for RT_{PTS} 270°F for plates, forgings, and axial welds and 300°F for circumferential welds. The values for RT_{PTS} at 48 EFPY are provided in Table 5-24, Table 5-25, and Table 5-26 for Units 1, 2, and 3, respectively. The RT_{PTS} values reported herein are based on updated 48 EFPY fluence projections using the evaluation based methodology described in BAW-2251 [Reference 48, Appendix D] and BAW-2241P [Reference 49]. The chemistry and surveillance data for the beltline materials are reported in BAW-2325 [Reference 50].

The projected RT_{PTS} values for Units 1, 2 and 3 are within the established screening criteria for 48 EFPY. For Unit 1, the limiting weld is SA-1073 with a projected value of RT_{PTS} at 48 EFPY of 230.3°F (screening limit of 270°F). For Unit 2, the limiting weld is WF-25, with a projected value of RT_{PTS} at 48 EFPY of 296.8°F (screening limit of 300°F). For Unit 3, the limiting weld is WF-67 with a projected value of RT_{PTS} at 48 EFPY of 253.5°F (screening limit of 300°F). [Reference 51]

Reference for this section: Final SER [Reference 39].

5.2.3.3.7 Closure (Reactor Vessel)

The reactor closure head flange is bolted to the reactor vessel flange as shown in Figure 5-14. Two hollow metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A line taps into the annulus between the two O-rings to afford a means to test the vessel closure seal after refueling and to monitor for leakage during operation.

After refueling, the vessel closure is tested to verify that it is properly sealed by pressurizing the annulus between the O-rings with demineralized water and monitoring for pressure decay, which indicates leakage. The line that taps into the annulus between the O-rings is configured to serve as both the demineralized water test line and as the drain line.

During steady-state operation and virtually all transient operating conditions, reactor closure head leakage past the metallic O-rings will be negligible. Only in the event of a rapid transient operation, such as an emergency cooldown, would there be some leakage past the inner-most O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min and no leakage would occur past the outer O-ring seal. Leakage past the inner O-ring is monitored by detecting flow through the line that taps into the annulus between the O-rings (or leak-off line). A temperature sensor on the leak-off line is provided with control room indication to monitor for a temperature increase, which indicates leakage flowing through the leak-off line.

The reactor closure head flange is attached to the reactor vessel flange with sixty 6-1/2 in. diameter studs. To insure uniform loading of the closure seal, the studs are hydraulically tensioned. The studs have a minimum yield strength of 130,000 psi. The studs, when tightened for operating conditions, will have a tensile stress of approximately 30,000 psi. An evaluation of stud failures shows that:

1. 10 adjacent studs can fail before leak occurs.
2. 25 adjacent studs can fail before the remaining studs reach yield strength.
3. 26 adjacent studs can fail before the remaining studs reach the ultimate tensile strength.
4. 43 symmetrically located studs can fail before the remaining studs reach yield strength.

The fatigue evaluation results of the studs is included in Table 5-7.

5.2.3.3.8 Control Rod Drive Service Structure

The control rod drive service structure is designed to support the control rod drives to assure no loss of function in the event of a combined loss of coolant accident and maximum hypothetical earthquake. Requirements for rigidity, imposed on the structure to avoid adversely affecting the natural frequency of vibration of the vessel and internals, as well as space requirements for service routing, result in stress levels considerably lower than design limits. The structure is more than adequate to perform its required function.

5.2.3.3.9 Control Rod Drive Mechanism

Appendix G to 10 CFR 50 requires that the adequacy of the fracture toughness properties of ferritic materials such as type 403 modified stainless steel be demonstrated to the Commission on a case-by-case basis. The type 403 modified steel is used as an RCPB material in the motor tube of the control rod drive mechanism. This section demonstrates that, for this application, the material has adequate fracture toughness for protection against non-ductile failure.

The nominal wall thickness of the motor tube section of interest is more than 1/2 inch and less than 5/8 inch. In the early editions of ASME Section III up to the Winter 1971 Addenda to the 1971 Edition, materials with a nominal section thickness of 1/2 inch or less did not require impact testing. Starting with the Summer 1972 Addenda, the nominal section thickness increased to 5/8 inch or less. Thus, in the early editions of ASME Section III, the Type 403 modified steel required impact testing, but in the new editions it does not. However, since this material was selected for use, B&W has ordered it to meet the impact toughness requirements for ASME Section III, Summer 1972 and later Addenda, the imposed acceptance standard for nominal wall thicknesses from 5/8 to 3/4 inch, inclusive is presented in paragraph NB-2332. The material has also been specified to meet the requirements of SA 182 grade F6 (forgings) or ASTM A276 (bars) as modified by ASME Code Case 1337.

When ordered according to the early revisions of Code Case 1337 (including Revision 6) and to the early editions of ASME Section III, the type 403 modified forgings or bars were required to be impact-tested at 20°F. The minimum average energy of a set of three Charpy V-notch specimens was 35 ft-lb, with one specimen allowed to be less than 35 but not less than 30 ft-lb. For both forgings and bars, the Charpy specimens were oriented in the axial (longitudinal) direction.

In the Summer 1972 Addenda to the 1971 Edition of ASME Section III, the fracture toughness requirements of all pressure boundary ferritic materials changed; however, no acceptance criterion was given for the martensitic high-alloy chromium steels, such as type 403 modified steel. A year later, the Summer 1973 Addenda re-established the acceptance criteria for the

type 4XX steels. Beginning with this addenda, the fracture toughness requirements and acceptance criteria for the type 4XX steels are described in paragraph NB-2332 of ASME Section III. This paragraph requires that three Charpy V-notch specimens be tested at temperatures lower than or equal to the lowest service temperature. The lateral expansion of each specimen must be equal to or greater than 20 mils. The test temperature has been specified as equal to or less than 40°F. The orientations of the specimens are transverse (normal to principal working direction) for the forgings and axial for the steel bars.

The fracture toughness requirements of Code Case 1337, starting with Revision 7, are the same as those of ASME Section III, Summer 1973 Addenda to the 1971 Edition.

It is considered that the fracture toughness requirements of the new edition of ASME Section III provide adequate protection against nonductile failure. The proof of adequate toughness is based on demonstrating that the type 403 modified steels used in the construction of components designed to an edition or addenda of ASME Section III prior to the Summer 1973 Addenda meet or exceed the toughness requirements of that addenda.

Based on actual test data, the lowest service temperature of the control rod drive mechanism can be as low as 40°F; however, for additional protection against non-ductile failure, B&W has defined the component's lowest service temperature at 100°F. This specified lowest service temperature is 60°F above the temperature at which the fracture toughness requirements are specified and met. The additional 60°F provides margins of safety beyond that required by the ASME code and by Appendix G to 10 CFR 50.

5.2.3.3.10 Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb ... ." The B&WOG positions on upper shelf energy for 32 EFPY are documented in the responses to Generic Letter 92-01, as reported in BAW-2166 and BAW-2222 and, the low upper shelf toughness analyses documented in BAW-2275 [Reference 52], which is included in BAW-2251 as Appendix B.

Regulatory Guide 1.99, Revision 2 provides two methods for determining Charpy upper-shelf energy (C_VUSE): Position 1 for material that does not have credible surveillance data available and Position 2 for material that does have credible surveillance data. For Position 1, the percent drop in C_VUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE. For Position 2, the percent drop in C_VUSE is determined by plotting the available data on Figure 2 and fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points.

The 48 EFPY C_VUSE values were determined for the reactor vessel beltline materials for each Oconee Unit and are reported in Table 5-27, Table 5-28, and Table 5-29. The T/4 fluence values reported in these tables were calculated in accordance with the ratio of inner surface to T/4 values (i.e. neutron fluence lead factors at T/4) determined in the latest Reactor Vessel Surveillance Program report. As shown in these tables, the C_VUSE is maintained above 50 ft-lb for base metal (plates and forgings), however, for Oconee the C_VUSE for weld metal drops below the required 50 ft-lb level at 48 EFPY. Appendix G of 10 CFR 50 provides for this by allowing operation with lower values of C_VUSE if "it is demonstrated ... that the lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code."

This equivalent margin analysis was performed for 48 EFPY and is reported in BAW-2275 for service levels A, B, C, and D. The analysis used very conservative material models and load combinations, i. e., treating thermal gradient stress as a primary stress. For service levels A and B, the analytical results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated, and again the analytical results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K of the ASME Code. The evaluations for all service levels conclusively demonstrate the adequacy of margin of safety against fracture for the reactor vessels within the scope of this report for 48 EFPY. NRC approval of the analysis in BAW-2275 is included in NUREG-1723 (Reference 39).

5.2.3.3.11 Intergranular Separation in HAZ of Low Alloy Steel under Austenitic SS Weld Cladding

Intergranular separations in low alloy steel heat-affected zones under austenitic stainless steel weld claddings were detected in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice, and clad by high-heat-input submerged arc processes. BAW-10013 contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40-year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separations found in B&W vessels would not lead to vessel failure. This conclusion was accepted by the Atomic Energy Commission[5]. To cover the period of extended operation, an analysis was performed using current ASME Code requirements; this analysis is fully described in BAW-2274 [Reference 53] which is contained in BAW-2251 as Appendix C.

In May 1973, the Atomic Energy Commission issued Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," [Reference 54]. The guide states that underclad cracking "has been reported only in forgings and plate material of SA-508 Class 2 composition made to coarse grain practice when clad using high-deposition-rate welding processes identified as 'high-heat-input' processes such as the submerged-arc wide-strip and the submerged-arc 6-wire processes. Cracking was not observed in clad SA-508 Class 2 materials clad by 'low-heat-input' processes controlled to minimize heating of the base metal. Further, cracking was not observed in clad SA-533 Grade B Class 1 plate material, which is produced to fine grain practice. Characteristically, the cracking occurs only in the grain-coarsened region of the base-metal heat-affected zone at the weld bead overlap." The guide also notes that the maximum observed dimensions of these subsurface cracks is 0.165-inch deep by 0.5-inch long.

The BAW-10013 fracture mechanics analysis is a flaw evaluation performed before the ASME Code requirements for flaw evaluation, the K_{Ia} curve for ferritic steels as indexed against RT_{NDT}, and the ASME Code fatigue crack growth curves for carbon and low alloy ferritic steels were available. The revised analysis uses current fracture toughness information, applied stress intensity factor solutions, and fatigue crack growth correlations for SA-508 Class 2 material. The objective of the analysis is to determine the acceptability of the postulated flaws for 48 EFPY using ASME Code, Section XI, (1995 Edition), IWB-3612 acceptance criteria.

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[5] R. C. DeYoung (USAEC) to J. F. Mallay (B&W), letter transmitting topical report evaluation, October 11, 1972.

The revised analysis was applied to three relevant regions of the reactor vessel: the beltline, the nozzle belt, and the closure head/head flange. The analysis conservatively considered 360 cycles of 100°F/hr normal heatup and cooldown transients. For the power maneuvering transients, the range in applied stress intensity factors for the closure head region were assumed to be the same as that determined for the beltline region. This assumption is considered conservative since the closure head region is subject to a low flow condition while the beltline region is subject to a forced flow condition.

An initial flaw size of 0.353-inch deep by 2.12-inch long (6:1 aspect ratio) was conservatively assumed for each of the three regions. The flaw was further assumed to be an axially oriented, semi-elliptical surface flaw in contrast to the observed flaws which are subsurface with a maximum size of 0.165-inch deep by 0.5-inch long.

The maximum crack growth and applied stress intensity factor for the normal and upset conditions were found to occur in the nozzle belt region. The maximum crack growth, considering all the normal and upset condition transients for 48 EFPY, was determined to be 0.180-inch, which results in a final flaw depth of 0.533-inch. The maximum applied stress intensity factor for the normal and upset condition results in a fracture toughness margin of 3.6 which is greater than the IWB-3612 acceptance criterion of 3.16.

The maximum applied stress intensity factor for the emergency and faulted conditions occurs in the closure head to head flange region and the fracture toughness margin was determined to be 2.24, which is greater than the IWB-3612 acceptance criterion of 1.41. It is therefore concluded that the postulated intergranular separations in the Oconee Unit 1, 2, and 3 reactor vessel 508 Class 2 forgings are acceptable for continued safe operation through the period of extended operation.

5.2.3.4 Steam Generators

Deleted paragraph(s) per 2004 update

<u>Design of Replacement Steam Generators</u>

The replacement steam generators (ROTSGs) for Oconee were manufactured by Babcock and Wilcox Canada. They incorporate the basic Once Through Steam Generator (OTSG) features of the original Oconee steam generators with a number of changes made to improve operation, maintenance, reliability and accident response. The basic operating characteristics in terms of heat transfer, transient response, primary volume, primary pressure loss, secondary inventory, and emergency feedwater performance were not changed significantly from the OTSG. The ROTSGs have the following design features that are considered the major improvements to the OTSG design.

1. Thermally treated Alloy 690 tubes
2. Stainless Steel (410S) broached plate tube supports
3. Elimination of the tube free lane
4. Addition of steam nozzle flow restrictors
5. All welded, erosion-corrosion resistant main and auxiliary feedwater headers
6. Conical support stool to improve access for inspection and ISI

For the replacement steam generators, stress analysis has been completed as documented in detail in the Base Design Condition Report (BWC-006K-SR-01), the Transient Analysis Stress Report (BWC-006K-SR-02) and the "ASME Design Report" (BWC-006K-SR-08). The results, as

compared to data for the original steam generator research and development reported above, are as follows:

During normal heat-up operation of the steam generator, the tube mean temperature should not be more than 80°F higher than the shell mean temperature. The maximum calculated mean tube to shell ΔT at normal operating conditions poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the tubes but would in no way violate the boundary integrity. The rupture of a main steam line would result in an overcooling transient in which the steam generator tubes cool down faster than the steam generator shell. The tubes are then subjected to a tensile load that may cause tube deformation. An analysis of the MSLB accident is performed to determine the input for the steam generator tube stress analysis. The MSLB accident is analyzed with the RETRAN-3D code (Reference 55). A spectrum of break sizes is analyzed from a full power initial condition. The limiting break size is a double-ended guillotine rupture since it maximizes the cooldown rate and the resulting stresses on the steam generator tubes. Main feedwater is isolated to the affected steam generator on low steam line pressure by the Automatic Feedwater Isolation System (AFIS) instrumentation. This circuit also inhibits the auto-start of or auto-stops the turbine-driven emergency feedwater (EFW) pump. The motor-driven EFW pump to the affected steam generator is tripped by the AFIS circuitry when the rate of depressurization setpoint is exceeded coincident with low steam line pressure. For smaller break sizes that do not exceed the rate of depressurization setpoint, operator action is credited at 10 minutes to isolate motor-driven EFW flow to the affected steam generator. The results of the RETRAN analysis, including the primary and secondary system pressures and the tube-to-shell temperature difference were used as input for the steam generator structural analysis. This analysis determined a tube axial load of 2240 lbf for the MSLB. The applicable tube stress acceptance criteria are based on the ASME Code and industry practice. Specifically, the steam generator tubes shall retain a margin of safety against burst of gross failure of three times normal operating differential pressure, or 1.43 times the limiting accident differential pressure. In addition, ASME Section III has established a limit of the lesser of 2.4 x Sm, or 0.7 x Su for design loads. The steam generator tubes have been evaluated for the 2240 lbf MSLB accident load and have been shown to meet these acceptance criteria.

Feedwater line breaks, the tornado event, and other overheating events impose compressive loads on the steam generator tubes as the RCS heats up and/or the steam generator shell cools down. The tornado protection analysis credits a maximum compressive tube-to-shell ΔT of +105 °F while the feedwater line break analysis crediting HPI forced cooling results in a lower compressive tube-to-shell ΔT. Analyses have demonstrated that steam generator tube integrity is maintained for these loads for the replacement steam generators.

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a loss-of-coolant accident. The basis for this analysis is a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet).

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 15,600 psi across the center ligaments which is well below the ASME Section III allowable limit of 45,000 psi at 650°F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress intensity at the juncture of the spherical head with the tube sheet is 16,100 psi which is

well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, the analytical procedure per ASME Code NM-3133.3 is completed to show that for the 5/8 in. o.d./0.034 inch wall Alloy 690 tubing, the allowable Design condition external pressure is 1022 psig. The hypothetical rupture pressure differential of 1050 psi is therefore acceptable considering higher allowable for ASME Level D faulted condition.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2,500 psi across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary (primary head, tube sheet, and tubes).

To meet this criterion, the stress limits delineated in the ASME Pressure Vessel Code, Section III for Design condition. An examination of stresses under these condition show that for the case of a 2,500 psi design pressure differential, the stresses in the tube, primary head and tube sheet are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 5-8.

The ratio of allowable stresses (based on the ASME Code Design condition allowable membrane stress of Sm and allowable membrane plus bending stress 1.5 Sm) to the computed stresses for a design pressure differential of 2,500 psi are summarized in Table 5-9.

5.2.3.5 Reliance on Interconnected Systems

The principal heat removal system interconnected with the Reactor Coolant System is the Steam and Power Conversion System. This system provides capability to remove reactor decay heat for the hypothetical case where all station power is lost. Under these conditions decay heat removal from the reactor core is provided by the natural circulation characteristics of the Reactor Coolant System. The turbine driven emergency feedwater pump supplies feedwater to the steam generators. Cooling water flow to the condenser is provided by the emergency discharge line which discharges to the tailrace of the Keowee Dam. Should the condenser not be available to receive the steam generated by decay heat, which is unlikely in view of emergency discharge line flow, the water stored in the feedwater system can be pumped to the steam generators and the resultant steam vented to atmosphere to provide required cooling. The analysis of the plant component functions credited for coping with the unlikely condition of total loss of station power is presented in Section 8.3.2.2.4.

5.2.3.6 System Integrity

The Reactor Protective System (Chapter 7) monitors parameters related to safe operation and trips the reactor to protect against Reactor Coolant System damage caused by high system pressure. The pressurizer code safety valves prevent Reactor Coolant System overpressure after a reactor trip as a result of reactor decay heat and/or any power mismatch between the Reactor Coolant System and the Secondary System.

As a pump-motor shaft is designed to have a natural frequency at least 20 percent above the critical speed, the shaft is too stiff to respond to any of the lower seismic frequencies. The pump and motor bearings are designed to be capable of meeting the seismic design criteria.

The design specification for the control rod drives requires that the drives be capable of withstanding the seismic loadings within the stress limits for Class I equipment.

The purchase specifications for the Emergency Core Cooling System (ECCS) pumps and valves require that the units be capable of operating under the seismic loads predicted to exist at the building elevations where the units will be located. The equipment supplier has certified that the units, based on tests which exceeded the specification requirements on similar units, do adequately meet the purchase specification requirements for operation under seismic loads. The instrumentation transmitters are tested to demonstrate their suitability for the specified seismic conditions.

The center of gravity for this type of equipment is low and both the pump and the driver are rigidly connected to a structural baseplate which in turn is bolted to the building. This type of equipment is structurally quite rigid and in most instances will accommodate very high “g” loadings.

5.2.3.7 Overpressure Protection

The Reactor Coolant System is protected against overpressure by the pressurizer code safety valves mounted on top of the pressurizer. The capacity of these valves is determined from considerations of: (1) the Reactor Protective System; (2) pressure drop (static and dynamic) between the points of highest pressure in the Reactor Coolant System and the pressurizer; and (3) accident or transient overpressure conditions.

The combined capacity of the pressurizer code safety valves is based on the hypothetical case of withdrawal of a regulating control rod assembly bank from a relatively low initial power. The accident is terminated by high pressure reactor trip with resulting turbine trip. This accident condition produces a power mismatch between the Reactor Coolant System and Secondary System larger than that caused by a turbine trip without immediate reactor trip, or by a partial load rejection from full load.

The Low Temperature Overpressure Protection (LTOP) System protects the reactor vessel from excessive pressures at low temperature conditions. As a result of Generic Letter 88-11 and a review of operating practices at Oconee, the supporting analyses for the LTOP System have been revised.

The following low temperature overpressure events have been evaluated:

1. Erroneous actuation of the High Pressure Injection System.
2. Erroneous opening of the core flood tank discharge valve.
3. Erroneous addition of nitrogen to the pressurizer.
4. Makeup control valve (makeup to the RCS) fails full open.
5. All pressurizer heaters erroneously energized.
6. Temporary loss of the Decay Heat Removal System's capability to remove decay heat from the RCS.
7. Thermal expansion of the RCS after starting a reactor coolant pump, as a result of the stored energy in the steam generators.

The reactor vessel is protected from damage during these events by the LTOP System. The LTOP System consists of two diverse trains. One train consists of the pressurizer power operated relief valve (PORV) with a lift setpoint based on the low temperature pressure limits. The pressure limits for low temperature operation are 100% of the steady-state Appendix G curve. The second train consists of operator action, assisted by administrative controls, alarms,

and an operating philosophy that maintains a steam or gas bubble in the pressurizer during all modes of operation (except for inservice hydrostatic testing).

The pressurizer PORV has a dual setpoint. During normal operation, the lift setpoint is 2450 psig. A lower PORV lift setpoint is used during startup and shutdown conditions. The lower setpoint is enabled by actuation of a switch in the control room whenever the RCS temperature is below 325°F. In order to prevent the LTOP pressure limits from being exceeded, a low pressure setpoint is specified within Technical Specifications.

The second LTOP train relies on operator action to mitigate a low temperature overpressure event. In order to assure that adequate time is available for operator action, administrative controls exist for:

1. RCS pressure;
2. Pressurizer level;
3. Nitrogen addition system;
4. Number of operating reactor coolant pumps;
5. Deactivation of the A and B injection trains of the HPI System;
6. Deactivation of both core flood tanks.
7. A dedicated operator provided with approved procedures monitors RCS pressure and pressurizer level during operations at RCS temperatures below 325°F. The sole duty of the operator is to detect and mitigate LTOP transients before the RCS pressure exceeds the low temperature pressure limits.
8. In addition, alarms are provided to alert the operator that an overpressure event is occurring. The LTOP analysis credits either a RCS pressure or pressurizer level alarm to alert the operator. These alarms ensure that a time is available for the operator to mitigate an overpressure event prior to exceeding the low temperature pressure limits.
9. Deactivation of one bank of pressurizer heaters

The low temperature overpressure scenarios have been analyzed using conservative assumptions (Reference 37). Assuming a single failure of either of the two diverse methods of overpressure protection, the analyses demonstrate that the reactor vessel is protected from damage during events which cause increasing pressure.

The two trains (active and passive) of the LTOP System taken together are single failure proof. The individual trains are not single failure proof.

LTOP System seismic, loss of air, loss of offsite power, and IEEE-279 design requirements are as follows:

1. The active (PORV) and passive (Operator action) LTOP mitigation trains do not have to be seismically designed,
2. A loss of instrumentation air event does not affect the LTOP mitigation trains' ability to mitigate an LTOP event,
3. A loss of offsite power event does not affect the LTOP mitigation trains' ability to mitigate an LTOP event,
4. The LTOP System does not meet IEEE-279 design requirements,

   <u>Because:</u>

1. A pressurizer nitrogen or steam bubble is maintained in the RCS at all times (except for hydrostatic testing).
2. It can be shown that a seismic event, a loss of air event, and a loss of offsite power do not cause an LTOP event.
3. Sufficient administrative controls are in place, per Technical Specifications, to further minimize the probability of an LTOP event.

The above criteria are based on the premise that neither a seismic event nor loss of instrumentation air event nor a loss of offsite power event randomly occur at the same time as an LTOP event at Oconee Nuclear Station.

5.2.3.8 System Incident Potential

Potential accidents and their effects and consequences as a result of component or control failures are analyzed and discussed in Chapter 15.

The pressurizer spray line contains an electric motor-operated backup valve which can be closed should the pressurizer spray valve malfunction and fail to close; this would prevent depressurization of the system to the saturation pressure of the reactor coolant. An electric motor-operated valve located between the pressurizer and the pressurizer electromatic relief valve can be closed to prevent pressurizer steam blowdown in the unlikely event the electromatic relief valve fails to reclose after being actuated. Because of the other protective features in the plant, it is unlikely that the code valves will ever lift during operation. In addition, it is extremely unlikely these valves would stick open, since there is adequate experience to indicate the reliability of code safety valves. The analyses in Chapter 15 bound an opening in the system equal to one pressurizer code safety valve in the open position.

The consequences of crud filling one of the two instrument lines from the flow annulus to the flow transmitters has been evaluated.

No mechanism can be postulated which would completely block one of these lines. The Reactor Coolant System is a very clean system and is continuously filtered to assure that no significant particulate matter is circulated. The boric acid in the coolant is in concentrations about a factor of two below its solubility limit at 70°F and no precipitation would occur. The entire flow monitoring system is essentially stagnant because it is a pressure-sensitive device. There is no flow in the sensing lines to induce material into these lines. Any matter of sufficient size to block the instrument lines would have to penetrate the annulus which is of a smaller size than the instrument lines. Blockage of less than four entry ports to the annulus does not significantly impare the flow reading.

If the assumption is made that the line did become blocked, however, two possible situations would arise. The blockage of the high-pressure line would cause the average flow to appear high as flow decreases. Similarly, if the low pressure line is blocked, the average flow will appear higher than normal as flow is decreased. In both cases, the loss of one pump will not cause trip based on flux-flow if the power is constant at rated power. The results of a single pump coastdown from rated power was analyzed without trip or power runback. The minimum Departure from Nucleate Boiling Ratio (DNBR) reached when the flow has settled to the three-pump steady state values is 1.34.

If power runback from the Integrated Control System (ICS) is assumed, the reactivity added by control rod insertion is sufficient to reduce the power to 89 percent by the time the flow has reached its new value. Therefore, the hypothetical blocking of the instrument line would not cause the core thermal design limit to be exceeded as a result of the loss of one pump from

rated power. These analyses of crud filling one of the two instrument lines from the flow annulus to the flow transmitters are not reflective of the current methods described in Section 15.6. These analyses are being retained for historical purposes only.

5.2.3.9 Redundancy

Each heat transport loop of the Reactor Coolant System contains one steam generator and two reactor coolant pumps. Operation at reduced reactor power is possible with one or more pumps out of service. For added reliability, power to each pump is normally supplied by one of two electrically separated buses. Each of the two pumps per loop is fed from separate buses.

Two core flooding nozzles are located on opposite sides of the reactor vessel to ensure core reflooding water in the event of a single nozzle failure. Reflooding water is available from either the core flooding tanks or the low pressure injection pumps. The high pressure injection lines are connected to the Reactor Coolant System on each of the four reactor coolant inlet pipes.

5.2.3.10 Safety Limits and Conditions

5.2.3.10.1 Maximum Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the Reactor Building atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products to the Reactor Building. The safety limit of 2,750 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III.

5.2.3.10.2 Maximum Reactor Coolant Activity

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the reactor coolant constitutes a hazard only if the amount of activity is excessive and it is released to the environment. The plant systems are designed for operation with activity in the Reactor Coolant Systems resulting from 1 percent defective fuel. Activity would be released to the environment if the reactor coolant containing gaseous activity were to leak to the steam side of the steam generator. Gaseous activity could then be released to the environment by the steam jet air ejector on the main condenser. In 10 CFR 20, maximum effluent concentrations (EC) for continuous exposure to gaseous activity have been established. These ECs will be used as the basis for maximum release of activity to the environment which has unrestricted access.

5.2.3.10.3 Leakage

Reactor Coolant System leakage rate is determined by comparing instrument indications of reactor coolant average temperature, pressurizer water level and letdown storage tank water level over a time interval. All of these indications are recorded. The letdown storage tank capacity is 31 gallons per inch of height, and each graduation on the level recorded represents two inches of tank height.

Reactor Coolant System leak detection is also provided by monitoring the Reactor Building normal sump level and the letdown storage tank level. The Reactor Building normal sump capacity is 15 gallons per inch of height, excluding embedded piping. Since the pressurizer level controller maintains a constant pressurizer level, any Reactor Coolant System volume change due to a leakage would manifest itself as a Reactor Building normal sump level change and/or a corresponding letdown storage tank level change. Alarm indication in the control room for the

Reactor Building normal sump is provided at a low level of 1 inch of water and a high level of 8 inches of water. For the Letdown Storage Tank, alarm (statalarm) indication is provided at a low level of 60 inches of water and a high level of 90 inches of water. Considering the most adverse initial conditions of a low level in the Reactor Building sump and a high level in the letdown storage tank, a 1 gpm leak from the Reactor Coolant System would initiate a Reactor Building sump high level alarm indication in the control room within 3 hours and a letdown storage tank low level alarm indication in the control room within 17 hours. A three gpm leak would be detected in 1/3 the time given above for detection of a one gpm leak. Normally, with the Reactor Building sump level and the letdown storage tank level between their high alarm and the low alarm respectively, these detection times would be reduced.

If the leak allows primary coolant into the containment atmosphere, additional leak detection is provided by the Reactor Building Process Monitoring System and the Reactor Building Area Monitoring System. The sensitivity and time for detection of a Reactor Coolant System leak by any of the radioactivity monitoring systems depends upon reactor coolant activity and the location of the leak. Alarm indication for each sample point in these systems is in the control room.

If the leak is in a steam generator, the leak can be detected by a decrease in the level of the letdown storage tank as described above, Secondary Tritium Analysis, Xenon Analysis, and also by main steam line and condenser air ejector off gas radiation monitors. The sensitivity of the radiation monitors for leak detection depends upon the activity of the Reactor Coolant System.

Class I fluid systems other than the Reactor Coolant System pressure boundary will be monitored for leakage by monitoring the various storage and/or surge tanks for the applicable systems. The Radiation Monitoring System for the station will aid in leak detection of systems containing radioactive fluids. In addition to the above, routine Operator and/or Health Physics radiation surveillance will detect leakage in both radioactive and non-radioactive systems.

Single phase natural circulation can be maintained in the Reactor Coolant System for decay heat removal following a complete loss of station power (Station Blackout Event) if Reactor Coolant System leaks are maintained within limits required for SSF RC makeup system operability. RCS leakage limits are based on the ability of the SSF RC makeup system to prevent RC pump seal failure (Reference resolution to GSI-23) and provide makeup flow for other normal RCS leakage. RCS leakage limits are also based on providing adequate decay heat removal from the RCS using the SSF ASW System. This prevents excessive RCS inventory loss through the pressurizer code safety valves. AC power is assumed available to necessary shutdown equipment within 4 hours from an off-site source or a class 1E on-site source(s).

5.2.3.10.4 System Minimum Operational Components

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. One steam generator is required to be operable prior to criticality as the steam generator is the means for normal decay heat removal at temperatures above 250°F.

A reactor coolant pump or low pressure injection pump is required to be in operation prior to reducing boron concentration by dilution with make-up water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor.

5.2.3.10.5 Leak Detection

The entire Reactor Coolant System is located within the secondary shielding and is inaccessible during reactor operation. Any leakage drains to the Reactor Building normal sump. Any coolant leakage to the atmosphere will be in the form of fluid and vapor. The fluid will drain to the sump and the vapor will be condensed in the Reactor Building coolers and also reach the sump via a drain line from the cooler.

For the reactor coolant pump, the leakage past the middle seal is routed to the Letdown Storage Tank; leakage past the outermost seal is routed to the quench tank or reactor building normal sump.

Locating the actual point of Reactor Coolant System leakage can most readily be accomplished when the reactor is shutdown, thereby allowing personnel access inside the secondary shielding. Location of leaks can then be accomplished by visual observation of escaping steam or water, or of the presence of boric acid crystals which would be deposited near the leak by evaporation of the leaking coolant.

Leakage of reactor coolant into the Reactor Building during reactor operation will be detected by sump level, tank levels, radioactivity, or any combination of these.

All leakage, both reactor coolant and cooling water is collected in the Reactor Building normal sump. The sump water level is indicated and annunciated at high level in the control room. Changes in sump water level are an indication of total leakage. Pursuant to the NUREG 0737, Item II.F.1.5 safety grade redundant level transmitter to the normal and emergency containment sumps have been installed. Both sump levels are indicated and recorded in the control room. Measurement of the letdown storage tank coolant level provides a direct indication of reactor coolant leakage. Since the pressurizer level is maintained constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a decrease in tank level. Both the pressurizer and letdown storage tank coolant levels are recorded in the control room. A comparison of these two recordings over a time period yields the total reactor coolant leakage rate.

Changes in the reactor coolant leakage rate in the Reactor Building may cause changes in the control room indication of the Reactor Building atmosphere particulate and gas radioactivities. The gaseous or the particulate containment atmosphere radioactivity monitors can be used to detect RCS leakage. Radioactivity detection systems are included for monitoring both particulate and gaseous activiities because of their sensitivities and rapid responses to RCS leakage, but have recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contaimination or cladding defects. If there are few fuel elements cladding defects and low levels of activation products, the gaseous or particulate containment atmosphere radioactivity monitors are limited to detech leakage; however, the requirements can be met at the design bases criteria for the detectors. TS 3.4.15 addresses RCS leakage detection instrumentation requirements (Reference 78).

5.2.3.11 Quality Assurance

Assurance that the Reactor Coolant System will meet its design bases insofar as the integrity of the pressure boundary is concerned, is obtained by analysis, inspection, and testing.

5.2.3.11.1 Stress Analyses

Detailed stress analyses of the individual Reactor Coolant System components including the vessel, piping, pumps, steam generators, and pressurizer have been performed for the Design Bases.

For the replacement steam generator analysis the complete reactor coolant system was treated as one entity for the analysis of the effect of the Operating Basis Earthquake (OBE) and the Maximum Hypothetical Earthquake (MHE, also called the SSE) on the piping and nozzle stresses.

Independent thermal and dynamic analyses have been performed to insure that piping connecting to the Reactor Coolant System is of the proper schedule and that it does not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA schedule 160. The reactor coolant pump casing has been completely analyzed including a dynamic analysis separately from the loop to insure that the stresses throughout the casing are below the allowable for all design conditions.

Stress analysis reports required by codes for the several components have been prepared by the manufacturer and reviewed for adequacy by a separate organization.

5.2.3.11.2 Shop Inspection

Inspection and non-destructive testing of materials prior to and during manufacturing in accordance with applicable codes and additional requirements imposed by the manufacturer have been carried out for all of the Reactor Coolant System components and piping. The extent of these inspections and testing is listed in Table 5-10 for each of the components in the system. Shop testing culminates with a hydrostatic test of each component followed by magnetic particle inspection of the component external surface. Piping will be hydrostatically tested in the field and will undergo a final field inspection.

Preoperational mapping of the reactor vessel by ultrasonic examination was accomplished to establish acceptability of the vessel for service. To meet the requirements of IS-232 of Section XI of the ASME Code, the acceptance standards contained in N625.4 of the 1965 edition of Section III of the ASME Code with Addenda through Summer 1967 were used.

Components were cleaned, packaged to prevent contamination, and shipped over a pre-selected route to the site. For materials purchased or manufactured outside of B&W, the results of the material inspection and testing program have been observed or audited by B&W, and audited by the applicant. In addition there was an independent audit by B&W's Nuclear Power Generation Department Quality Assurance Section.

5.2.3.11.3 Field Inspection

Field welding of reactor coolant piping and piping connecting to nozzles is performed using procedures which will result in weld quality equal to that obtained in shop welding. Non-destructive testing of the welds is identical to that performed on similar welds in the shop and is shown in Table 5-10. Accessible shop and field welds and weld repairs in the reactor coolant piping are inspected by magnetic particle or liquid penetrant tests following the system hydrostatic test.

5.2.3.11.4 Testing

The Reactor Coolant System including the reactor coolant pump internals, reactor closure head, control rod drives, and associated piping out to the first stop valve undergoes a hydrostatic test

following completion of assembly. The hydrostatic test is conducted at a temperature 60°F greater than the highest nil-ductility temperature. During the hydrostatic test, a careful examination is made of all pressure boundary surfaces including gasketed joints.

5.2.3.12 Tests and Inspections

This section discusses tests and inspections performed during and after the assembly of the individual components into a completed Reactor Coolant System. These tests and inspections are performed to demonstrate the functional capabilities of the components after assembly into a completed system, to inspect the quality of the system closure weldments, and to monitor system integrity during service.

5.2.3.12.1 Construction Inspection

The coolant piping for each loop is shipped to the field in six subassemblies. The loops are then assembled in the field. In order to accommodate the small fabricating and field installation tolerances, a number of the subassemblies are fabricated with excess length. Thus, the final fitting of the coolant piping is accomplished in the field. The ends with excess length are field machined. All carbon steel-to-carbon steel field welds are back-clad with stainless steel following removal of the backing rings. Consumable inserts are used in stainless-to-stainless welds, such as surge line and some coolant pump welds. All welding is inspected in accordance with requirements of the applicable codes or better.

Welding of the auxiliary piping to Reactor Coolant System nozzles is done to the same standards as the main coolant piping. Consumable inserts are used in all cases.

Cleaning of reactor coolant piping and equipment is accomplished both before and after erection of various equipment. Piping and equipment nozzles will require cleaning in the area of the connecting weldments. Most of the piping and equipment are large enough for personnel entry and are cleaned by locally applying solvents and demineralized water and by wire brush to remove trapped foreign particles. Where surfaces and equipment cannot be reached by personnel entry and have been cleaned in vendor shops to the required cleanliness for operation and appropriately protected to maintain cleanliness during handling, shipping, storage, and installation, further cleaning will not be performed. Appropriate checks to verify maintenance of required cleanliness will be performed prior to operation.

5.2.3.12.2 Installation Testing

The Reactor Coolant System will be hydrostatically tested in accordance with USAS B31.7, Nuclear Power Piping Code. The test pressure will affect all parts of the Reactor Coolant System up to and including means of isolation from auxiliary systems, such as valves and blank flanges. The hydrostatic test will be performed at temperature above Design Transition Temperature.

The Reactor Coolant System relief valves will be inspected and shop-tested in accordance with Section III of the ASME code for Nuclear Vessels. The relief pressure setting will be made during the shop test.

5.2.3.12.3 Functional Testing

Prior to initial fuel loading, the functional capabilities of the Reactor Coolant System components will be demonstrated at operating pressures and temperatures. Measurement of pressures, flows, and temperatures will be recorded for various system conditions. Operation of reactor coolant pumps, pressurizer heaters, Pressurizer Spray System, control rod drive mechanism,

and other Reactor Coolant System equipment will be demonstrated. For descriptions of the various functional tests performed, refer to Chapter 14.

5.2.3.12.4 Inservice Inspection

Inservice examination of ASME Code Class I, 2 and 3 components are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g)(4), to the extent practical within the limitations of design, geometry and materials of construction of the components, except where specific written relief has been granted by the Commission.

Vessels, heat exchangers, pumps, valves, and piping, are classified in accordance with 1OCFR50.55a and NRC Regulatory Guide 1.26. For each ASME code class, systems have been identified which will be examined. Appropriate Duke drawings and documents provide the exact boundaries for each system to be examined.

The examination categories to be used are those listed in Tables IWB, IWC, IWD, and IWF-2500-1 of ASME Section XI. Specific examinations will be identified by an Item Number specified in the Oconee Inservice Inspection Plan.

The examination techniques to be used for inservice inspection include radiographic, ultrasonic, magnetic particle, liquid penetrant, eddy current and visual examination methods.

Repair procedures are prepared as necessary by Duke Power Company Nuclear Generation Department. These procedures are reviewed for compliance with Section XI. Reexamination to Section XI is included in the repair process.

5.2.3.13 Reactor Vessel Material Surveillance Program

The original Oconee design included three reactor vessel surveillance specimen holder tubes (SSHT) located near the reactor inside vessel wall. Each of these SSHT housed two capsules containing reactor vessel surveillance specimens. When failures of the SSHT occurred at other Babcock & Wilcox (B&W) designed plants, the three Oconee units were shut down in succession, starting in March 1976 to inspect the SSHT. The inspection revealed that all of the SSHTs had suffered some damage. To prevent further damage all surveillance capsules and all parts of the SSHT that had failed or were deemed likely to fail during the remainder of that operating cycle were removed from the vessels.

Since the discovery of the damage to the SSHT, B&W has undertaken the design, manufacture and testing of an improved SSHT. SSHT of this improved design were installed in Davis-Besse 1, Crystal River 3 and Three Mile Island 2. (Three Mile Island 2 no longer operating but capsules were salvaged for irradiations at other host plants.) All of these plants have the same basic B&W 177 fuel assembly reactor design as Oconee 1, 2, and 3. The acceptability of the redesigned SSHT has been demonstrated by a test program reviewed and approved by the NRC staff and conducted in conjunction with the hot functional test performed at Davis-Besse 1.

Installation of the redesigned SSHT in the Davis Besse I, Crystal River 3 and Three Mile Island 2 reactor vessels did not present any unusual radiological difficulties because installation was prior to neutron activation of the reactor internals. Studies of methods of installing the redesigned SSHT in the irradiated B&W reactors indicate that substantial installation difficulties will be experienced primarily because precision machining, alignment and inspection must be performed remotely and under water. Although such problems do not in themselves justify relief from a requirement to reinstall the SSHT in Oconee 1, 2, and 3, they would be likely to cause significant radiation to personnel. Based on its experience in removing the SSHT at Three Mile Island 1 and Rancho Seco 1, B&W estimated that installing SSHT in irradiated reactors could

result in personnel exposures totaling about 100 man-rem per reactor. In the interest of maintaining the radiation exposure of plant personnel as low as reasonably achievable, the licensee, in cooperation with B&W and the owners of other B&W 177 fuel assembly plants, has proposed an alternative program that does not require reinstalling the SSHT in Oconee 1, 2, and 3 and the other irradiated B&W plants.

The capsules removed from the Oconee vessels which had damaged SSHT were placed in a host reactor, Crystal River 3, as part of the integrated surveillance program discussed herein. These capsules contain samples of plate or forging material and heat-affected zone material from the vessel beltline as well as weld metal. The weld metal is expected to be controlling because it is more radiation sensitive.

This program includes provisions to provide additional information, if required under 10 CFR 50, Appendix G, Paragraph IV.A.1.b, in addition to the normal requirements of Appendix H.

The plan involves integrating the interrupted surveillance program at Oconee and other plants with the programs for new plants in a manner generally similar to that covered in 10 CFR 50, Appendix H, Paragraph III.B, except that the plants are at different sites. There are three distinct features of this plan.

1. The original surveillance materials from one or more reactors that have been in service will now be irradiated in a new host reactor, that can be fitted with the newly-designed capsule holders on the thermal shield in less time and without significant radiation exposure of the workmen, and
2. There will be more weld metal specimens and some larger fracture mechanics (compact tension or CT) specimens placed in the capsules, and
3. A data-sharing feature in which all available irradiation data for the beltline welds of a given reactor some of which will come from other surveillance programs, will be considered in predicting its adjusted reference temperature and in making any fracture analyses for that reactor. Typically, several of the welds in any one vessel were made with the same weld wire and flux as those used on some other reactors. The data sharing feature is required because the welds in these reactors have high radiation sensitivity due to high copper content and low initial upper shelf energy.

The specific program for Oconee 1, 2, and 3 involved installing the Oconee surveillance capsules in extra locations provided in the Crystal River 3 vessel. This plan accomplished the original purpose of obtaining information on the effect of radiation on material that is representative of the material in the Oconee reactor vessels on a schedule that provides an appropriate lead time over the vessel irradiation rate. The overall integrated program also provides information relevant to Oconee 1, 2, and 3 from surveillance programs in Crystal River 3, and Davis Besse 1 on material considered to be essentially identical to the actual welds in the Oconee vessels. Details are provided below.

5.2.3.13.1 Oconee 1

The limiting weld materials for the Oconee 1 vessel are Procedure Qualification (P.Q.) numbers SA-1426, SA-1430, SA-1229, and SA-1585, except for pressurized thermal shock (PTS) for which the limiting material is SA-1073 (Reference 32)[6] (BAW-2192, BAW-2178, References 30

[6] Weld materials are specifically identified by the ASME Code by the procedure Qualification Test number. A procedure qualification test is required on each combination of heat of weld wire and batch of flux.

and 31). The first two are longitudinal welds in the lower shell course, the second two are beltline circumferential welds, and the last material is a longitudinal weld in the intermediate shell. The end of life (EOL) fluences for these welds are estimated to be 7.67 x 10^{18}, 7.67 x 10^{18}, 8.44 x 10^{18}, 8.99 x 10^{l8}, and 6.55 x 10^{18} (Reference 27) nvt, (E > 1 MeV) at the inner surface, respectively.

The original surveillance material, WF-112, was made using the same heat of filler wire but a different batch of flux as WF-154, one of the radiation sensitive welds in Oconee 2. Metallurgical considerations suggests that the radiation behavior is affected more by the wire than the flux, thus WF-112 is expected to respond to radiation much like WF-154. This data will be a useful part of the data base for B&W vessels.

BAW-1543 (NP), Revision 4, Supplement 6-A, Reference 14 documents where samples of the pertinent weld materials have been or are being irradiated in the integrated program, what kinds of specimens will be used, and when information will be available. The irradiation schedule and withdrawal dates will be modified to optimize the information obtained as indicated to be appropriate as test results are obtained and evaluated. Reference 14 is updated periodically to reflect the most recent capsule reports.

ANP-2650 “Updated Results for Additional Information Regarding Reactor Pressure Vessel Integrity” (Reference 76) dated July 2007 includes the data from capsules tested between January 1999 and May 2007 and provides the following:

1. Credibility and surveillance capsule data chemistry factor assessments for each Linde 80 heat including new capsules since BAW-2325, Revision 1 (Reference 50).
2. Pressurized Thermal Shock (PTS) values for each of the plants participating in the Babcock and Wilcox Master Integrated Surveillance Program (Oconee 1, 2 and 3 included) are updated for the plant’s current licensing period (60 calendar years for plants with license renewal) considering the surveillance data obtained from the new capsules. The PTS values are consistent with the plants current licensing basis (May 2007).
3. Adjusted Reference Temperature (ART) values for each of the participating plant’s current effective P/T curves considering surveillance data obtained from the new capsules (May 2007).

Information from capsules tested prior to January 1999 can be found in BAW-2325 Revision 1 (Reference 50).

5.2.3.13.2 Oconee 2

The limiting weld material for the Oconee 2 vessel is P.Q. number WF-25 which is used in the center circumferential weld. (BAW-2192 and BAW-2178, References 30 and 31). The end of life (EOL) fluence for this weld is estimated to be 8.70 x 10^{18} nvt (E > 1 MeV) (Reference 28) at the inner surface.

The original surveillance material, WF-209-1, while not identical to any of the beltline welds in B&W reactors, is of the same weld wire heat as WF-70 (but different flux lot) and is predicted to be radiation sensitive, based on its copper and nickel contents. Data from WF-209-1 will be a useful addition to the data base for these reactors.

BAW-1543 (NP), Revision 4, Supplement 6-A, Reference 14 documents where samples of the pertinent weld materials have been or are being irradiated in the integrated program, what kinds of specimens will be used, and when information will be available. Reference 14 is updated periodically to reflect the most recent capsule reports.

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3. Adjusted Reference Temperature (ART) values for each of the participating plant's current effective P/T curves considering surveillance data obtained from the new capsules (May 2007).

Information from capsules tested prior to January 1999 can be found in BAW-2325 Revision 1 (Reference 50).

5.2.3.13.3 Oconee 3

The limiting weld material for the Oconee 3 vessel is P.Q. Number WF-67 (BAW-2192 and BAW-2178, References 30 and 31). WF-67 is used for the center circumferential weld (inner 75%). The end of life (EOL) fluence for WF-67 is estimated to be 8.59 x 10^{18} nvt, (E > 1 MeV) (Reference 29) at the inner surface.

The original surveillance material, WF 209-1, is the same as that used in Oconee 2. This discussion of WF-209-1 in 5.2.3.13.2 applies here.

BAW-1543 (NP), Revision 4, Supplement 6-A, Reference 14 documents where samples of the pertinent weld materials have been or are being irradiated in the integrated program, what kinds of specimens will be used, and when information will be available. Reference 14 is updated periodically to reflect the most recent capsule reports.

ANP-2650 "Updated Results for Additional Information Regarding Reactor Pressure Vessel Integrity" (Reference 76) dated July 2007 includes the data from capsules tested between January 1999 and May 2007 and provides the following:

1. Credibility and surveillance capsule data chemistry factor assessments for each Linde 80 heat including new capsules since BAW-2325, Revision 1 (Reference 50).
2. Pressurized Thermal Shock (PTS) values for each of the plants participating in the Babcock and Wilcox Master Integrated Surveillance Program (Oconee 1, 2 and 3 included) are updated for the plant's current licensing period (60 calendar years for plants with license renewal) considering the surveillance data obtained from the new capsules. The PTS values are consistent with the plants current licensing basis (May 2007).
3. Adjusted Reference Temperature (ART) values for each of the participating plant's current effective P/T curves considering surveillance data obtained from the new capsules (May 2007).

Information from capsules tested prior to January 1999 can be found in BAW-2325 Revision 1 (Reference 50).

5.2.3.13.4 Integrated Surveillance Program

BAW-1543 (NP), Revision 4, Supplement 6-A, Reference 14, Supplement to the Master Integrated Reactor Vessel Material Surveillance Program, June 2007, specifies the Oconee specimen capsules that were irradiated in Crystal River 3. These capsules include the weld material and other materials such as plate or forging material samples and weld heat affected zone material samples from the Oconee vessels.

For those welds where no surveillance specimens exist, guidance for predictions is based on 10CFR50.61 and Regulatory Guide 1.99 Rev. 2.

BAW-1543, Rev. 4, February 1993 (Reference 77), presents a "Master Integrated Reactor Vessel Surveillance Program" that provides for additional surveillance capsules which contain tension test, Charpy V-notch, and larger-sized compact fracture specimens of 8 different "Linde 80" weld wire heats (14 different wire/flux combinations). These specimens will provide direct data for those materials represented and will provide a statistical base for those other materials for which archive material is not available. For Oconee-1, the weld wire heat used in SA-1229 was irradiated in 2 supplemental capsules. For Oconee-1 and Oconee-2, WF-25 was irradiated in 7 supplementary capsules. For Oconee-3, WF-67 was irradiated in 6 supplementary capsules.

All Oconee RVSP capsules, except for standby capsules, have been tested, essentially completing the requirement for reactor vessel surveillance irradiations. In addition, the supplementary capsules have provided additional irradiation shift and fracture toughness data.

Research programs being funded by the NRC have provided information on the effect of radiation on these specific weld materials and on several additional Linde 80 weld materials expected to respond to radiation in a similar manner. These programs, Heavy Section Steel Technology (HSST), consist of many tension test, C_v and CT specimens irradiated in a test reactor.

The information developed from the “Master Integrated Reactor Vessel Surveillance Program” and the HSST programs help provide assurance of safety margins against vessel failure per 10 CFR 50, Appendix G.

There are uncertainties involved in applying radiation effects information obtained in other reactors to the Oconee vessels. The major uncertainties involved include:

1. Accuracy of neutron fluence calculations,
2. Magnitude and effect of variation in neutron spectra between reactors,
3. Magnitude and effect of variations in irradiation temperature between reactors,
4. Magnitude and effect of variations in rate of irradiation on material properties.

The effects of these variables have been studied for many years and are discussed below.

1. Neutron flux calculations for the reactor vessel wall and irradiation capsule locations have been developed over many years. The dosimetry used in irradiation capsules has furnished information that was used to check out and refine the calculational methods. It is generally believed that the fast neutron flux and fluence in these locations can be calculated to an accuracy of ±20 percent, particularly if some dosimetry checks are available. Dosimeters from the original Oconee surveillance program were removed and tested for verification of vessel fluence calculations.

It should be emphasized that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence, so uncertainties of 20 to 50 percent influence are not highly significant.

The design of the Oconee vessels, internals and cores is almost identical to that of the other reactors that are used to obtain radiation effects information.

These considerations are the basis for the conclusion that uncertainties in the calculation of neutron fluence are small, and the effect of such uncertainties on the assessment of the radiation effects on the vessel material will also be small.

2. Although differences in neutron energy spectra can cause uncertainties in the effects of radiation on material when evaluated without considering spectrum effects, only very large differences in spectra are significant. The variations from one B&W 177 fuel assembly reactor to another are relatively minor, because they have almost identical geometry.

   The possible differences in neutron spectra that could occur between the B&W power reactors to be involved in the integrated program has been considered. Such effects can be dealt with, if necessary, through the use of neutron damage functions that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects.

3. The effect of the temperature of irradiation has also been the subject of considerable research. It is well known that radiation damage is less severe at 600°F than at 500°F (the temperature range of concern). The differences in effect on the steel appear to be noticeable and should be taken into account if the irradiation temperature difference is over about 25°F. Enough information is known to permit conservative evaluations of the effect of temperature differences of at least 50°F, and probably even 100°F or more. The differences in the temperature of the surveillance capsules and vessel walls between the B&W power reactors involved in the proposed integrated program are estimated to be less than 25°F, and can be conservatively evaluated.

4. The effect of irradiation has also been evaluated by research programs at NRL and other laboratories. The general consensus of experts on this subject is that there will be no major differences in material property changes by irradiation rates varying over 2 to 3 orders of magnitude. However, the differences in the rates of irradiation of specimens in the integrated program and the limiting material in the walls of the affected vessels are less than one order of magnitude, therefore, it is concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

The “Master Integrated Reactor Vessel Surveillance Program” provides the information for Oconee 1, 2, and 3 to comply with 10 CFR 50, Appendix G. It also provides assurance that the uncertainties involved in using data obtained from appropriate surveillance specimens irradiated in other B&W power reactors to establish Oconee 1, 2, and 3 vessel operating limitations are small and can be accounted for by imposition of appropriate margins.

Additionally, the “Master Integrated Reactor Vessel Surveillance Program” provides more useful information than could have been extracted from the original surveillance program. The program also gives results of the kind required to meet 10 CFR 50, Appendix G, Paragraph IV.A.1.b.

An extension of the exemption for Oconee Units I, 2 and 3 from the requirements for an in-vessel material surveillance program as set forth in 10 CFR 50, Appendix H, was requested by the Duke Power Company in January 1982 (Reference 10). In its submittal to the NRC, Duke Power Company stated that at present there were no plans to modify the Surveillance

Specimen Holder Tubes (SSHT's) or the Core Support Assembly on any Operating B&W plant which would change the geometrical similarity of the reactors or preclude the continued irradiation of the surveillance capsules in the host plants. Thus, adequate surveillance information will continue to be obtained for the Oconee units. An evaluation of the Surveillance Capsules removed from operating B&W plants and an evaluation of the reactor vessel fluence were included in the Duke Submittal to demonstrate the adequacy of the Surveillance Program. Duke Power Company submittal concluded that:

1. Based on the Surveillance capsule data obtained on all the B&W-177FA plants to date, it has been demonstrated that the prediction techniques used in establishing the vessel operation limits (i.e., Reg. Guide 1.99, Rev. 2) are conservative.
2. A high degree of accuracy has been demonstrated by B&W in estimation of the reactor vessel fluence using the power histories of the reactors and the dosimetry measurements from the host plants with SSHT's.
3. The Specimen Capsules being irradiated at Crystal River-3 have received neutron fluence greater than the fluence received by the Oconee Reactor Vessels by 7 to 10 EFPYs. The Specimen Capsules are expected to continue to lead the respective reactor vessels accumulated peak fluence for the life of the plant.

NRC granted an extension to the exemption for the Oconee Nuclear Station, Units I, 2 and 3 from the requirement for an in-vessel Material Surveillance program as set forth in 10 CFR 50, Appendix H, for a period of five years in June 1982 (Reference 11). The Commission stated in its safety evaluation that the information derived from the surveillance specimens in the host vessel, relevant to Oconee Nuclear Station Units I, 2 and 3 reactor vessels would be sufficient to provide assurance of safety margins and comply with 10 CFR 50, Appendix G. In addition, the NRC concurred with the Duke position that the dosimetry results have shown that the fluences can be estimated from the power histories with reasonable accuracy and accepted the methodology contained in BAW 1485, June 1978. In June, 2007, the NRC accepted BAW-1543, Rev. 4, Supplement 6-A, and found the program capable of monitoring the effect of neutron irradiation and the thermal environment on the fracture toughness of ferritic reactor vessel beltline materials in the plants that are participating in the material surveillance program. This includes Oconee 1, 2 and 3.

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5.3 Reactor Vessel

5.3.1 Description

The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. The reactor vessel is supported by a cylindrical support skirt.

The original reactor vessel heads were replaced in 2003-2004 because of cracking discovered in a number of penetration nozzles. The replacement reactor vessel closure head is a one piece low alloy steel forging clad with stainless steel. All internal surfaces of the vessel and closure head are clad with stainless steel or nickel-chromium-iron (Ni-Cr-Fe) weld deposit. Cracking in the penetration nozzles was determined to be caused by Primary Water Stress Corrosion Cracking (PWSCC) and was associated with long service at reactor operating temperatures (References 3,4). The replacement penetration nozzles are made from Alloy 690 which is more resistant to PWSCC than the original Alloy 600 nozzles. Ongoing inspection programs are part of the Duke In-Service Inspection Program (See Section 18.3.1.2).

The reactor vessel outlines are shown in Figure 5-14 (Oconee 1), Figure 5-15 (Oconee 2), and Figure 5-16 (Oconee 3). The replacement reactor vessel closure head is shown in Figure 5-33. The general arrangement of the reactor vessel with internals is shown in Figure 4-26 and Figure 4-27. Reactor vessel design data is listed in Table 5-11.

All major reactor vessel nozzles are installed with full penetration welds. All control rod drive and incore instrument nozzles are installed with partial penetration welds. The gasket leakage tap is installed in each reactor vessel flange with a partial penetration weld.

Deleted Paragraph(s) per 2009 Update

The reactor vessels are constructed of a combination of formed plates and forgings. The ring forgings in the reactor vessel shells, other than closure flanges, for Oconee 1, 2, and 3 are identified in Figure 5-14, Figure 5-15, and Figure 5-16. The replacement reactor vessel closure heads are a single piece forging.

The core support assembly is supported by a ledge on the inside of the vessel flanges, and its location is maintained on this elevation by the closure head flange. The core support assembly directs coolant flow through the reactor vessel and core, supports the core, and guides the control rods in the withdrawn position.

The coolant enters the reactor through the inlet nozzles, passes down through the annulus between the thermal shield and vessel inside wall, reverses at the bottom head, passes up through the core, turns around through the plenum assembly, and leaves the reactor vessel through the outlet nozzles.

The vessel has two outlet nozzles through which the reactor coolant is transported to the steam generators and four inlet nozzles through which reactor coolant reenters the reactor vessel. Two smaller nozzles located between the reactor coolant inlet nozzles serve as inlets for decay heat cooling and emergency cooling water injection (core flooding and low-pressure injection engineered safety features functions). The reactor coolant and the control rod drive penetrations are located above the top of the core to maintain a flooded core in the event of a rupture in a reactor coolant pipe or a control rod drive pressure housing. The reactor vessel is vented through the control rod drives.

The bottom head of the vessel is penetrated by instrumentation nozzles. The closure head is penetrated by flanged nozzles which provide for attaching the control rod drive mechanisms and for control rod extension shaft movement.

Guide lugs welded inside the reactor vessel's lower head limit a vertical drop of the reactor internals and core to 1/2 inch or less and prevent rotation about the vertical axis in the unlikely event of a major internals component failure.

The reactor vessel shell material is protected from fast neutron flux and gamma heating effects by a series of water annuli and stainless steel barriers located between the core and the vessel's wall.

5.3.2 Vessel Materials

5.3.2.1 Materials Specifications

The materials used in the reactor vessel are discussed in Section 5.2.3.2 and listed in Table 5-5. The original reactor vessel material properties, as used in licensing Oconee, are presented in Table 5-12 and Table 5-13. Additional material physical properties are presented in Table 5-14. These properties have been updated as new data became available as explained in Section 5.2.3.3.

5.3.2.2 Special Processes for Manufacturing and Fabrication

The reactor vessel and appurtenances are constructed in accordance with the ASME Code, Section III edition and addenda listed in Table 5-4. Processes and materials, including product form used in fabrication of the reactor vessel, are discussed in Section 5.2.3, and were selected to ensure reactor vessel integrity, and to meet regulatory requirements and recommendations. Special or unusual processes not meeting the above requirements were not used in construction of the reactor vessel.

5.3.2.3 Special Methods for Nondestructive Examination

The required nondestructive examinations carried out during fabrication are presented in Table 5-10. These inspections were performed in accordance with procedures meeting the requirements of the edition and addenda of the ASME Code, Section III listed in Table 5-4. Nondestructive examination techniques used were selected to provide adequate sensitivity, reliability, and reproducibility to inspect surfaces and detect internal discontinuities. Acceptance standards were in accordance with the requirements of the ASME Code, Section III for the given product and/or fabrication process.

5.3.3 Design Evaluation

The summary description of the reactor vessel, including major considerations in achieving reactor vessel safety and vessels contributing to the vessel's integrity, is contained in Section 5.2. B&W is the reactor vessel designer and fabricator.

5.3.3.1 Design

The ASME Code, Section III, is the Primary design criteria for the reactor vessel. Chapter 5 describes the reactor vessel design, including construction features and arrangement drawing. Materials of construction are listed in Table 5-5. The design code is given in Table 5-4. Table 5-11 gives the design basis values used in the design.

5.3.3.2 Materials of Construction

The materials of construction for the reactor vessel are listed in Table 5-5. Special requirements, reason for selection, and suitability of the materials used are included in Section 5.2.3. The materials selected have been used extensively in nuclear vessel construction and exhibit well defined properties and serviceability.

5.3.3.3 Fabrication Methods

Fabrication methods used in constructing the reactor vessel are described in Section 5.2.2. The suitability of the fabrication methods is demonstrated by the excellent service history of vessels constructed using these methods.

5.3.3.4 Inspection Requirements

Fabrication inspection requirements imposed on the reactor vessel are summarized in Section 5.2.3.11 and Table 5-10. Preservice and inservice inspection requirements are summarized in Section 5.2.3.12.

5.3.3.5 Shipment and Installation

B&W specified cleanliness requirements during shipment of the reactor vessel to ensure its arrival at the site in satisfactory condition. B&W also provided appropriate instructions and consultation to the owner for onsite cleaning and vessel protection. Temporary protective coatings and/or covers were applied to the vessel during shipment and storage as appropriate for expected environmental conditions. Water chemistry was controlled during initial fill, testing, and operation of the vessel to prevent an environment that may be conducive to material failure.

5.3.3.6 Operating Conditions

The operational limits specified to ensure reactor vessel safety are described in Section 5.2.1. These are compared with normal intended and upset operating conditions in Section 5.2.1. The design transients for the reactor vessel are specified in Section 5.2.1.

5.3.3.7 Inservice Surveillance

A discussion of the reactor vessel material surveillance program is given in Section 5.2.3.13.

5.3.4 Pressure - Temperature Limits

5.3.4.1 Design Bases

B&W Topical Report BAW-10046A, Reference 1, provides the bases for setting operational limits on pressure and temperature. This topical report provides detailed assurance that, throughout the life of the plant, operations will comply to requirements of 10 CFR 50, Appendix G. Regulatory Guide 1.99 is used to predict the effects of neutron irradiation on the beltline region materials. For assurance of compliance with 10 CFR 50, Appendix H, through out the life of the plant, see Section 5.2.3.12.

5.3.4.2 Limit Curves

Topical Report BAW-10046A provides the following information:

1. Procedures and criteria used

2. Safety margins
3. Bases used to determine the limits
4. Procedures that will be used to revise the limits

The limits of pressure and temperature for the following conditions are provided in Technical Specification 3.4.3.

1. Inservice leak and hydrostatic tests
2. Normal operation, including heatup and cooldown
3. Reactor core operation

5.3.5 References

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5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

The reactor coolant pumps installed on Oconee 1 are Westinghouse Model 93A, while those installed on Oconee 2 and 3 are Bingham. The following briefly describes the significant changes for Oconee 1. Except where noted, the Oconee 1 design is the same as that of Oconee 2 and 3. The reactor coolant flow distribution with less than four pumps operating is presented in Table 5-15.

5.4.1.1 Reactor Coolant Pumps (Oconee 1 Only)

Each reactor coolant loop contains two vertical single stage centrifugal-type pumps which employ a 3 stage mechanical seals assembly. A cutaway view of the pump is shown in Figure 5-17 and the principal design parameters for the pumps are listed in Table 5-16. The estimated reactor coolant pump performance characteristic is shown in Figure 5-18. Connections to the pumps are shown on Figure 5-1.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the bottom of the impeller, discharged through passages in the guide vanes and out through a discharge in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are constructed of austenitic stainless steel or equivalent corrosion resistant materials. Reactor coolant pump materials of construction are listed in Table 5-5.

The Shaft Seal System consists of face-type mechanical Seals operating in tandem. The shaft seal system is made up of three mechanical seals operating in Tandem, wherein about one-third of the system pressure is expanded in each seal. Each seal is capable of operation at full system pressure. The fluid which leaks past the face type mechanical seal passes in to a seal leakage chamber and out to the quench tank. A low pressure mechanical seal prevents the escape of fluid to atmosphere.

A portion of the high pressure water flow from the high pressure injection pumps is injected into the reactor coolant pump between the impeller and the mechanical seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the mechanical seal, and finally out of the pump. A small amount which leaks through the final seal is also collected and removed from the pump.

Component cooling water is supplied to the thermal barrier cooling coil. The pump may be operated with loss of either injection water or cooling water per GSI-23.

5.4.1.2 Reactor Coolant Pumps (Oconee 2 & 3)

The reactor coolant pumps are single suction, single stage, vertical, radially balanced, constant speed centrifugal pumps. This type of pump employs mechanical seals to prevent reactor coolant fluid leakage to the atmosphere. A view of the pump is shown in Figure 5-19 and the principal design parameters are listed in Table 5-17. The estimated reactor coolant pump performance characteristics are shown in Figure 5-20. Connections to the pumps are shown on Figure 5-21 (Oconee 2) and Figure 5-22 (Oconee 3).

The pump casing design utilizes a quad-volute inner case permanently welded to a pressure containing outer case. The configuration of the pressure containing outer case is kept simple so that the casing quality will meet the required radiographic level and the stresses can be analyzed to meet the requirements of the design specification. The quad-volute inner casing consists of four volute passages spaced 90° apart which receive the discharge from the pump impeller and guide it efficiently into the outer casing where it flows to the discharge nozzle through a passage having a constantly increasing cross-sectional area. The pump casing is welded into the piping system and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

The pump cover and stuffing box is a unit containing a thermal barrier, recirculation impellers, shaft, journal bearing, and mechanical face-type seals. The pump shaft is coupled to the motor with a spacer coupling which will permit removal and replacement of the seals without removing the motor. The pump cover has a cooling jacket to remove the heat which passes through the thermal barrier. This jacket has a capacity large enough to remove all heat which is transmitted to the cover. However, additional cooling capacity is provided, in case injection cooling water is lost. A recirculation impeller on the shaft immediately above the journal bearing circulates water in the bearing chamber to a heat exchanger and returns it to the chamber. The pump may be operated with loss of either injection water or cooling water per GSI-23.

The Shaft Seal System consists of face-type mechanical seals operating in tandem. Injection water, at a pressure above the pump suction pressure, is injected into the pump bearing chamber. A small portion of the injection water flows into the pump through a restriction bushing. The major portion flows through cooling slots in the o.d. of the bearing steel. The shaft seal system is made up of three mechanical seals operating in tandem, wherein about one-third of the system pressure is expanded in each seal. Each seal is capable of operation at the full system pressure. The fluid which leaks past the face-type mechanical seal passes into a seal leakage chamber and then out to the quench tank. A low pressure mechanical seal at the top of the seal leakage chamber prevents the escape of fluid to the atmosphere.

Electroslag welding is used to make the seven-inch thick circumferential butt weld which welds together the upper and lower halves of the pump casing. This weld is performed in accordance with ASME Code Case 1355-2 which permits electroslag welding of Class A pressure vessels. The casings are cast and welded by ESCO, who is the leading supplier of RCP casings for the industry.

Electroslag welding is a welding process wherein coalescence is produced by heat generated in a conductive molten slag which melts the filler metal and the surfaces of the work to be welded. The weld pool is shielded by this slag and moves along the full cross section of the joint as the welding progresses. The conductive slag is maintained molten by its resistance to the flow of electric current passing between the electrode and the work. Water cooled, non-fusing metal shoes are used to contain the molten metal on both sides of the weld. The welding is performed in a vertical position with the start and finished performed on run-out tabs affixed to the casting. These run-out tabs are later cut off and discarded. The only variables contained in the method of welding are the wide range of amperage (480-units 720H) and voltage (44-52V) needed to control the molten pool of metal.

The weld is examined 100 percent using liquid penetrant and radiographic examination methods in accordance with Section III of the ASME Code. Ultrasonic inspection is not performed because the pump casing material, austenitic stainless steel, precludes achieving meaningful inspection results.

The pump casing receives two heat treatment cycles. The first is a solution annealing treatment where the pump casing halves are furnace heated to 1900°F, held for a specified time, and

water quenched. The second heat treatment is a stabilizing treatment in which the welded pump casing is heated to 725°F and air cooled.

Three types of analyses are performed on the pump to verify compliance with ASME Section III: thermal, stress and closure. The first two types are performed using mathematical models of the structure which are analyzed with computer techniques, the third using a merging of preceeding- math model results using an assumption of displacement compatibility at contiguous boundaries. The approaches and computer programs are examined in greater length in calculation OSC-1812, Section 3.0 and Section 4.0.

In the analysis, to determine temperatures throughout the pump, the pump is broken into two mathematically modeled sections which are analyzed using the THAN thermal analysis program. The first model is of typical pump casing wall section, transient analysis are performed on this section. The second model is of the cover. A steady state thermal analysis is made of this region for both wet and drained cooling jacket conditions.

The stress calculations utilize the STARDYNE I, Wilson Jones and NAOS computer programs. Stresses are below their nominal allowables stated in the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition with addenda through summer, 1970.

A summary of the code allowables and maximum stresses is listed in Table 5-18 and shown pictorially on Figure 5-23 and Figure 5-24. The reinforcement area is as defined in paragraph N-454 of the ASME Code Section III. The stress analysis performed on the bowl and the attached nozzles showed that the stresses are within the allowable limits. Note that a factor of two was applied to the nozzle loading due to seismic reactions and when these were combined with the dead weight and thermal expansion reactions, the stress levels were within the realistic allowable stress intensities shown in Table 5-18.

The casing cover analysis indicates that the thermal stresses and pressure stresses on the cover are within the Section III code allowables.

There are no deviations from the applicable ASME Code requirements in the design and fabrication of the pump casings other than code stamping.

5.4.2 Steam Generator

The steam generator general arrangement is shown in Figure 5-25. Principal design data are tabulated in Table 5-20.

The once-through steam generator supplies superheated steam and provides a barrier to prevent fission products and activated corrosion products from entering the Steam System.

The steam generator is a vertical, straight tube, tube and shell heat exchanger which produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the hemispherical heads, the tube sheets and the tubes between the tube sheets. Tube support plates maintain the tubes in a uniform pattern along their length. The replacement steam generators are supported by a pedestal.

The shell, the outside of the tubes, and the tube sheets form the boundaries of the steam producing section of the vessel. Within the shell, the tube bundle is surrounded by a cylindrical baffle. There are openings in the baffle at the feedwater inlet nozzle elevation to provide a path for steam to afford contact feedwater heating. The upper part of the annulus formed by the baffle plate and the shell is the superheat steam outlet, while the lower part is the feedwater inlet heating zone.

Vent, drain, and instrumentation nozzles, and inspection handholes are provided on the shell side of the unit. The reactor coolant side has manway openings in both the top and bottom heads. The replacement steam generators have a flat bottom lower head that eliminated the need for a drain nozzle. Venting of the reactor coolant side of the unit is accomplished by a vent connection on the reactor coolant inlet pipe to each unit.

Feedwater or Emergency Feedwater is supplied to the steam generator through an emergency feedwater ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all reactor coolant pumps.

Four heat transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet these are:

5.4.2.1 Feedwater Heating Region

Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into the downcomer annulus formed by the shell and the cylindrical baffle around the tube bundle. Steam is drawn by aspiration into the downcomer and heats the feedwater to saturation temperature.

The saturated water level in the downcomer provides a static head to balance the static head in the nucleate boiling section, and the required head to overcome pressure drop in the circuit formed by the downcomer, the boiling sections, and the bypass steam flow to the feedwater heating region. The downcomer water level varies with steam flow from 15 - 100 percent load. A constant minimum level is held below 15 percent load.

5.4.2.2 Nucleate Boiling Region

The saturated water enters the tube bundle just above the lower tube sheet and the steam-water mixture flows upward on the outside of the tubes counter current to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until DNB is reached, and then film boiling and super heating occurs.

5.4.2.3 Film Boiling Region

Dry saturated steam is produced in the film boiling region of the tube bundle.

5.4.2.4 Superheated Steam Region

Saturated steam is raised to final temperature in the superheater region. The amount of surface available for superheat varies inversely with load. As load decreases the superheat section gains surface from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat transfer regions vary. Changes in temperature, pressure, and load conditions cause an adjustment in the length of the individual heat transfer regions and result in a change in the inventory requirements. If the inventory is greater than that required, the pressure increases. Inventory is controlled automatically as a function of load by the feedwater controls in the Integrated Control System.

Steam Generator Feedwater quality is addressed in the Chemistry Section Manual.

5.4.3 Reactor Coolant Piping

The general arrangement of the reactor coolant piping is shown in Figure 5-3, Figure 5-4, Figure 5-5, Figure 5-6, Figure 5-7, and Figure 5-8. Principal design data are tabulated in Table 5-21.

The major piping components in this system are the 28-inch i.d. cold leg piping from the steam generator to the reactor vessel and the 36-inch i.d. hot leg piping from the reactor vessel to the steam generator. Also included in this system are the 10-inch surge line and the 2-1/2-inch spray line to the pressurizer. The system piping also incorporates the auxiliary system connections necessary for operation. In addition to drains, vents, pressure taps, injection, and temperature element connections, there is a flow meter section in each 36-inch line to the steam generators to provide a means of determining the flow in each loop.

The 28-inch and 36-inch piping is carbon steel clad with austenitic stainless steel. Short sections of 28-inch stainless steel transition piping are provided between the pump casing and the 28-inch carbon steel lines.

For Oconee 1 only a 28 in. i.d. x 31 in. i.d. stainless steel transition section is installed between the existing 28 in. i.d. coolant piping and the 31 in. i.d. pump suction.

Also a 28 in. i.d. small angle elbow section between the pump discharge nozzle and the reactor inlet pipe is installed to account for the radial discharge of the replacement pump. The original pump had a tangential discharge nozzle. The elbow section is carbon steel with a section of stainless for welding to the pump casing nozzle.

Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe-ends are designed so that there will not be any furnace sensitized stainless steel in the pressure boundary material. This is accomplished either by installing stainless steel safe-ends after stress relief or using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic stainless steel. All piping connections in the Reactor Coolant System, larger than 2 inch diameter, are butt-welded except for the flanged connections on the pressurizer relief valves.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the following nozzles: the four high pressure injection nozzles on the reactor inlet pipes; the two core flooding low pressure injection nozzles on the reactor vessel; and the surge line nozzle and spray line nozzle on the pressurizer.

As part of the Steam Generator Replacement Project, the lower cold leg 45^{o} elbow at the OTSG outlet nozzles were cut at a location of 22.5^{o}. The 22.5^{o} section of the elbow that was removed with the OTSG was replaced by an equivalent elbow integrally forged with the ROTSG outlet nozzle. The material for the replacement elbow is SA-508 Cl. 3a.

5.4.4 Reactor Coolant Pump Motors

The reactor coolant pump motors are large, vertical, squirrel cage, induction machines. The motors have flywheels to increase the rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. An anti-reverse device is included in the flywheel assembly to eliminate reverse rotation when there is back flow. Prevention of back rotation also reduces motor starting time.

The motors are enclosed with water-to-air heat exchangers so as to provide a closed circuit air flow through the motor. Radial bearings are floating pad type, and the thrust bearing is a double-acting Kingsbury type designed to carry the full thrust of the pump. A High Pressure Oil System with separate pumps is provided with each motor to jack and float the rotating assembly before starting. Once started, the motor provides its own oil circulation.

Instrumentation is provided to monitor motor cooling, bearing temperature, winding temperature, winding differential current, and speed. Instrumentation is also provided for measuring shaft displacement and frame velocity vibration.

In evaluating the design of the reactor coolant pump motor as it relates to the safety of the Reactor Coolant System, many items have been considered, namely: the overspeed of the motor; flywheel and shaft integrity; bearing design and system monitoring; seismic effects; and quality control and documentation.

An analysis of these considerations are given as follows as an indication of the safety and reliability that is integral with the motors:

5.4.4.1 Overspeed Considerations

The reactor coolant pump motors normally receive their electrical power from the nuclear generating unit through the unit's Auxiliary Electric System. On load rejection, the generating unit is designed to separate from the transmission network and remain in a standby operating condition carrying its own auxiliaries.

Figure 5-27 shows the turbine speed response following load rejection with the steam control valves wide open (VWO). On load rejection with VWO, the speed of the turbine-generator will increase under the control of the Normal Speed Governing Control System. The maximum speed attainable under the Normal Speed Governing Control System is less than 106 percent with the unit auxiliaries connected. This governing system is comprised of three independent control activities, namely: the speed control unit, power unbalance relay and the fast acting intercept valves all of which function to limit overspeed to below 106 percent.

As indicated in Figure 5-27 there are additional safety devices backing up the speed governing system, namely:

1. Mechanical overspeed trip which operates at 110 percent turbine-generator speed.
2. Generator overfrequency relay trip which is an electrical trip that operates at 111 percent turbine-generator speed.
3. Electrical back-up overspeed relay trip which operates at 112 percent turbine-generator speed.

In addition, each individual reactor coolant pump motor control circuit includes an overfrequency relay which trips the motor at 115 percent motor (or turbine-generator) speed. Therefore, it is evident that the reactor coolant pump motors speed will be limited to less than 115 percent.

5.4.4.2 Flywheel Design Consideration

For conservatism, the design of the flywheel on the reactor coolant pump motor is based on a design speed of 125 percent. The primary stress at the flywheel bore radius, with a speed of 125 percent, is 20,000 psi which is less than 50 percent of the 50,000 psi minimum yield strength of the flywheel material. This, therefore, yields a centrifugal stress design safety margin of 250 percent at 125 percent speed.

The Duke Power Company specification on the motor calls for 500 motor starts in forty years; the flywheels have been designed for 10,000 starts yielding a safety factor of 20. However, calculation based on the material used in the flywheel results in 400,000 cycles required for crack initiation which results in a flywheel fatigue design safety factor of 800.

The reactor coolant pump motors are large, vertical, squirrel cage, induction motors. The motors have flywheels to increase rotational-inertia, thus prolonging pump coastdown and

assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. The assumed operation of the reactor coolant pumps was 500 motor starts over forty years. The aging effect of concern is fatigue crack initiation in the flywheel bore key way from stresses due to starting the motor. Therefore, this topic is considered to be a time-limited aging analysis for license renewal.

The flywheels have been designed for 10,000 starts that provide a safety factor of 20 over the original operation assumptions. Reaching 10,000 starts in 60 years would require on average a pump start every 2.1 days. This conservative design is valid for the period of extended operation.

References for this section: Application [Reference 5] and Final SER [Reference 6]

5.4.4.3 Flywheel Material, Fabrication, Test and Inspection

5.4.4.3.1 Material

The flywheel is manufactured from vacuum degassed ASTM 533 steel.

5.4.4.3.2 Fabrication and Test

1. Flywheel blanks are flame cut from a plate with enough surplus material to allow for the removal of the flame affected metal.
2. At least three charpy tests are made on each plate parallel and normal to the rolling direction to determine that the blank meets specifications.
3. A complete 100 percent volumetric ultrasonic test is made on the blank and tension and bend tests are also made prior to shipment of a blank to Westinghouse Electric Company.
4. Following the machining of the flywheel at the Westinghouse plant, a complete 100 percent volumetric ultrasonic test is conducted on the fly wheel and a liquid penetrant test is conducted on the bore.
5. After the flywheel is installed and the motor is completely assembled, a 125 percent overspeed test for one minute is conducted on the assembled unit.
6. Following the overspeed test, a periphery sonic test is conducted on the flywheel through access holes in the motor frame.
7. To assure the original integrity of each flywheel during operation, the following inservice inspections will be performed.

   At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subject to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspections is greater than 6 2/3 years. Results of the examination will be evaluated by the original acceptance criteria and compared with the original examination data to assure the absence of unacceptable defects.

5.4.4.4 Shaft Design and Integrity

The shear stress on the shaft in the vicinity of the flywheel is 5520 psi with short circuit torque on the motor. The minimum strength of the shaft material is 23,000 psi which results in a safety

factor of four under the maximum torque condition. Because of the conservatism used in the design of the shaft, it is concluded that shaft failure is not credible.

5.4.4.5 Bearing Design and Failure Analysis

The motor pump assembly is supported by a Kingsbury type thrust bearing which consists of a runner and upper and lower thrust plates. The history of the Kingsbury type bearing design indicates that the device is highly reliable and has a non-locking failure mode.

Provided on the motor are a number of devices to warn the operator of bearing trouble and these devices are each independent in their operation. The thrust bearing monitoring devices are as follows:

1. Two thermocouples located diametrically opposite to each other in the upper thrust plates.
2. Two thermocouples located diametrically opposite to each other in the lower thrust plates.
3. One thermocouple in the upper oil pot.
4. Oil pot level alarm device.
5. Shaft displacement and frame velocity vibration devices.

These devices are arranged to provide alarm indications to the control room operator. If a thrust bearing fails with the motor operating, the result would be melting of the bearing babbit and, finally automatic tripout of the motor on overload. However, bearing degradation which would lead to this point would be evident to the control room operators in at least one of the indicators discussed above and would be mitigated by manually securing the pump. Therefore, since seizure of the bearing will not result from a bearing failure, it is concluded that missiles will not be produced.

5.4.4.6 Seismic Effects

The pump motor units have been analyzed against the combination effects of mechanical and seismic loads including the gyroscopic effects of the flywheel to verify that the stress limits will not be exceeded and the pump motor unit will operate through the maximum hypothetical earthquake.

5.4.4.7 Documentation and Quality Assurance

The Duke Power Company and the motor supplier, Westinghouse Electric Corp., have a rigid quality assurance program directed at assuring the integrity of the reactor coolant pump motors.

A quality assurance folder was initially developed by Duke Power Company on each motor and included the following:

1. Specifications and addendum
2. Description of the manufacturer's quality control organization and engineering order handling.
3. Copies of all inspection reports relating to the appropriate motor.
4. Samples of quality control drawings.
5. Copies of all test reports including flywheel material vendor test reports; Westinghouse motor test reports; bearing assembly reports; shaft tests; sonic test reports on the machined flywheel prior to assembly on the motor and following the 125 percent speed test; and

certification on the motor test report that the overspeed test was conducted on the assembled motor.

6. Copies of the Duke Form QA-2 which is the manufacturer's certification to Duke Power Company Design Engineering that the motors were manufactured per specification and the Duke Power quality assurance program.
7. Copies of Duke Form QA-1 which the indication to the field quality control engineer that the motor described thereon was manufactured to the specification and the Duke quality assurance program.
8. Copies of Duke Form QC-31 which is the field receiving report on the motor.

   A copy of each quality assurance folder was sent to the field quality control engineer and a copy was placed in the Design Engineering Department file. See the applicable controlled Procurement Package for RCP motor QA information.

Babcock & Wilcox has analyzed the reactor coolant pump assembly action resulting from postulated Reactor Coolant System breaks. B&W Topical Report, BAW-10040, Reference 1, describes the homologous pump model used for the speed calculations and presents results for the spectrum of breaks analyzed.

A discussion of the linear elastic fracture analysis to determine the structural failure speed of the reactor coolant pump motor flywheel assembly is also included.

5.4.5 Reactor Coolant Equipment Insulation

The majority of the Reactor Coolant System components are insulated with metal reflective type insulation. This insulation is supported by rings welded to weld pads on the components during field installation of the insulation. The weld pads to which the holding rings are attached are added to the components prior to final stress relief of the component. The replacement OTSGs do not have insulation support rings welded to the OTSGs instead, the support rings are friction supported. The remaining portion of the RCS is insulated with approved removable blanket insulation, secured with velcro fasteners.

The insulation units are removable and are designed for ease of removal and installation in such areas as field welds, nozzles, and bolted closures. The insulation units permit free drainage of any condensate or moisture from within the insulation unit.

5.4.6 Pressurizer

The pressurizer general arrangement is shown in Figure 5-28 and principal design data are tabulated in Table 5-22.

The electrically heated pressurizer establishes and maintains the Reactor Coolant System pressure within prescribed limits, and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant piping at the reactor outlet. The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section. Heat is removed or added to maintain Reactor Coolant System pressure within desired limits. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line and by an internal diffuser located above the surge pipe entrance to the vessel.

During outsurges, as Reactor Coolant System pressure decreases, some of the pressurizer water flashes to steam, thus assisting in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure. During insurges, as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by the pressure controller. The pressurizer water level is controlled by the level controller.

Since all sources of heat in the system, core, pressurizer heaters, and reactor coolant pumps, are interconnected by the reactor coolant piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one electromatic relief valve.

To eliminate abnormal buildup or dilution of boric acid within the pressurizer, and to minimize cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates a minimum of one gpm of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for monitoring boric acid concentration. A steam space sampling line provides capability for monitoring of or venting accumulated gases.

During cooldown and after the decay heat system is placed in service, the pressurizer can be depressurized and cooled by circulating through a connection from the High Pressure Injection System to the pressurizer spray line.

Electroslag welding is utilized in the fabrication of the pressurizer, only in the longitudinal seams of the shell courses. A total of three individual electroslag welds are made in the fabrication of each pressurizer. The techniques used in the electroslag welding are identical to those used in the electroslag welding program reported as Appendix F of Dockets No. 50-237 and 50-249 (Dresden Units 2 and 3). The procedures used were appropriately modified to reflect the difference in materials of the components being welded.

Each weld is subjected to radiographic inspection, ultrasonic inspection, and the finished surfaces of the weld are magnafluxed. In addition, each plate is ordered with excess width so that test specimens may be removed after heat treatment. Physical property test specimens including tensile and impact specimens of the base material heat affected zone and weld metal is obtained from this excess material in accordance with Section III of the ASME Code. Radiographic, ultrasonic, and magnetic particle inspection is preformed in accordance with Section III of the ASME Code and as required by Code Case 1355 which permits such welds for Class A vessels.

Physical tests are performed per Section N-511 of Section III of the ASME Code. For example:

1. All weld metal tensile specimens from each heat of weld wire, batch of flux, and for each combination of heat of wire and batch of flux used is obtained and tested after heat treatment.
2. Charpy impact test specimens representing weld metal and heat affected base material for every heat of wire, batch of flux, and combination of heat of wire and batch of flux used is tested.
3. Charpy V-notch impact specimens and tensile specimens are tested for 15 percent of all production welds. Included in this 15 percent are the tests required by 1 and 2 above.

Two men, one on the inside and one on the outside of the vessel, are used to check the progress of the weld, and to insure that the prescribed welding procedure is being followed. The weld is started in a U-shaped starting fixture about six inches deep attached to the bottom

of the joint. The weld stabilizes in this starting tab which is later cut off and discarded. The weld once started is not stopped until the total seam is completed.

The weld receives a heat treatment which consists of a water quench for 1625°F, and a temper of 1150°F, followed by an air cool. This post-weld heat treatment refines the grain of the weld and the base material heat affected zone such that it is virtually indistinguishable from the unaffected base material. The microstructure is the same through the weld.

Normal Reactor Coolant System pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray, electromatic relief valve, and heaters. The system is protected against overpressure by Reactor Protective System circuits such as the high pressure trip and by pressurizer relief valves located on the top head of the pressurizer. The schematic arrangement of the relief valves is shown in Figure 5-1 and Figure 5-2. Reactor Coolant System pressure settings and relief valve capacities are listed in Table 5-1.

Reduction of pressure during Reactor Coolant System cooldown is accomplished by the pressurizer spray provided by the reactor coolant pump. Below a system temperature of approximately 250°F, the Low Pressure Injection System is used for system heat removal and the steam generators and reactor coolant pumps are removed from service. During this period, spray flow is provided by a branch line from one high pressure injection line to the pressurizer spray line for further pressure reduction or complete depressurization of the Reactor Coolant System.

5.4.6.1 Pressurizer Spray

The pressurizer spray line originates at the discharge of a reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by a solenoid valve using an on-off control in response to the opening and closing pressure set points. An electric motor operated valve in series with the spray line is to provide for remote spray line isolation.

5.4.6.2 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during Reactor Coolant System heatup from the cooled down condition, and restore system pressure following transients. The heaters are arranged into four banks, which are then divided into eleven groups. The heaters are controlled by the pressure controller. The first bank utilizes proportional control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point. On-off control is used for the remaining three banks. A low level interlock prevents the heaters from being energized with the heaters uncovered.

The total pressurizer ambient heat loss is dependent in part on the insulation losses, which can vary due to tightness of fit and condition. Any pressurizer steam space leakage can also remove energy from the pressurizer. Pressurizer heater input may also decrease over the course of an operating cycle due to tripped breakers or burnt elements. A minimum required heater capacity capable of being powered from an emergency power source is necessary to offset these losses and ensure that RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling cannot be maintained indefinitely. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat.

The pressurizer heaters for each unit are supplied from non-safety-related motor control centers (MCC) with the exception of SSF group B and C pressurizer heaters. The Group B and C heaters are supplied from safety related MCCs. The non-safety related MCCs as well as the SSF group B heaters are in turn powered via load centers from the 4160-volt engineered safeguard buses. These buses are powered from a hydro station which is the emergency generation source (EGS) in the event of loss of offsite power. This emergency source has ample capacity to provide emergency power to all pressurizer heaters and is capable of doing so promptly following an accident. The pressurizer heaters are divided among the three 4160 volt EGS buses such that the loss of one entire 4160 volt bus will not preclude the capability to supply sufficient pressurizer heaters to maintain natural circulation in MODE 3 with average Reactor Coolant temperature ≥525°F.

SSF, Bank 2, Group C heaters for all three Units are powered via SSF Switchgear OTS1 which is normally powered from Unit 2 B2T Compartment 4. Although Unit 2 B2T is a safety related MCC, Compartment 4 is load shed during some scenarios in which the EGS supplies power. During these scenarios, the Group C heaters for all units would be unavailable. The loss of Group C of heater capacity per unit does not reduce overall heater capacity below what is adequate for the pressurizer heaters to perform their design function.

Uncovering energized direct immersion heaters does not immediately harm the heaters. Three original heaters, one for each bundle assembly, were tested in air to provide an accelerated life test as follows:

1. Tested for 100 hours at sheath temperature of 600°F to 1600°F with a watt density of 85 watt/in.^{2}.
2. Cycled 400 times with a cycle time of 15 minutes on and 15 minutes off with a watt density of 65 watt/in.^{2}.

The heaters successfully completed this test, which simulated a total of 200 hours “on” time for the heaters in an uncovered environment while in an energized condition. Moreover, the heater sheath is designed for 2500 psig and 670°F with the heater terminal also designed for these same conditions. Therefore, the heater sheath could fail and the pressurizer vessel integrity would be maintained. This conclusion has been substantiated in tests conducted by the heater vendor for a similar design.

The original heater bundles in Unit 1 were replaced using heaters of a new design. A new heater was selected at random from the production lot and tested in air with tests representative of the original heater tests-in-air. The new design heater successfully passed both tests.

5.4.6.3 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet is flanged to facilitate removal for maintenance or set point testing.

5.4.6.3.1 Safety Valve Testing and Qualification

During the EPRI Safety Valve testing, it was determined that the short inlet Dresser 31739A valve successfully met all the test requirements with the “reference” ring settings. The performance of the valve was determined to be dependent on the ring settings. Duke Power Company evaluated the safety impact of the inadequate safety ring settings and determined that for the limiting RCS overpressure transients the plant safety can be maintained. In October

1982, all the Oconee Nuclear Station safety valves were adjusted to the recommended settings resulting from EPRI tests. Duke Power Company has committed to the optimal ring settings for the Dresser 31739A safety valves, which are described in the corresponding NRC Safety Evaluation Report (Reference 4).

5.4.6.4 Pressurizer Electromatic Relief Valve

The pressurizer electromatic relief valve, also called power operated relief valve (PORV), is mounted on a separate nozzle on the top head of the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

The Power Operated Relief Valve (PORV) in each Oconee unit is actuated by a DC solenoid-operated pilot valve that is connected to a Class IE DC system. The block valve for the PORV is an AC motor operated valve and is connected to an AC emergency power supply. The power supplies for the PORV and its associated block valve are therefore independent and diverse.

5.4.6.4.1 PORV and Block Valve Testing and Qualification

Under the EPRI Test Program, for all tests applicable to Oconee, the Dresser PORV was opened and closed on demand. The functionability of the Dresser PORV has been shown for all expected operating and accident conditions applicable to Oconee Nuclear Station and the requirements of NUREG-0737, Item II.D.1.A have been met.

Under the EPRI PORV Block Valve Test Program, a Westinghouse motor-operated gate valve was tested on steam to full differential pressure conditions. Oconee Nuclear Station uses the same Westinghouse valve and LimiTorque operator for PORV block valve application. Based upon the successful EPRI tests for the valve-operator combination, the Oconee PORV block valves meet the intent of NUREG-0737 Item 11.D.1.B. The program test results were submitted to NRC in April 1982 and October 1985 (Reference 2).

With the initiation of NRC Generic Letter 89-10 (GL89-10) the Nuclear Industry and NRC have taken a much more focused, rigorous approach to assuring Active Motor Operated Valves (MOVs) have sufficient operating margin. The EPRI Performance Prediction Methodology (PPM) is a more conservative calculation guideline for determining MOV operating margins. Based on the EPRI PPM and other inputs, the operator for valves 1, 2, 3RC-4, the unit's PORV Block valves, have been upgraded from an SB-00-15 to an SB-0-25 operator.

5.4.6.5 Relief Valve Effluent

Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank which condenses and collects the relief valve effluent. After the quench tank receives relief valve effluent, the tank contents are cooled to normal temperature by the component drain pump and quench tank cooler of the Coolant Storage System. The tank fluid is circulated from the tank through the cooler and returned to the tank by spraying into the tank vapor space. The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank can be remotely vented to the Gaseous Waste Disposal System.

An Acoustical Monitoring System is installed on each unit. It is a reliable, single channel system, powered from a battery backed vital bus. It will provide the operator with positive

indication of valve position and an annunciation of an open valve in the control room. The valve position indication components have been seismically and environmentally qualified as appropriate for conditions applicable to their location.

Backup valve position indication is provided by temperature sensors located downstream of the PORV and safety valves and by the quench tank level indicator.

5.4.7 Interconnected Systems

5.4.7.1 Low Pressure Injection

The Low Pressure Injection System provides the capability below about 250°F for cooling the Reactor Coolant System during plant cooldown. During this mode of operation, coolant is drawn from the Reactor Coolant System through a nozzle on the reactor outlet pipe, circulated through the low pressure injection coolers by the low pressure injection pumps and then injected back into the Reactor Coolant System through two nozzles on the reactor vessel into the inlet side of the core. The heat received by this system is rejected to the Low Pressure Service Water System. Components in these two systems are redundant for reliability purposes.

The Low Pressure Injection System also performs an emergency injection function for a loss of coolant accident and provides long term emergency core cooling; this is described in Chapter 6.

5.4.7.2 High Pressure Injection

The High Pressure Injection System controls the Reactor Coolant System coolant inventory, provides the seal water for the reactor coolant pumps, and recirculates Reactor Coolant System letdown for water quality maintenance and reactor coolant boric acid concentration control. Letdown of reactor coolant is through a nozzle on the outlet coolant pipe from one steam generator. The discharge of the high pressure injection pumps connects to a nozzle on each of the reactor inlet pipes downstream of the reactor coolant pumps. The reactor coolant which is letdown is returned to the Reactor Coolant System through the nozzles in a different heat transport loop from the heat transport loop containing the letdown line. Components are redundant for reliability purposes (Section 9.3.2).

The High Pressure Injection System utilizes four injection nozzles in carrying out the high pressure emergency injection function after a loss of coolant accident.

The High Pressure Injection/Makeup (HPI/MU) Nozzle assemblies at Oconee incorporate a thermal sleeve to provide a thermal barrier between the cold HPI/MU Fluid and the HOT HPI Nozzle. In 1982, High Pressure Injection/Makeup Nozzle cracking problems were identified on several operating B&W plants. A task force formed by B&W owners group identified the root cause of the failures and undertook modifications, in consultations with NRC to eliminate such future failures.

Site inspections of Oconee 1, 2 and 3 were conducted. Oconee 2 and Oconee 3 were found to have nozzle cracking and thermal sleeve displacement problems. The radiographic and ultrasonic testing of Oconee 1 indicated that no abnormal conditions were present in any of the nozzles; this is attributable to the unique double thermal sleeve design of the Oconee 1 nozzles.

The B&W Owner's task force studied the safe end nozzle cracking problems on a generic basis and reported its findings to the NRC. The B&W Owner's task force developed a report that included its findings and recommendations to address the nozzle problems. Duke sent a letter to the NRC providing information that Duke supported the findings and recommendations provided in the B&W report (Reference 3). The task force concluded that all cracked safe ends

of the HPI/MU nozzles were associated with loose thermal sleeves; the cracked safe ends were associated with the makeup nozzles only, and the cracks were propagated by thermal fatigue. The B&W Owner's task force report provided recommendations regarding the HPI/MU nozzles. These recommendations included that inspections be made to the HPI/MU nozzles and that if the inspections indicated a gap existed or abnormal conditions were present, to perform recommended modifications to the design of the HPI/MU nozzles. The modified design installs a hard rolled thermal sleeve which prevents thermal shock to the nozzle assembly and helps reduce flow induced vibrations more effectively. An in-service inspection program had been developed to provide early detection of the safe-end cracking problems. The Oconee 1 makeup nozzles did not require modifications but are now subject to an augmented ISI program.

Augmented HPI thermal sleeve inspections (bore scope) have continued to identify cracking at the RCS end of the modified thermal sleeves. In an effort to eliminate all HPI thermal sleeve cracking, Oconee embarked on an HPI thermal sleeve redesign effort in 2002. A 2-ply HPI thermal sleeve design was developed and qualified. The new design was used to replace HPI/MU cracked nozzles on ONS 2 and ONS 3 in 2004. This redesign also included the replacement of the Alloy 600 weld between the stainless steel HPI nozzle safe end and the carbon steel RCS HPI nozzle with Alloy 690 weld material. This 2-ply design has an inner and outer thermal sleeve that significantly reduces through wall stress gradients and the potential for crack initiation and growth.

5.4.7.3 Core Flooding System

The Core Flooding System floods the core in the event of a loss of coolant accident. Connection to the reactor vessel is through the two nozzles described above for low pressure injection. The low pressure injection and core flooding lines tie together and connect to the same nozzle on the reactor vessel.

The core flood nozzles have flow restrictors installed to minimize blowdown due to postulated core flood line break.

5.4.7.4 Secondary System

The principal Decay Heat Removal System interconnected with the Reactor Coolant System is the Steam and Power Conversion System. The Reactor Coolant System is dependent upon the Steam and Power Conversion System for decay heat removal at normal operating conditions and for all reactor coolant operating temperatures above 250°F. The system is discussed in detail in Chapter 10.

The Turbine Bypass System routes steam to the condensers when the turbine has tripped or is shutdown and also during large plant load reduction transients when steam generation exceeds the demand. Overpressure protection for the secondary side of the steam generators is provided by the turbine bypass system and by safety valves mounted on the main steam lines outside of the Reactor Building. The Emergency Feedwater System will supply water to the steam generators in the event that the Main Feedwater System is inoperative. The physical layout of the Reactor Coolant System provides natural circulation of the reactor coolant to ensure adequate core cooling following a loss of all reactor coolant pumps.

5.4.7.5 Sampling

A sample line from the pressurizer steam space to the Chemical Addition and Sampling System permits detection of non-condensible gases in the steam space. This sample line also permits a bleeding operation from the vapor space to the letdown line of the High Pressure Injection

System to transport accumulated noncondensible gases in the pressurizer to the letdown storage tank.

5.4.7.6 Remote RCS Vent System

The Oconee design has the capability for venting post-accident non-condensible gases that, in sufficient quantities, could accumulate at high points in the RCS and impair natural circulation. Although such an event is highly unlikely, the remote RCS vents on the RCS hot legs and reactor vessels will enable venting of these gases. The reactor vessel head vents are also opened during RCS cooldowns conducted with natural circulation cooling to provide cooling flow through the upper head area and minimize steam void formation in that area. This venting was added to the emergency procedure guidelines in response to Generic Letter 81-21 (Reference 11), in order to simplify RCS pressure control during natural circulation cooldown.

The design of the RCS High Point Vent System consists of two valves installed in series in each of the following existing vent connections: steam generator piping high points, and reactor vessel head high point. The redundant valve in each vent line assures that venting operations can be terminated under postulated single failure. The three pairs of valves each receive electrical power from a different safety related power source. Vent valve position indication is provided by limit switches within each solenoid valve. The valves require power to open and fail close on loss of power. The existing power operated relief valve can be used to vent the pressurizer.

The reactor vessel head vent is attached to an existing Axial Power Shaping Rod motor tube and closure assembly. Two normally deenergized solenoid valves are installed in the vent line and controlled from the control room. The vent ties into a hot leg vent and discharges into the air stream from the Reactor Building Cooling Units when operated.

One independent remotely operated vent is provided at the high point of each 36-inch RCS hot leg line. Each vent makes use of the existing manual vent line. A tee has been added after the first manual valve and a new manual valve has been added after the tee. The first manual valve (1RC-19, 1RC-38, 2RC-19, 2RC-196, 2RC-38, 3RC-19, 3RC-38) is in the open position. The function of the first valve has been transferred to the second valve (1RC-168, 1RC-169, 2RC-168, 2RC-169, 3RC-168, 3RC-169). The new vent runs from the tee through two solenoid valves and discharges into the air stream from the Reactor Building Coolant Units. The solenoid valves are remotely controlled from the Control Room. The function of the manual vent is unaffected.

The reactor coolant vent system is acceptable to the NRC and in conformance with the requirements of 10CFR 50.44 paragraph (c)(3)(iii) and the guidelines of NUREG 0737 Item II.B.1, and NUREG-0800 Section 5.4.12.

5.4.8 Component Foundations and Supports

The supports for all major components listed in this section are analyzed in detail to insure adequate structural integrity for their intended function during normal operating, seismic, and accident conditions. Following calculation of sources of loading, stresses and motions at significant locations are computed and compared to applicable criteria. Details of this analysis are given in Chapter 3.

5.4.8.1 Reactor Vessel

The reactor vessel is bolted to a reinforced concrete foundation designed to support and position the vessel and to withstand the forces imposed on it by a combination of loads including

the weight of vessel and internals, thermal expansion of the piping, design basis earthquake (DBE), and dynamic load following reactor trip.

The foundation, in addition, is designed and built to restrain the vessel during the combined forces imposed by the circumferential rupture of a 36-inch reactor outlet line and a simultaneous maximum hypothetical earthquake (MHE). With the implementation of LBB, the foundation is no longer required to withstand the forces associated with the full rupture of a 36-inch reactor outlet line. However, the foundation has not been modified and this capability provides defense in depth.

The vessel foundation further is designed to provide accessibility for the installation and later inspection of incore instrumentation, piping, and nozzles; to contain ductwork and vent space for cooling air to remove heat losses from the vessel insulation; and to provide a sump and drainage line for leak detection.

5.4.8.2 Pressurizer

The pressurizer is supported on a structural steel foundation by eight lugs welded to the side of the vessel.

The foundation and supports are designed to withstand the loads imposed by the weight of the pressurizer including its contents and attached piping, relief valve reaction forces, and forces imposed by the design basis earthquake. In addition, the foundation and supports will restrain the vessel during the combined forces imposed by the circumferential rupture of the 10-inch surge line coupled with the MHE.

The foundation is also designed to permit accessibility to pressurizer surfaces for inspection. Oversized/slotted holes are provided to prevent development of stresses due to thermal expansion of the Pressurizer.

5.4.8.3 Steam Generator

The steam generator foundation is designed to support and position the generator. The foundation is designed to accept the loads imposed by the generators and feedwater piping filled with water, the attached reactor coolant piping also filled with water, and steam lines under the MHE. For the Replacement Steam Generators (RSG), with the implementation of Leak-Before-Break (LBB), the design of the RSG connection to the reactor building foundation no longer considers the rupture of the 28-inch reactor coolant lines. The applicable loading for the RSG foundation consists of deadweight, thermal, seismic, main steam pipe break (MSLB), and main feedwater pipe break (FWLB) load cases.

For the RSG, with the implementation of LBB, the design of the upper lateral support and the Lubrite bearing plates no longer considers the rupture of the 36-inch reactor coolant line. The applicable loading for the RSG upper lateral support and Lubrite bearing plates consists of deadweight, thermal, seismic, main steam pipe break (MSLB), and main feedwater pipe break (FWLB) load cases. Also, the revised seismic analysis for the reactor coolant system piping, with the RSG component, considers the upper lateral support to be an active seismic support.

5.4.8.4 Piping

The reactor coolant piping, inlet and outlet lines, are supported by the reactor vessel and steam generator nozzles. The piping will withstand the forces imposed on it by the MHE.

5.4.8.5 Pump and Motor

The reactor coolant pump casing, internals, and motor weight are supported by the 28 inch coolant lines and constant load hangers attached to the motor. In the cold condition, the coolant piping will support the coolant pump and motor without the hangers. The hangers are designed to withstand the forces imposed on them by the MHE.

5.4.8.6 LOCA Restraints

Each steam generator has a support located opposite the upper tube sheet and transfers forces from the generator into the shield walls in the event of a circumferential rupture of the 36-inch line.

Each 28-inch reactor coolant inlet line and 36-inch reactor coolant outlet line has a restraint located outside of and bolted to the primary shield to limit pipe motion in the event of a circumferential rupture of the piping inside the primary shield.

A detailed study of the primary loop was performed to determine potential pipe break locations which could possibly cause either fluid impingement or pipe impact forces on the Secondary System. The results of this evaluation indicated the most credible break locations which could cause either of these effects are:

1. A guillotine break at the pump discharge in the cold leg piping;
2. A longitudinal split in the vertical pump suction segment of the cold leg piping; or,
3. A longitudinal split in the vertical segment of the hot leg piping.

All of the above breaks could potentially affect the generator because of their proximity to it. The main steam lines, however, are shielded from the effects of pipe breaks by the generator.

The primary piping and steam generator were analyzed for each of the above breaks and supports provided to restrain the pipe from whipping into the generator. In addition, the stresses in the generator shell due to the fluid impingement forces were calculated and found to be within acceptable limits.

The restraints on the primary loop are shown in Figure 5-29 and Figure 5-30. The coolant pump is restrained by steel supports from the primary shield wall. The hot leg piping is restrained by the concrete support at the primary cavity penetration, an intermediate steel support from the primary wall, and another steel support near the generator upper tube sheet. The vertical segment of the cold leg piping is restrained by a steel support midway along its length, which would spread any rupture load over a larger area of the generator shell.

The original design of the Oconee Reactor Coolant System included LOCA restraints on the RCS hot leg and cold leg piping and on the Reactor Coolant Pumps (RCPs). Their original design function was to limit the hot and cold leg RCS piping movement in the event of a guillotine break of the RCS piping.

The B&W Owners Group Topical Report BAW-1847, Revision 1 (September 1985, Reference 7), demonstrated, with a fracture mechanics evaluation of the RCS piping, that such postulated RCS piping breaks had an extremely low probability of occurrence. This fracture mechanics evaluation of the RCS piping is known as Leak-Before-Break (LBB, see Section 5.2.1.9).

The NRC approved the B&W Owners Group Topical Report in a Safety Evaluation Report dated December 12, 1985 (Reference 8). This SER and the subsequent February 18, 1986 letter to Duke (Reference 9) provided the NRC's authorization for the implementation of LBB for the Oconee Reactor Coolant System. With the implementation of LBB, the RCS piping large break

LOCA's are no longer required to be postulated for the dynamic effects on the RCS piping and components, thus eliminating the need for the RCS piping and RCP LOCA restraints.

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As shown in Figure 5-29 and Figure 5-30, some of the original LOCA restraints were retained after steam generator replacement. Each steam generator has a support located opposite the upper tube sheet and transfers forces from the steam generator into the shield walls in the event of a circumferential rupture of the 36-inch line.

Each 28-inch reactor coolant pump inlet line and 36-inch reactor coolant pump outlet line has a restraint located outside of and bolted to the primary shield to limit pipe motion in the event of a circumferential rupture of the piping inside the primary shield. The hot leg restraints were partially deleted during steam generator replacement.

The original restraints were installed based on a detailed study of the primary loop that was performed to determine potential pipe break locations which could possibly cause either fluid impingement or pipe impact forces on the Secondary System. The results of this evaluation indicated the most credible break locations which could cause either of these effects are:

1. A guillotine break at the coolant pump discharge in the cold leg piping;
2. A longitudinal split in the vertical reactor coolant pump suction segment of the cold leg piping; or,
3. A longitudinal split in the vertical segment of the hot leg piping.

All of the above breaks could potentially affect the steam generator because of their proximity to it. The main steam lines, however, are shielded from the effects of pipe breaks by the steam generator. The primary piping and steam generator were analyzed for each of the above breaks and supports provided to restrain the pipe from whipping into the steam generator. In addition, the stresses in the steam generator shell due to the fluid impingement forces were calculated and found to be within acceptable limits.

The restraints on the primary loop are shown in Figure 5-29 and Figure 5-30. The reactor coolant pump was restrained by steel supports from the primary shield wall. The hot leg piping is restrained by the concrete support at the primary cavity penetration (which was partially deleted during steam generator replacement), an intermediate steel support from the primary wall, and another steel support near the steam generator upper tube sheet. The vertical segment of the cold leg piping was restrained by a steel support midway along its length, which would spread any rupture load over a larger area of the steam generator shell.

To verify the location and size of the piping supports, the piping was analyzed for rupture loads occurring at the worst point along its length. The rupture thrust force was assumed equal to P x A, where P is the coolant pressure and A the flow-sectional area of the pipe. The thrust was applied as an equivalent static force using a dynamic load factor of 2.0. Assuming the force to be a point load acting at the midpoint of the span between supports, the piping stresses were calculated using beam models. The supports were located so as to prevent the formation of plastic hinges in the piping, which would lead to an unstable linkage-type structure and possible impacting against the generator.

To evaluate the effect of fluid jet impingement on the generator, an equivalent static pressure load on the shell was calculated. A break of 14 ft^2 for the hot leg or 8.5 ft^2 for the cold leg was

assumed. The maximum initial mass velocity was computed using the methods outlined in the report "Maximum Two-Phase Vessel Blowdown From Pipes, APED-4827," by F. J. Moody. It was assumed that the fluid leaves the break in a direction normal to the pipe and that its velocity undergoes a 90° change in direction upon impinging on the steam generator. The resulting shell pressure loading was calculated to be 1300 psi.

A shell analysis was performed on the steam generator to determine the stress intensity due to the above loading. A B&W proprietary digital computer code, which considers two-dimensional shells with asymmetric loading, was utilized. The loading distribution and stress model are shown in Figure 5-31 and Figure 5-32. The maximum stress intensity was computed to be 38,600 psi. This is less than the allowable stress of 46,670 psi. Based on these results for the 36- inch i.d. pipe break, it was concluded that the steam generator shell could also withstand the reduced loading which would be generated by a 28-inch i.d. break.

5.4.8.6.1 Replacement Steam Generator LOCA Analysis

For the replacement steam generator RCS structural analysis, there are ten high energy line breaks considered.

1. Single Main Steam Line Break
2. Double Main Steam Line Break
3. Single Main Feedwater Line Break
4. Double Main Feedwater Line Break
5. Surge Line Break at the Hot Leg Nozzle
6. Surge Line Break at the Pressurizer Nozzle
7. Surge Line Break at the Intermediate Surge Line Drain, North Direction Thrust
8. Surge Line Break at the intermediate Surge Line, East Direction Thrust
9. Decay Heat Line Break at the Hot Leg Nozzle
10. Core Flood Line Break

For each of the ten breaks the replacement steam generator and primary piping whip restraints are considered inactive because the component displacements are not large enough to cause a contact between the restraint and the component. The restraints on the reactor coolant pumps were not included in the analysis because they are to be removed during the steam generator replacement.

Each of the above high energy line breaks were analyzed using the proprietary Framatome ANP computer program BWSPAN to calculate the loads incurred through out the reactor coolant system due to the effects of jet impingement and asymmetric cavity pressure. All of the reactor coolant system piping, components and supports have been shown to be acceptable for the loading applied by each of the above high energy line breaks.

5.4.9 References

1. BAW-10040, Reactor Coolant Pump Assembly Overspeed Analysis
2. Letter from H. B. Tucker (Duke) to H. R. Denton (NRC) dated October 1, 1985. Subject: Performance Testing of Relief and Safety Valves.

3. Babcock & Wilcox Owners Group Safe End Task Force Report on Generic HPI/MU Nozzle Component Cracking. B&W Document #77-1140611-00, submitted by Duke to the NRC in a letter dated February 15, 1983.
4. Letter from L. A. Weins (NRC) to H. B. Tucker (Duke) dated July 19, 1989. Subject: Safety Evaluation Report for NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves for Oconee Units 1, 2, and 3 (TACS 44600, 44601, and 44602).
5. *Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3*, submitted by M. S. Tuckman (Duke) letter dated July 6, 1998 to Document Control Desk (NRC), Docket Nos. 50-269, - 270, and -287.
6. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3*, Docket Nos. 50-269, 50-270, and 50-287.
7. B&W Owners Group Topical Report “The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS”, BAW-1847, Revision 1, September 1985.
8. Letter from Dennis M Crutchfield (NRC) to L. C. Oakes (B&W Owner Group) dated December 12, 1985, Subject: “Safety Evaluation of B&W Owners Group Report with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops”
9. Letter from John F. Stoltz (NRC) to H. B. Tucker (Duke) dated February 18, 1986, Subject: “Safety Evaluation of B&W Owners Group Report with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops”
10. Letter from Leonard A. Wiens (NRC) to J. W. Hampton (Duke) dated March 24, 1995, Subject: "Evaluation of the Oconee, Units 1, 2, and 3 Generic Safety Issues (GSI) Resolution" (GSI-23, GSI-105, and GSI-153)
11. H. B. Tucker (Duke) Letter to H. R. Denton (NRC) dated December 12, 1984, Subject: “DPC Intent Regarding Natural Circulation Cooldown Per Generic Letter 81-21”

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