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5.0 Reactor Coolant System

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5.1 Summary Description

The Reactor Coolant System shown in Figure 5-1 consists of similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator and associated piping and valves. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping and instrumentation necessary for operational control. All major components are located in the Containment building.

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

The Reactor Coolant System pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the unit.

Reactor Coolant System pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters or water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

The extent of the Reactor Coolant System is defined as:

1. The reactor vessel including control rod drive mechanism housings.
2. The reactor coolant side of the steam generators.
3. Reactor coolant pumps.
4. A pressurizer attached to one of the reactor coolant loops.
5. Safety and relief valves.
6. The interconnecting piping, valves and fittings between the principal components listed above.
7. The piping, fittings and valves leading to connecting auxiliary or support systems up to and including the second isolation valve (from the high pressure side) on each line.

REACTOR COOLANT SYSTEM COMPONENTS

Reactor Vessel

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core supporting structures, control rods and other parts directly associated with the core. The upper (closure) head contains 82 penetrations (78 for control rod mechanisms and instrumentation devices and 4 auxiliary adapters).

The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

Steam Generators

The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the

nozzles located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

Reactor Coolant Pumps

The reactor coolant pumps are identical single-speed centrifugal units driven by water-to-air-cooled, three-phase induction motors. The shaft is vertical with the motor mounted above the pumps. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The inlet is at the bottom of the pump; discharge is on the side.

Piping

The Reactor Coolant System loop piping is specified in sizes consistent with system requirements.

The hot leg inside diameter is 29 inches and the cold leg return line to the reactor vessel is 27-1/2 inches. The piping between the steam generator and the pump suction is increased to 31 inches in diameter to reduce pressure drop and improve flow conditions to the pump suction.

Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads that is connected to the Reactor Coolant System on one of the hot legs of a coolant loop. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief and safety valve connections are located in the top head of the vessel.

Pressurizer Relief Tank

The pressurizer relief tank is a horizontal, cylindrical vessel with elliptical ends. Capacity of the pressurizer relief tank is indicated in Table 5-1. Steam from the pressurizer safety and relief valves is discharged into the pressurizer relief tank through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature.

Safety and Relief Valves

The pressurizer safety valves are of the totally enclosed pop-type. The valves are spring-loaded, self-activated with back-pressure compensation. The power-operated relief valves limit system pressure for large power mismatch. They are operated automatically or by remote manual control. Remotely operated valves are provided to isolate the air-operated valves for repair if excessive leakage occurs. Position indicator lights are provided in the Control Room for these valves.

REACTOR COOLANT SYSTEM PERFORMANCE CHARACTERISTICS

The following paragraphs describe how the NSS supplier (Westinghouse) calculated the reactor coolant design flow. Actual reactor coolant flow is verified by the elbow tap method semi-daily and by the calorimetric heat balance method each fuel cycle.

Tabulations of important design and performance characteristics of the Reactor Coolant System are provided in Table 5-1.

Reactor Coolant Flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established with a detailed design procedure supported by operating unit performance data, by pump model tests and analysis, and by pressure drop tests and analyses of the reactor vessel and fuel assemblies. Data from all operating units have indicated that the actual flow has been well above the flow specified for the thermal design of the unit. By applying the design procedure described below, it is possible to specify the expected operating flow with reasonable accuracy.

Three reactor coolant flow rates are identified for the various unit design considerations. The definitions of these flows are presented in the following paragraphs, and the application of the definitions is illustrated by the system and pump hydraulic characteristics on Figure 5-3.

Best Estimate Flow

The best estimate flow is the most likely value for the actual unit operating condition. This flow is based on the best estimate of the reactor vessel, steam generator and piping flow resistance, and on the best estimate of the reactor coolant pump head, with no uncertainties assigned to either the system flow resistance or the pump head. System pressure losses based on best estimate flow are presented in Table 5-1. Although the best estimate flow is the most likely value to be expected in operation, more conservative flow rates are applied in the thermal and mechanical designs.

Thermal Design Flow

Thermal design flow is the basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal unit parameters used throughout the design. To provide the required margin, the thermal design flow accounts for the uncertainties in reactor vessel, steam generator and piping flow resistances. The combination of these uncertainties, which includes a conservative estimate of the pump discharge weir flow resistance, is equivalent to increasing the best estimate Reactor Coolant System flow resistance by approximately 19 percent. The intersection of this conservative flow resistance with the best estimate pump curve, as shown in Figure 5-3 establishes the thermal design flow. This procedure provides a flow margin for thermal design of approximately 4.7 percent. The thermal design flow is confirmed when the unit is placed in operation. Tabulations of important design parameters based on the thermal design flow are provided in Table 5-1.

Mechanical Design Flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. To assure that a conservatively high flow is specified, the mechanical design flow is based on a reduced system resistance (90 percent of best estimate) and on the maximum uncertainty on pump head capability (107 percent of best estimate prior to machining pump impellers). The intersection of this flow resistance with the higher pump curve, as shown on Figure 5-3, establishes the mechanical design flow. The resulting flow is approximately 4.5 percent greater than the best estimate flow.

Pump overspeed, due to a turbine generator overspeed of 20 percent, results in a peak reactor coolant flow of 120 percent of the mechanical design flow. The overspeed condition is applicable only to operating conditions when the reactor and turbine generator are at power.

INTERRELATED PERFORMANCE AND SAFETY FUNCTIONS

The interrelated performance and safety functions of the Reactor Coolant System and its major components are listed below:

1. The Reactor Coolant System provides sufficient heat transfer capability to transfer the heat produced during power operation and when the reactor is subcritical, including the initial phase of cooldown, to the steam and power conversion systems.
2. The system provides sufficient heat transfer capability to transfer the heat produced during the subsequent phase of cooldown and cold shutdown to the Residual Heat Removal System.
3. The system heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, shall assure no fuel damage within the operating bounds permitted by the Reactor Control and Protection Systems.
4. The Reactor Coolant System provides the water used as the core neutron moderator and reflector and as a solvent for chemical shim control.

5. The system maintains the homogeneity of soluble neutron poison concentration and rate of change of coolant temperature such that uncontrolled reactivity changes do not occur.
6. The reactor vessel is an integral part of the Reactor Coolant System pressure boundary and is capable of accommodating the temperatures and pressures associated with the operational transients. The reactor vessel functions to support the reactor core and control rod drive mechanisms.
7. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of unit load, reactor coolant volume changes are accommodated via the surge line to the pressurizer.
8. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.
9. The steam generator provides high quality steam to the turbine. The tube and tube sheet boundary are designed to prevent the transfer of activity generated within the core to the secondary system.
10. The Reactor Coolant System piping serves as a boundary for containing the coolant under operating temperature and pressure conditions and for limiting leakage (and activity release) to the Containment atmosphere. The Reactor Coolant System piping contains demineralized light water which is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

Interlocks on critical motor-operated valves are discussed in Chapter 6.

SYSTEM OPERATION

Brief descriptions of normal anticipated system operations are provided below. These descriptions cover unit startup, power generation, hot standby, hot shutdown, cold shutdown and refueling.

Unit Startup

Unit startup encompasses the operations which bring the reactor from cold shutdown to 15 percent full power. Before unit startup, the reactor coolant loops and pressurizer are filled completely by the use of the charging pumps, with water containing the cold shutdown concentration of boron. The secondary side of the steam generator is filled to normal startup level with water which meets the water chemistry requirements.

The Reactor Coolant System is then pressurized, by use of the letdown back pressure control valve and the centrifugal charging pumps, to obtain the required NPSH and a minimum 6 gpm seal injection flow rate. The pumps may then be operated intermittently to assist in venting operations.

During operation of the reactor coolant pumps, one charging pump and the low pressure letdown path from the Residual Heat Removal System loop to the Chemical and Volume Control System are used to maintain the Reactor Coolant System pressure. Operation of the reactor coolant pumps must only be initiated when the minimum NPSH requirements are met, and a minimum 6 gpm seal injection flow rate is confirmed. Fracture prevention temperature limitations of the reactor vessel impose an upper limit of approximately 450 psig. The charging pump supplies seal injection water for the reactor coolant pump shaft seals. A nitrogen atmosphere and normal operating temperature, pressure and water level are established in the pressurizer relief tank.

Upon completion of venting, the Reactor Coolant system is pressurized, all reactor coolant pumps are started and the pressurizer heaters are energized to begin heating the reactor coolant. When the pressurizer temperature exceeds approximately 450°F, a steam bubble is formed while the reactor coolant pressure is maintained at approximately 400 psig. The pressurizer liquid level is reduced until the no-load power level volume is established. During the initial heatup phase, hydrazine is added to the reactor coolant to scavenge the oxygen in the system; the heatup is not taken beyond 250°F until the oxygen level has been reduced to the specified level.

The reactor coolant pumps and pressurizer heaters are used to raise the reactor coolant temperature to a level beyond which the overall moderator temperature coefficient is negative.

As the reactor coolant temperature increases, the pressurizer heaters are manually controlled to maintain adequate suction pressure for the reactor coolant pumps. When the normal operating pressure of 2235 psig is reached, pressurizer heat and spray controls are transferred from manual to automatic control.

Power Generation and Hot Standby

Power generation includes steady-state operation, ramp changes not exceeding the rate of five percent of full power per minute, step changes of ten percent of full power (not exceeding full power), and step load changes with steam dump not exceeding the design step load decrease, between 15 percent full power and 100 percent full power.

During power generation, Reactor Coolant System pressure is maintained by the pressurizer controller at or near 2235 psig, while the pressurizer liquid level is controlled by the charging-letdown flow control of the Chemical and Volume Control System.

When the reactor power level is less than 15 percent, the reactor power is controlled manually. At power above 15 percent, the Reactor Control System controls automatically maintain an average coolant temperature, consistent with the power relationships, by control rod movement.

Two methods are employed to shutdown a unit from mode 1 (Power Operation) to mode 3 (Hot Standby). One method involves the manual insertion of control rods into the core as power is reduced to zero percent power. The second method relies on a manual reactor trip from less than 20 percent power.

During the hot standby operations, when the reactor is subcritical, the Reactor Coolant System temperature is maintained by steam dump to the main condenser. This is accomplished by a controller in the steam line, operating in the pressure control mode, which is set to maintain the steam generator steam pressure. Residual heat from the core or operation of a reactor coolant pump provides heat to overcome Reactor Coolant system heat losses.

Unit Shutdown

Unit shutdown is the operation which brings the reactor from 15 percent full power to cold shutdown. Before unit shutdown, concentrated boric acid solution from the Chemical and Volume Control System is added to the Reactor Coolant System to increase the reactor coolant boron concentration to that required for cold shutdown. The hydrogen and fission gas in the reactor coolant is reduced by degassing the coolant in the volume control tank. The pressurizer steam space is degassed through the sample system to the volume control tank.

Unit shutdown is accomplished in two phases. The first is by the combined use of the Reactor Coolant System and steam systems, and the second by the Residual Heat Removal System. During the first phase of shutdown, residual core and reactor coolant heat is transferred to the steam system via the steam generator. Steam from the steam generator is dumped to the main condenser. At least one reactor coolant pump is kept running to assure uniform Reactor Coolant System cooldown. The pressurizer heaters are de-energized and spray flow is manually controlled to cool the pressurizer while maintaining the required reactor coolant pump suction pressure. As the pressurizer cools to approximately 400°F during the second phase of shutdown, the steam bubble gradually collapses as the letdown back pressure control valve and the charging pumps maintain the pressure at approximately 325 psig.

When the reactor coolant temperature is below 350°F and the pressure is less than 450 psig, the second phase of shutdown commences with the operation of the Residual Heat Removal System.

One reactor coolant pump (either of those in a loop containing a pressurizer spray line) is kept running until the coolant temperature is less than 160°F. At this temperature, the reactor coolant pump is turned

off. Pressurizer cooldown is continued by initiating auxiliary spray flow from the Residual Heat Removal System. Unit shutdown continues until the reactor coolant temperature is 140°F or less.

Refueling

Before removing the reactor vessel head for refueling, the system temperature has been reduced to 140°F or less and hydrogen and fission product levels have been reduced. The Reactor Coolant system is then drained until the water level is below the reactor vessel flange. The vessel head is then raised and the refueling canal is flooded. Upon completion of refueling, the reactor vessel head is replaced and the system is refilled for startup.

5.1.1 Schematic Flow Diagram

The Reactor Coolant System is shown in Figure 5-1 principal pressures, temperatures, flow rates and coolant volume data under normal-steady state full power operating conditions are provided in Table 5-1.

5.1.2 Piping and Instrumentation Diagram

A piping and instrumentation diagram of the Reactor Coolant System is shown on Figure 5-1. The diagram shows the extent of the systems located within the Containment, and the points of separation between the Reactor Coolant System, and the secondary (heat utilization) system. The isolation provided between the Reactor Coolant Pressure Boundary and connected systems is discussed in 6.2.4.

5.1.3 Elevation Drawing

Elevation drawings providing principal dimensions of the Reactor Coolant System in relation to surrounding concrete structures are presented on Figure 1-15 and Figure 1-16.

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5.2 Integrity of the Reactor Coolant Pressure Boundary

In support of the McGuire Unit 1 and Unit 2 Measurement Uncertainty Recapture (MUR) power uprate, the reactor vessel material discussions and values presented in affected tables have been re-performed to reflect the uprated power operation level for the reactor core of 3469 MWt (1.02 times the original licensed power level (3411 MWt), minus measurement uncertainty (0.3%)). The updated reactor vessel materials discussions/tables re-performed at 3469 MWt bound the original licensed power operation for the reactor core at 3411 MWt.

5.2.1 Design of Reactor Coolant Pressure Boundary Components

The Reactor Coolant System boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of operation including all anticipated transients, and to maintain the stresses within applicable stress limits. The system is protected from overpressure by means of pressure relieving devices as required by applicable codes. Material construction are specified to minimize corrosion and erosion and to provide a structural system boundary throughout the life of the units. Fracture prevention measures are taken to prevent brittle fracture. Inspection in accordance with applicable codes and provisions are made for surveillance of critical areas to enable periodic assessment of the boundary integrity as described in Section 5.2.8.

Fatigue Evaluation for License Renewal:

McGuire Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 5.2.1 to assure that components are maintained within design limits. This requirement is managed by the McGuire Thermal Fatigue Management Program.

Thermal Fatigue Management Program:

The four key actions of the Thermal Fatigue Management Program are:

Determining the Thermal Cycles to be Monitored and Their Character and Number of Allowed Occurrences: The set of transients events to be managed by the Thermal Fatigue Management Program is derived from the associated component information. Included are their thermal and pressure profile characterisitics and the minimum of the numbers of occurrences used in the evaluations. As updates occur to associated component information such as analyzed conditions, operational practices, inservice inspection results, flaw growth analyses or, fatigue environmental effect modifications required for the extended period of operation (after 40 years), the set of transients and their limits may require revision.

Monitoring the Thermal Cycles Experienced: From continual monitoring of plant operating conditions, plant conditions that meet the definition of a transient cycle defined by this program are noted. Upon discovery of each transient cycle required to be documented by the program, the cycle count for that transient event is updated. For those events that are logged, the Thermal Fatigue Management Program specifies appropriate parameters such as minimum/maximum temperature limits and rates of temperature change that are assumed in the analysis. The logging process captures these values for review.

Comparison of Observed Events to Allowable Events: For the transients that have occurred since the previous assessment, two evaluations are performed to determine if parameters are within limits. The first evaluation compares the observed values for those parameters applicable to each transient to the limits described in the Thermal Fatigue Management Program (e.g. a maximum or minimum temperature limit). The second evaluation is a comparison to the allowable number of occurrences.

Corrective Action and Confirmation Process: Should the thermal and pressure profile for a specific transient be outside of the parameters defined for that transient set or should an allowable cycle count limit for a transient cycle set be approached or exceeded, this is identified to the appropriate engineering group(s) for resolution. The corrective action program is triggered immediately if profile values are exceeded. Similarly, the corrective action program is triggered if the number of events is expected to exceed the thermal fatigue basis limits within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay with acceptable cycle count limits.

Future Modification to the TFMP for Environmentally Assisted Fatigue:

The Thermal Fatigue Management Program will address the effects of the coolant environment on component fatigue life (environmentally assisted fatigue or EAF) by assessing the impact of the reactor coolant environment on a sample of critical locations selected from NUREG/CR-6260 and other locations expected to have high usage factors when considering environmentally assisted fatigue. The objective to meet in choosing locations will be to ensure by example that no plant location will have an EAF-adjusted CUF that exceeds 1.0 in actual operation.

The sample of critical components can be evaluated by applying the environmental correction factors to the existing ASME Code fatigue analyses and either (1) computing and tracking an EAF adjusted CUF against an allowable of 1.0 or (2) tracking the instances of transients identified in Paragraph 1.1 above against an EAF adjusted allowable number of transients.

Base formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NURE/CR-5704 for austenitic stainless steels. Duke recognizes these formulas as the current methodology for determining such factors.

The exercise of the above procedure will be at a time prior to the end of the 40th year of each unit's operation. This lead time shall be sufficient to ensure that implementation of corrective actions will prevent the exceedance of 1.0 of EAF-adjusted CUF within the extended period of operation. No requirement exists that any resulting adjustments in allowables be applied prior to the end of the initial 40 years of operation. It is recognized that a discontinuity exists at the 40 year point in the need to apply this adjustment.

Duke may chose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with Duke.

Leak-Before-Break Evaluation for License Renewal:

Leak-before-break analyses evaluate postulated flaw growth in the primary loop piping of the Reactor Coolant System. These analyses consider the thermal aging of the cast austenitic stainless steel material of the piping as well as the fatigue transients that drive the flaw growth over the operating life of the plant. Because all of the criteria contained in §54.3 are met, leak before break is a TLAA for McGuire. The leak before break analyses have been determined to be acceptable for the period of extended operation.

5.2.1.1 Performance Objectives

The performance objectives of the Reactor Coolant System for normal operation are described in Section 5.1. The performance objectives for upset and faulted conditions are given in Section 5.2.1 above. No transient is classified as emergency condition.

Equipment code and classification list for the components within the Reactor Coolant System boundary are given in Table 3-4.

The Reactor Coolant System, in conjunction with the Reactor Control and Protection Systems, is designed to maintain the reactor coolant at conditions of temperature, pressure and flow adequate to protect the core from damage. The design requirement for safety is to prevent conditions of high power, high reactor coolant temperature or low reactor coolant pressure or combinations of these which could result in a DNBR less than the analysis limit.

The Reactor Coolant System is designed to allow controlled changes in the boric acid concentration and the reactor coolant temperature. The reactor coolant is the core moderator, reflector, and solvent for the chemical shim. As a result, changes in coolant temperature or boric acid concentration affect the reactivity level in the core.

The following design basis have been selected to ensure that the uniform Reactor Coolant System boron concentration and temperature is maintained:

1. Coolant flow is provided by either a reactor coolant pump or a residual heat removal pump to ensure uniform mixing whenever the boron concentration is decreased.
2. The design arrangement of the Reactor Coolant System eliminates dead ended sections and other areas of low coolant flow in which nonhomogeneities in coolant temperature or boron concentration could develop.
3. The Reactor Coolant System is designed to operate within the operating parameters, particularly the coolant temperature change limitations.

5.2.1.2 Design Parameters

The design pressure for the Reactor Coolant System is 2485 psig except for the pressurizer relief lines from the safety valves to the pressurizer relief tank, which is 500 psig, and the pressurizer relief tank, which is 100 psig. For components with design pressures of 2485 psig the normal operating pressure is 2235 psig. The design temperature for the Reactor Coolant System is 650°F except for the pressurizer including the surge and relief lines which are designed to 680°F, piping downstream of the pressurizer safety valves and the power operated relief valves which are designed to 500°F, and the pressurizer relief tank which is designed to 340°F. The seismic loads for McGuire are given in Section 3.7 of the FSAR. Reactor Coolant System and component test pressures are discussed in Section 5.2.1.5.

5.2.1.3 Compliance with 10CFR 50, Section 50.55a

The Reactor Coolant System and its components are designed and fabricated in accordance with the rules of 10CFR 50, Section 50.55a, “Codes and Standards”. Applicable code addenda are shown in Table 5-7.

5.2.1.4 Applicable Code Cases

Westinghouse meets the intent of Regulatory Guides 1.84 and 1.85 by controlling its suppliers to:

1. Avoid the use of Code Cases unendorsed by the NRC except where specific authorization of the NRC is obtained, and
2. Disallow the use of new Code Cases until acceptability to the NRC is assured.

5.2.1.5 Design Transients

The design transients and the limiting allowable number of occurrences of each that were used for fatigue eval;uations or fracture mechanics evaluations are shown in Table 5-49. In accordance with the ASME Boiler and Pressure Vessel Code, faulted conditions are not included in fatigue evaluations. The loading combinations used in the design of reactor coolant boundary components are given in Table 5-3.

Section 5.2.1.5 of revision 1 of Regulatory Guide 1.70 states "provide a complete list of transients to be used in the design and fatigue analysis of all the applicable components within the reactor coolant pressure boundary discussed in Sections 5.4 and 5.5. Specify all design transients and their number of cycles such as startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, seismic events, etc., that are contained in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary."

As stated above, McGuire Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 5.2.1 to assure that components are maintained within design limits.

Table 5-49 meets the RG 1.70 requirement to provide a listing of transients for which the applicable components have been qualified, with the following exceptions:

1. For practicality purposes, local transient events that occur as sub events of listed events are not tabulated in Table 5-49, but are defined in the details of the event descriptions of listed events as found and maintained in the applicable plant Engineering Records.
2. As stated in Sections 5.2.1.1, 5.2.1.5, 5.2.1.14 and 5.2.1.15, emergency conditions are not part of the design basis for McGuire. Thus no emergency conditions extraneously specified in any ASME Design Specifications are listed in Table 5-49.
3. As given in Note 4 of Table 5-49, some components/piping segments are individually qualified for faulted events which are not listed because the Unit as a whole is therefore not qualified for such.
4. In order to provide a definition of the envelope against which operation is to be compared, in compliance with McGuire Technical Specification 5.5.6, the occurrence quantity provided in Table 5-49 is the limiting allowable number of occurrences predicted or analyzed for each transient event, for all components and piping segments, considering ASME Section III and XI limits. Providing a complete tabulation of the design quantities (as required by RG 1.70), which varies from component to component, would be impractical. Such design information is however found and maintained in the applicable plant Engineering Records.

The following five ASME operating conditions are considered in the design of the Reactor Coolant System.

1. Normal Conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

2. Upset Conditions

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the design specifications.

3. Emergency Conditions

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any

damage developed in the system. The total number of postulated occurrences for such events shall not cause more than twenty-five stress cycles that have a Sa value greater than that for 10^6 cycles from the applicable fatigue design curves of ASME Section III.

4. Faulted Conditions

   Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Testing Conditions

   Testing conditions are those tests in addition to the hydrostatic or pneumatic tests permitted by ASME Section III including leak tests or subsequent hydrostatic tests.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the unit. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during unit operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of actual transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the unit.

The following is a description of a selected subset of the primary system transients from Table 5-49 taken from RCS component Equipment Specifications written in accordance with the ASME Code.

The five following transients are considered normal conditions:

1. Heatup and Cooldown

   For design evaluation, the heatup and cooldown cases are represented by continuous heatup or cooldown at a rate of 100°F per hour (200°F per hour for cooldown of the Pressurizer) which corresponds to a heatup or cooldown rate under abnormal or emergency conditions. The heatup occurs from ambient to the no load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour is not usually attained because of other limitations such as:

   a. Criteria for prevention of non-ductile failure which establish maximum permissible temperature rates of change, as a function of pressure and temperature.

   b. Slower initial heatup rates when using pumping energy only.

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

2. Unit Loading and Unloading

   The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15 percent load and full load. This load swing is the maximum possible, consistent with operation with automatic reactor control. The reactor temperature varies with load as prescribed by the temperature control system.

3. Step Increase and Decrease of Ten Percent

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

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the Reactor Coolant System average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure also decreases from its peak pressure value and follows the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently the pressurizer heaters come on to restore the pressure to its normal value.

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

4. Large Step Decrease in Load

   This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature automatically initiates a secondary side steam dump system that prevents shutdown or lifting of the Main Steam System safety valves. Thus, when a unit is designed to accept a step decrease of 50 percent from full power, it signifies that a steam dump system provides a heat sink to accept 40 percent of the turbine load. The remaining 10 percent of the total step change is assumed by the Rod Control System. If a steam dump system were not provided to cope with this transient, there would be such a large mismatch between what the turbine is demanding and what the reactor is furnishing that a reactor trip and lifting of the Main Steam System safety valve would occur.

5. Steady State Fluctuations

   The reactor coolant average temperature, for purposes of design, is assumed to increase or decrease a maximum of 6°F in one minute. The temperature changes are assumed to be around the programmed value of T_{avg}, (T_{avg}±3°F). The corresponding reactor coolant average pressure is assumed to vary accordingly.

The following six transients are considered upset conditions:

1. Loss of Load Without Immediate Turbine or Reactor Trip

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

2. Loss of Power

   This transient applies to a blackout situation involving the loss of outside electrical power to the station with a reactor and turbine trip. Under these circumstances, the reactor coolant pumps are de-energized and following the coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater from the Auxiliary Feedwater System operating from diesel generator power. Steam is removed for reactor cooldown through Main Steam System atmospheric relief valves provided for this purpose.

3. Loss of Flow

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

4. Reactor Trip From Full Power (Nominal)

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

5. Inadvertent Pressurizer Auxiliary Spray Initiation

   The inadvertent pressurizer auxiliary spray transient occurs if the auxiliary spray valve is opened inadvertently during normal operation of the unit. This introduces cold water into the pressurizer with a very sharp pressure decrease as a result.

   The temperature of the auxiliary spray water is dependent upon the performance of the regenerative heat exchanger. The most conservative case is when the letdown flow is shut off and the charging fluid enters the pressurizer unheated. Therefore, for design purposes, the temperature of the spray water is assumed to be 100°F. The spray flow rate is assumed to be 200 gpm.

   The pressure decreases rapidly to the low pressure reactor trip point. At this pressure the pressurizer low pressure reactor trip is assumed to be actuated; this accentuates the pressure decrease until the pressure is finally limited to the hot leg saturation pressure. At five minutes the spray is stopped and all the pressurizer heaters return the pressure to 2250 psia. This transient is more severe on a two loop unit than on a four loop unit, e.g., a bigger and more rapid pressure decrease. Therefore, the transient for a two loop unit is used as design basis for McGuire.

For design purposes it is assumed that no temperature changes in the Reactor Coolant system occur as a result of initiation of auxiliary spray except in the pressurizer.

6. Operational Basis Earthquake

   The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transient loads resulting from fluid pressure and temperature. Their magnitude, however, is considered in the design analysis for comparison with appropriate stress limit.

No transient is classified as an emergency condition.

The four following transients are considered faulted conditions:

1. Reactor Coolant System Boundary Pipe Break

   This accident involves the postulated rupture of a pipe belonging to the Reactor Coolant boundary. It is conservatively assumed that the system pressure is reduced rapidly and the Emergency Core Cooling System is initiated to introduce water into the Reactor Coolant System. The Safety injection signal also initiates a turbine and reactor trip.

   The criteria for locating design basis pipe ruptures used in the design of the supports and restraints of the Reactor Coolant System in order to assure continued integrity of vital components and Engineered Safety Features is given in Section 3.6.

   Analyses reported in Reference 1 and service experiences show that the criteria given in Section 3.6 insure the protection of public health and safety. Westinghouse Nuclear Steam Supply System piping is designed to these criteria. Westinghouse performed the analysis for McGuire reactor coolant loop equipment.

   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 15 in Section 5.2.9.

2. Steam Line Break

   For component evaluation, the following conservative conditions are considered:

   a. The reactor is initially in hot, zero power subcritical condition assuming all rods in except the most reactive rod which is assumed to be stuck in its fully withdrawn position.

   b. A steam line break occurs inside the Containment resulting in a reactor and turbine trip.

   c. After the break the reactor coolant temperature cools down to 212°F.

   d. The emergency Core Coolant System pumps restore the reactor coolant pressure.

   The above conditions results in the most severe temperature and pressure variations which the component encounters during a steam break accident.

   The dynamic reaction forces associated with circumferential steam line breaks are considered in the design of supports and restraints in order to assure continued integrity of vital components and Engineered Safety Features. Protection Criteria against dynamic effects associated with pipe breaks is covered in Section 3.6.

3. Steam Generator Tube Rupture

   This accident postulates the double ended rupture of a steam generator tube resulting in a decrease in pressurizer level and Reactor Coolant System pressure.

Reactor trip occurs due to a safety injection signal on low pressurizer pressure coincident with low pressurizer water level. The planned procedure for recovery from this accident calls for isolation of the steam line leading from the affected steam generator. Therefore, this accident results in a transient which is not more severe than that associated with a reactor trip.

4. Safe Shutdown Earthquake

   The stresses resulting from the safe shutdown earthquake are considered on a component basis.

The above design conditions are given in the Equipment-Specifications which are written in accordance with the ASME Code.

The design transients and the number of cycles of each that is normally used for fatigue evaluations are shown in Tables 5-2 and 5-49. In accordance with the ASME Boiler and Pressure Vessel Code, faulted conditions are not included in fatigue evaluations. The loading combinations used in the design of reactor coolant boundary components are given in Table 5-3.

Prior to startup the following tests were carried out:

1. Turbine Roll Test

   This transient is imposed upon the unit during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature and the steam generated is used to perform a turbine roll test. However, the system cooldown during this test exceeds the 100°F per hour maximum rate.

2. Hydrostatic Test Conditions

   The pressure tests are outlined below:

   a. Primary Side Hydrostatic Test Before Initial Startup

      The pressure tests covered by this section include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed prior to initial fuel loading at a water temperature which is compatible with reactor vessel fracture prevention criteria requirements and a maximum test pressure of 3106 psig (1.25 times the design pressure, or 2690 psig, depending on the component). In this test, the primary side of the steam generator is pressurized coincident with no pressurization of the secondary side. The Chemical and Volume Control System provides the means to hydrostatically test the Reactor Coolant System.

   b. Secondary Side Hydrostatic Test Before Initial Startup

      The secondary side of the steam generator is pressurized to 1481 psia or 1.25 times the design pressure of the secondary side coincident with the primary side at 150 psig.

   c. Primary Side Leak Test

      After each time the primary system has been opened, a leak test would be performed. For design purposes, the primary system pressure is assumed to be raised to 2500 psia during the test with the system temperature above design transition temperature, while the system is checked for leaks.

      This test is no longer performed separate from normal heatup and pressurization.

   Since the tests outlined under items a. and b. occur prior to startup, the number of cycles is independent of unit life.

5.2.1.6 Identification of Active Pumps and Valves

Pumps and valves are classified as either active or inactive components for faulted conditions. Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered for the respective operating condition categories. Inactive components are those whose operability (e.g., valve opening, or closure pump operation or trip) are not relied upon to perform the system function during the transients or events considered in the respective operating condition category. The reactor coolant pumps which are the only pumps in the Reactor Coolant System boundary are classified as inactive for pipe rupture. Table 5-5 lists the active and inactive valves in each line connected to the Reactor Coolant System up to and including the system boundary. All of the active valves in this table are check valves except for the pressurizer safety valves and PORVs. The check valves function almost immediately when a ΔP is exerted across the disc.

Every valve and pump is hydrostatically tested to ASME Boiler and Pressure Vessel Code requirements to insure the integrity of the pressure boundary parts. This test is followed by a seat leak test to MSS-SP-61 criteria to insure that no gross deformation is caused by the hydrostatic test.

The control and instrumentation are discussed in Chapter 7.

5.2.1.7 Design of Active Pumps and Valves

There are no pumps within the reactor coolant pressure boundary which act as active components as defined in Section 5.2.1.6.

The design criteria for active valves described in Section 5.2.1.6 are specified in ASME Boiler and Pressure Vessel Code (1971). These valves are designed for seismic loading in accordance with ASME Code Section III. Table 3-6 lists system valve classifications.

5.2.1.8 Inadvertent Operation of Valves

Those valves used in the isolation of the Reactor Coolant System boundary during normal operation, and not relied upon to function after an accident, are redundant. The inadvertent movement of one of these redundant valves does not serve to increase the severity of any transient.

5.2.1.9 Stress and Pressure Limits

ASME Class 1 components (piping included) were designed and analyzed in accordance with ASME Section III Code requirements. The loading conditions and associated stress limits (for normal and upset conditions) were as noted in the Code. For faulted conditions, the stress intensity limits presented in Table 5-4 were used.

5.2.1.10 Stress Analysis for Structural Adequacy

The design evaluation of the Reactor Coolant System, including the type of analyses that are performed to ensure the performance and the structural adequacy of the Reactor Coolant System, is provided in the "Design Evaluation" below. Also provided are additional evaluations and stress analyses that were performed as part of the Steam Generator Replacement Project.

Design Evaluation

The Reactor Coolant System provides for heat transfer from the reactor to the steam generators under conditions of forced circulation flow and natural circulation flow. The heat transfer capabilities of the Reactor Coolant System are analyzed in Chapter 15 for various transients.

The heat transfer capability of the steam generators is sufficient to transfer, to the steam and power systems the heat generated during normal operation, and during the initial phase of cooldown under natural circulation conditions.

During the second phase of cooldown and during cold shutdown and refueling, the heat exchangers of the Residual Heat Removal System are employed. Their capability is discussed in Section 5.5.7.

The pumps of the Reactor Coolant System assure heat transfer by forced circulation flow. Design flow rates are discussed in conjunction with the reactor coolant pump description in Section 5.5.1.

Initial Reactor Coolant System tests were performed to determine the total delivery capability of the reactor coolant pumps. Thus, it was confirmed prior to operation that adequate circulation is provided by the Reactor Coolant System.

To assure a heat sink for the reactor under conditions of natural circulation flow, the steam generators are at a higher elevation than the reactor. In the design of the steam generators consideration is given to provide adequate tube area to ensure that the residual heat removal rate is achieved with natural circulation flow.

Whenever the boron concentration of the Reactor Coolant System is reduced, operation is such that good mixing is provided in order to ensure that the boron concentration is maintained uniformly throughout the Reactor Coolant System.

Although mixing in the pressurizer is not achieved to the same degree, the fraction of the total Reactor Coolant System volume which is in the pressurizer is small. Thus the pressurizer liquid volume is of no concern with respect to its effect on boron concentration.

Also, the design of the Reactor Coolant System is such that the distribution of flow around the system is not subject to the degree of variation which would be required to produce nonhomogeneities in coolant temperature or boron concentration as a result of areas of low coolant flow rate. An exception to this is the pressurizer, but for the same reasons as discussed above, it is of no concern. Operation with one reactor coolant pump inoperable is possible under certain conditions, and in this case there would be backflow in the associated loop, even though the pump itself is prevented from rotating backwards by its anti-rotation device. The backflow through the loop would cause departure from the normal temperatures distribution around the loop but would maintain the boron concentration in the loop the same as that in the remainder of the Reactor Coolant System.

The range of coolant temperature variation during normal operation is limited and the associated reactivity change is well within the capability of the rod control group movement.

For design evaluation, the heatup and cooldown transients are analyzed by using a rate of temperature change equal to 100°F per hour which corresponds to abnormal heatup and cooldown conditions. Over certain temperature ranges, fracture prevention criteria imposes lower limit to heatup and cooldown rates.

Operating procedures require that operators maintain the NC System boron concentration greater than required Shutdown Margin (SDM) boron concentration during cooldown. This SDM boron concentration includes the affects of temperature on core reactivity.

It is therefore concluded that the temperature changes imposed on the Reactor Coolant System during its normal modes of operation do not cause any abnormal or unacceptable reactivity changes.

The design cycles as discussed in the preceeding section are conservatively estimated based on engineering judgment and experience for equipment design purposes and are not intended to be an accurate representation of actual transients.

Certain design transients, with an associated pressure and temperature curve, have been chosen and assigned an estimated number of design cycles for the purpose of equipment design. These curves

represent an envelope of pressure and temperature transients on the Reactor Coolant System boundary with margin in the number of design cycles chosen based on operating experience.

To illustrate this approach, the reactor trip transient can be mentioned. Two hundred thirty design cycles are considered in this transient. One cycle of this transient would represent any operational occurrence which would result in a reactor trip. Thus, the reactor trip transient represents an envelope design approach to various operational occurrences.

This approach provides a basis for fatigue evaluation to ensure the necessary high degree of integrity for the Reactor Coolant system components.

System hydraulic and thermal design parameters are used as the basis for the analysis of equipment, coolant piping, and equipment support structures for normal and upset loading conditions. The analysis is performed using a static model to predict deformation and stresses in the system. Results of the analysis give six generalized force components, three bending moments and three forces. These moments and forces are resolved into stresses in the pipe in accordance with the applicable codes.

In addition to the loads imposed on the system under normal and upset conditions, the design of mechanical equipment required that consideration also be given to abnormal loading conditions such as seismic and pipe rupture.

Analysis of the reactor coolant loops and support systems for seismic loads is based on a three dimensional, multi-mass elastic dynamic model. The calculated floor spectral accelerations are used as input forcing functions to the detailed dynamic model which includes the effects of the supports and the supported equipment. The loads developed from the dynamic model are incorporated into a detailed loop and support model to determine the support member stresses.

The dynamic analysis employs the displacement method, lumped parameter, stiffness matrix formulations and assumes that all components behave in a linearly elastic manner. Seismic analyses are covered in detail in Section 3.7.

Analysis of the reactor coolant loops and support systems for blowdown loads resulting from a loss of coolant accident is based on the time history response of simultaneously applied blowdown forcing functions on a single broken and unbroken loop dynamic model. The forcing functions are defined at points in the system loop where changes in cross section or direction of flow occur such that differential loads are generated during the blowdown transient. Reference 15 in Section 5.2.9 provides the basis for eliminating previously postulated reactor coolant system pipe breaks with the exception of those breaks at branch connections. Stresses and loads are checked and compared to the corresponding allowable values.

The stresses in components resulting from normal sustained loads and the blowdown analysis are combined with the seismic analysis to determine the maximum stress for the combined loading case. This is considered a very conservative method since it is highly improbable that both maximums will occur at the same instant. These stresses are combined to demonstrate that the reactor coolant loops and support system does not lose its intended functions under this highly improbable situation.

Protection criteria against dynamic effects associated with pipe breaks are described in Section 3.6.

For fatigue evaluation, in accordance with the ASME Boiler and Pressure Vessel Code, maximum stress intensity ranges are derived from combining the normal and upset condition transients given in Section 5.2.1.5. The stress ranges and number of occurrences are then used in conjunction with the fatigue curves in the ASME Boiler and Pressure Vessel Code to get the associated cumulative usage factors.

The criterion presented in the ASME Boiler and Pressure Vessel Code is used for the fatigue failure analysis. The cumulative usage factor is less than 1.0, and hence the fatigue design is adequate.

The reactor vessel vendor's stress report is reviewed and approved by Westinghouse Electric Corporation. The stress report includes a summary of the stress analysis for regions of discontinuity analyzed in the

vessel, a discussion of the results including a comparison with the corresponding code limits, a statement of the assumptions used in the analyses, descriptions of the methods of analysis and computer programs used, a presentation of the actual calculations used, a listing of the input and output of the computer programs used, and a tabulation of the references cited in the report. The content of the stress report is in accordance with the requirements of the ASME Boiler and Pressure Vessel Code and Duke's Design Specification.

A stress analysis of the McGuire and Catawba replacement steam generators was completed in accordance with the 1986 ASME Code Section III, Division 1, Class 1 (Ref. 1) for the Level A, B, C and D service condition loading and for the test condition loading for component fatigue evaluation that are specified in the Duke Specification DPS-1201.01-00-0002.

Finite element and classical methods were used to determine the stresses at critical locations.

For the nonductile failure analysis, an ASME Code Appendix G analysis was completed to meet the requirements of NB-3211 (d) of the ASME Code.

The McGuire and Catawba Replacement Steam Generator pressure boundary was shown to satisfy the requirements of the 1986 ASME Boiler and Pressure Vessel Code, Section III (Ref. 1) for the Level A, B, C, and D service condition loading and for the test condition loading for component fatigue evaluation that are specified in the Duke Replacement Steam Generator Design Specification.

Tubing for the BWI steam generators meets the requirements of the ASME Code for the Design, Test and Levels A, B, C and D Service (accident conditions) loading conditions specified in the DPC Certified Design Specification.

Tube to tubesheet attachment welds are made in accordance with ASME NB-4350. In addition, it is shown by analysis that the welds meet the requirements of the ASME Code when subjected to tube axial forces and torsional moments under the Design, Test and Levels A, B, C and D Service (accident conditions) loading conditions specified in the DPC Certified Design Specification.

It is concluded that the tubes meet the ASME Section XI, IWB-3630 for OD flaws. A wasted tube with 40% loss of nominal wall thickness uniformly around the OD satisfies acceptance criteria for the minimum acceptable wall thickness established in Regulatory Guide 1.121 paragraph C.2 and the ASME Code. Loads are based on Regulatory Guide 1.121 paragraph C.3 [a]-[c] from the nominal and faulted conditions given in the DPC Certified Design Specification. Note that an additional tube thickness allowance should be added to the analyzed minimum acceptable tube wall thickness to establish the operational tube thickness acceptable for continued service, per Regulatory Guide 1.121 C.2 (b). This tube shall exhibit an overall fatigue strength reduction factor (FSRF) no larger than 2.15 in the U-bend region above the top lattice grid or 2.75 in the straight tube section below the top lattice grid in consideration of geometric and/or environmental effects. The limiting FSRF's were conservatively derived in the fatigue analysis based on the entire 60 year design service life. Higher FSRF may be justified for shorter service intervals between tube inspection periods.

The vessels, piping, valves, pumps, and supports of the reactor coolant pressure boundary are designated ANS Safety Class 1.

ASME Section III, Class 1 component, including piping, are designed and analyzed in accordance with the ASME Section III requirements. For faulted conditions, the stress intensity limits presented in Table 5-4 are used.

Valves in sample lines are not considered to be part of the Reactor Coolant System boundary i.e., not ANS Safety Class 1. This is because the nozzles where these lines connect to the Reactor Coolant System are orificed to a 3/8 inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging.

Analytical Methods for Supports and Loop Analysis - Westinghouse Methodology

The load combinations that are considered in the design of structural steel members of component supports are given in Section 5.2.1.5. The design is described in Section 5.5.14. The following are definitions of terms used in the analysis:

Deadweight - The deadweight loading imposed by the piping on the supports is defined to consist of the dry weight of the coolant piping and the weight of the water contained in piping during normal operation. In addition, the total weight of the primary equipment components including water forms a deadweight loading on the individual component supports.

Thermal Expansion - The free vertical thermal growth of the reactor vessel nozzle centerlines is considered to be an external anchor movement transmitted to the reactor coolant loop (RCL). The weight of the water in the steam generator and reactor coolant pump is applied as an external force in the thermal analysis to account for equipment nozzle displacement as an external movement, that causes additional secondary stresses in RCL piping.

The cold and hot moduli of elasticity, the coefficient of thermal expansion at the metal temperature, external movements transmitted to the piping as described above, and the temperature rise above the ambient temperature define the required input data to perform the flexibility analysis for thermal expansion.

Earthquake Loads - The intensity and character of an earthquake motion which produces forced vibration of the equipment mounted within the Containment building are specified in terms of the floor response spectrum curves at various elevations within the Containment building. The OBE floor response spectrum curves for earthquake motions at various elevations are given in Section 3.7.

Pressure - The steady state hydraulic forces based on the system initial pressure are applied as external loads to the RCL model for determination of the RCL/support system deflections and support forces.

Pipe Rupture Loads - Blowdown loads are developed in the broken and unbroken reactor coolant loops as a result of the transient flow, pressure fluctuations following a postulated loss-of-coolant-accident (LOCA) in one of the reactor coolant loops. The postulated LOCA is assumed to have one-milli-second opening time to simulate the instantaneous occurrence.

Analytical Methods - The static and dynamic structural analyses assume linear elastic behavior and employ the displacement (stiffness) matrix method and the normal mode theory for lumped-parameter, multi-mass structural representation to formulate the solution. The complexity of the physical system to be analyzed requires the use of a computer for solution. Herein lies the need for accurate and adequate representation of the physical system by means of an idealized (mathematical) model.

The loadings on the component supports are obtained from the analysis of an integrated reactor coolant loop supports system dynamic structural model as shown in Figure 5-8. With regard to FSAR Figure 5-8,

1. The spring elements which are shown at locations of component supports and restraints represent stiffness matrices which are computed to represent the restraint or lack of restraint provided in translational and rotational directions at each location of support or piping restraint.
2. There are six degrees-of-freedom of each node point and at each support location.
3. At locations where there is a restraint located at the outer diameter of the pipe, and the piping mass is lumped on the center line, the local flexibility of the piping is considered in determining the overall stiffness of the restraint. This modeling technique was developed so as to allow calculation of an accurate total flexibility at the restraint location and, subsequently, allow accurate calculations of the loadings expected in the pipe and at the restraint.

Computational Methods - The basic computer programs used in the structural analysis of the reactor coolant loop are described below. All of these programs have been tested by hand calculations and compared with known solutions and closed form solutions with satisfactory agreement.

STRUDL

STRUDL, part of the ICES civil engineering computer system (Reference 2), is a general purpose matrix structural analysis program which can solve for stresses and deflections of structures subjected to static or thermal loads. The basis of the program is the general beam finite element. It is applicable to linear elastic two- and three-dimensional frame or truss structures, e.g., steam generator lower, steam generator upper lateral, and reactor coolant pump lower support structures. STRUDL employs the stiffness formulation, and is valid only for small displacements. Structure geometry, topology, and element orientation and cross-section properties are described in free format. Member and support joint releases, such as pin and rollers, are specified. Otherwise, six restraint components are assumed at each end of each member and at each support joint.

The STRUDL system performs structural stability and equilibrium checks during the solution process and prints error messages if these conditions are violated. However, the system cannot detect geometry or topology errors. Type, location, and magnitude of applied loads or displacements are specified for any number of loading conditions. These can be combined as desired during the solution process.

One important feature of STRUDL is that any desired changes, deletions, or additions can be made to the structural model during the solution process. This produces results for a number of structure configurations, each with any number of loading conditions.

The output includes member forces and distortions, joint displacements, support joint reactions, and member stresses.

Analysis procedures for component supports are discussed further in Section 3.8.3.4.

STASYS

STASYS is a 1, 2, and 3-dimensional finite element program capable of solving elastic-plastic structural problems, transient and steady state thermal problems, and linear and non-linear dynamic structural problems.

For static problems the following element types are available:

1. 2 & 3 dimensional pin-jointed bars (spars)
2. 2 & 3 dimensional beam elements
3. Constant and linear strain triangular elements for plane stress, plane strain, and axisymmetric analysis
4. Three dimensional solid elements (6 and 8 cornered)
5. Triangular plate and shell elements
6. Axisymmetric shell elements
7. Non-linear effects (breaking bar and friction element)
8. 3 Dimensional Pipe and Elbow elements

For dynamic problems the spar, beam, pipe, and non-linear elements may be combined with springs, lumped masses and viscous dampers. STASYS is capable of predicting mode shapes and natural frequencies, maximum response to harmonic excitation, or complete time history response to arbitrary forcing functions.

STHRUST

STHRUST calculates (for a loss of coolant accident in a pressurized water system) the transient (blowdown) forces exerted by the fluid on the primary coolant loop system. The program uses the results from the SATAN program (transient pressures, flow rates and other coolant properties as a function of

time) as input and calculates forces at up to 26 locations (elbows, pumps, steam generator plenums, tubes, etc.) along the coolant system piping as a function of time.

FIXFM

FIXFM is a digital computer program which determines the time-history response of a three-dimensional structure excited by an internal forcing function. FIXFM accepts (input) normalized mode shapes, natural frequencies, forcing functions, and an initial deflection vector inserted, and the geometry stiffness an damping properties of non-linear elements.

The program sets up the modal differential equations of motion. The modal differential equations are then solved numerically by a predictor-corrector technique of numerical integration. The modal contributions are then summed at various modal or mass points throughout the structure to get the actual timehistory response.

During the course of the solution the program checks the non-linear elements to see if they are active (in contact) or inactive. If an element is active the program then calculates the force the element would have on the system from its stiffness and damping properties and the deflection and velocity vectors at the attachment point.

WESTDYN

WESTDYN is a special purpose program designed for the static and dynamic solution of redundant piping systems with arbitrary loads and boundary conditions. It computes, at any point in the piping system, the stresses, forces, moments, translations, and rotations which result from the imposed anchor or junction loads in any combination of three orthogonal axes. The section properties have been specialized to piping cross sections plus the addition of curved members or elbow. Valves may also be represented as stiffer members. The piping system may contain a number of sections, a section being defined as a sequence of straight and/or curved members lying between two network points. A network point is 1) a junction of two or more pipes, 2) an anchor or any point at which motion is prescribed, or 3) any arbitrary point.

Any location in the system may sustain prescribed loads or may be subject to elastic constraint in any of its six degrees of freedom. For example, hangers may be arbitrarily spaced along a section and may be of the rigid, flexible, or constant force type.

The response to seismic excitation is determined by using normal mode techniques with a lumped mass system. The maximum spectral acceleration is applied for each mode at its corresponding frequency from response spectra. A basic assumption is that the maximum modal excitation of each model occur simultaneously. The modal participations are then summed.

THESSE

THESSE performs reactor coolant loop equipment support structures analysis and evaluation. Two versions are used: one for normal, upset and emergency condition leading using A1SC-69 allowable stress equations and the other for faulted condition loading where loss of coolant accident loads are used in time-history form and ultimate stress equations are used; loads on the structure are combined, transformed to the structure coordinate system and multiplied by member influence coefficients. The resulting member forces are then used with member properties in stress and interaction equations to determine the adequacy of each member in the structure.

SATAN

The SATAN computer code is discussed in Section 15.4.

Reactor Coolant Loop Model

The reactor coolant loop (RCL) model is constructed for the WESTDYN computer program. This is a special purpose program designed for the static and dynamic analysis of redundant piping systems with arbitrary loads and boundary conditions. The RCL lumped-mass model represents an ordered set of data that numerically describes the physical system to WESTDYN program. The node point coordinates and the incremental lengths of the elements are calculated. The lumping of distributed mass of a segment or elbow is accomplished by locating the total mass at the mass center of gravity.

The valid representation of the effect of the equipment motion on the RCL piping and its support system is assured by modeling the mass and stiffness characteristics of the equipment in the overall RCL model. Since the reactor pressure vessel is very massive and relatively rigid, it is represented by a fixed boundary condition for the RCL model. The requirement in the time history dynamic analysis, that the external forcing functions be applied at only mass points, influences the construction of the steam generator and reactor coolant pump models described below.

The steam generator is represented by a three-mass, lumped model. The lower mass position is located at the intersection of the inlet and outlet nozzles of the steam generator. The middle mass position is located at the steam generator upper support elevation. The upper mass position is located at the top of the steam generator.

The reactor coolant pump is represented by a two-mass, lump model. The lower mass position is located at the intersection of the pump section and discharge nozzles. The upper mass position is located at the center of gravity of the pump motor.

Hydraulic Models

The hydraulic model is constructed to quantitatively represent the behavior of the coolant fluid within the reactor coolant loops in terms of the concentrated time-dependent loads which it imposes upon the loops.

In evaluating the hydraulic forcing functions during a loss-of-coolant accident, the pressure and the momentum flux terms are dominant. Inertia and the gravitational terms are neglected; however, they are taken into account to evaluate the local fluid conditions.

Thrust forces resulting from a LOCA are calculated in two steps using two digital computer codes. The first code SATAN (Reference 3), calculates transient pressure, flow rates and other coolant properties as a function of time. The second code, STRUST, uses the results obtained from the first code and calculates time history of forces at locations where there is a change in either direction or area of flow within the RCL.

In SATAN blowdown analysis, both the broken and the unbroken loops are represented. The SATAN code employs a one-dimensional, control volume approach in which the entire primary coolant system is divided into approximately 65 elements. The fluid properties are considered uniform and thermodynamic equilibrium is assumed within each element. Pump characteristics such as coast down and cavitation, core and steam generator heat transfer, as well as nuclear kinetics, are properly simulated.

In the STHRUST calculation of blowdown forces, the RCS is represented by the same model employed in the SATAN code. Twenty-six node points are selected along the geometric model of the RCL where the vector forces and their coordinate components are calculated.

The force components at each aperture are vectorially summed to obtain the total force components in global coordinate system at the nodes. These forces are stored on magnetic tape and, after proper coordinate transformation, applied as external loadings on the RCL cynamic model.

Static Load Solutions

The static solutions for deadweight, thermal expansion and pressure load conditions are obtained by using the WESTDYN computer program. The computer input consists of the RCL model, stiffness matrices representing various supports for static behavior, and the appropriate load condition. Coordinate

transformations for rotation from the local or support coordinate system to the global system are applied to the stiffness matrices prior to their input.

Normal Mode Response Spectral Seismic Load Solution

The stiffness matrices representing various supports for dynamic behavior are incorporated into the RCL model after transformations for rotation from local to the RCL global system. The response spectra for the OBE or SSE load case are applied along a horizontal and vertical axis simultaneously. From the input data, the overall stiffness matrix of the three-dimension RCL is generated. The stiffness matrix is manipulated to obtain a reduced stiffness matrix associated with the mass points only. The reduced matrix is inverted to give the flexibility matrix of the system. A product matrix (also known as the dynamical matrix) formed by the multiplication of the flexibility and mass matrices is used to solve for the natural frequencies and normal modes by the modified Jacobi method. The modal participation factor matrix is computed and combined with the appropriate seismic response spectra values to give the amplitude of the modal coordinate for each mode. Then the forces, moments, deflections, rotations, support structure reactions and piping stresses are calculated for each significant mode. The total seismic response is computed by combining the contributions of the significant modes by the square root of the sum of the square method.

Time History Dynamic Solution for LOCA Loading

The initial displacement configuration of the mass points is computed by applying the initial steady state condition to the unbroken RCL model. For this calculation, the support stiffness matrices for the static behavior are incorporated into the RCL model. For dynamic solution, the unbroken RCL model is modified to simulate the physical severance of the pipe due to the postulated LOCA under consideration. This model includes definition of the support stiffness matrices for dynamic behavior. The natural frequencies and normal modes for the modified RCL dynamic model are determined. After proper coordinate transformation to the RCL global coordinate system, the hydraulic forcing functions to be applied to each lumped mass point are stored on magnetic tape for later as input to the FIXFM program.

The initial displacement conditions, natural frequencies, normal modes and the time-history hydraulic forcing functions from the input to the FIXFM program which calculates the dynamic time-history displacement response for the dynamic degrees of freedom in the RCL model. The displacement response is plotted at all mass points. The displacement response at support points is reviewed to validate the use of the chosen support stiffness matrices for dynamic behavior. The time-history displacement response from the valid solution is saved on magnetic tape for later use to compute loads and to analyze the RCL piping stresses.

Analytical Methods for Supports and Loop Analysis - Steam Generator Replacement Methodology

The reactor coolant system was reanalyzed to take into account the effects of the Babcock and Wilcox International (BWI) replacement steam generators. This analysis was preformed by B&W Nuclear Technologies (BWNT) of Lynchburg, VA. The reanalysis was defined to be a parametric analysis where the response of the reactor coolant system with the replacement steam generators was compared to the system response with the original steam generators.

The finite element method was used in obtaining the solution for the static, seismic dynamic, and loss-of-coolant-accident dynamic analyses. The response specturm method was used to generate the results for the seismic analysis while the time history method was used to generate the results for the loss-of-coolant-accident analysis. The integrated reactor coolant system model consisted of the piping, components, and component supports. The component supports are described in Section 5.5.14. The NSSS model was coupled to the containment interior structure model (Figure 3-20) to facilitate the input of the design ground response spectra for the seismic analysis.

The NSSS system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, component supports and the containment interior structure. The model geometry is based on the reactor coolant loop piping layout and equipment drawings. All joint coordinates and piping/equipment element lengths are obtained from the drawings. The physical and material properties of each element are obtained from the drawings, specifications, and the ASME code. The piping and replacement steam generators were modeled using consistent mass model elements; the reactor vessel, reactor coolant pumps, and the containment interior structure were modeled with discrete lumped masses. The component supports are represented by spring elements except for the steam generator and reactor coolant pump columns which are modeled as beam elements.

The static solutions for the deadweight and thermal loading conditions are calculated using the stiffness method of analysis. The stiffness matrix and load vector are assembled and solved using the BWNT proprietary computer code BWSPAN (see Reference 16 in Section 5.2.9). The piping deflections, internal forces, and stresses were determined at each node point. Support loads were calculated for each active component support. Operating steady state pressure loads (see previous section entitled "Analytical Methods for Supports and Loop Analysis - Westinghouse Methodology") were not considered in the NSSS reanalysis. BWNT evaluated these steady state hydraulic loads and determined that they could be deemed negligible.

The steam generator upper and lower lateral supports are inactive during plant heatup, cooldown, and normal plant operating conditions. This is also true of the reactor coolant pump lateral restraints. (The reactor vessel vertical and lateral restraints are subject to thermal loads as are the steam generator and reactor coolant pump columns.) There lateral restraints become active (only) when the plant is at power. All dynamic analyses are performed for the full power condition.

The seismic analysis of the NSSS model uses the responses spectrum method, which is based on the principal of modal superposition. Damping values are specified in Section 3.7.1.3 and Table 3-25. Nonproportional damping that reflects the material composition of the model (as described in Section III, Appendix N of the ASME Code) is used in the analysis. The plant design ground response spectra (see figures in Former Appendix 2E) are used in the seismic analysis as the NSSS model is coupled to the containment interior structure model. The spectra are applied along each horizontal axis simultaneously with the vertical axis (i.e. a pair of two-dimensional spectrum analyses are performed). The seismic acceleration in the vertical direction is two-thirds of that applied in the horizontal direction. The modal responses are combined as described in Section 3.7.3.4. The three components of earthquake motion are then combined as described in Section 3.7.3.7 in order to obtain the maximum forces, moments, deflections, rotation, support reactions, and piping stresses.

The NSSS piping model was also evaluated for loss of coolant accident loads. The double ended guillotine breaks on the main reactor coolant loop piping were eliminated through the application of leak-before-break (see Section 3.9.1.5). Break loads were only evaluated for primary system branch line (pressurizer surge line, accumulator line, and residual heat removal line) and secondary system line (main steam line and main feedwater line) pipe breaks. The piping finite element model was subjected to a time history load for each defined pipe break case. The force time histories were calculated using the CRAFT2 computer code (see Reference 17 in Section 5.2.9). Support participation was reviewed and those supports at which the specified gap did not close were considered to be inactive. Asymmetric pressure load time histories were applied concurrently with the pipe rupture force time histories. All piping forces, moments, displacements, and stresses as well as components support loads were calculated using the BWSPAN computer code.

The thermal transients associated with the replacement steam generators were compared to the original system design thermal transients. The original design thermal transients were found to bound the thermal transients for the replacement steam generators. Stresses due to the thermal transients were not evaluated further.

Primary Components Reanalysis for Steam Generator Replacement

The results of the reactor coolant loop analysis with replacement steam generators were used to evaluate the qualification of the reactor vessel, reactor coolant pumps, and replacement steam generators. The nozzle loads calculated in the NSSS reanalysis were compared to those used for the original equipment design to show all primary components were acceptable. Other loading conditions for the existing primary components did not change.

The replacement steam generators were designed for the loading combinations shown in Table 5-3. The replacement generators were designed to withstand normal loads (deadweight, pressure, and thermal), mechanical transients (OBE, SSE, and pipe rupture, including the effects of asymmetric pressure loads), and pressure and temperature transients associated with normal and abnormal plant operating conditions (see Section 5.2.1.5). Umbrella nozzle loads for the replacement steam generators were supplied for all loading conditions. Actual nozzle loads from the reactor coolant loop reanalysis were compared to the umbrella loads to insure that the design was acceptable. The seismic analysis for the replacement steam generators was performed using damping values of 2% for OBE and 3% for SSE per Regulatory Guide 1.61. The replacement steam generators were designed to meet all requirements of Section III, Subsection NB of the ASME Boiler and Pressure Vessel Code.

Changes to piping not in the primary loop analysis was performed as described in Section 3.9.

5.2.1.11 Analysis Method for Faulted Condition

The analysis method for faulted condition is provided in Section 5.2.1.10 above.

5.2.1.12 Protection Against Environmental Factors

A discussion of the protection provided for the principal components of the Reactor Coolant System against environmental factors is found in Chapter 3.

5.2.1.13 Compliance with Code Requirements

A brief description of the analyses and methods used to assure compliance with the applicable codes is provided in Section 6.2.1.3.

5.2.1.14 Stress Analysis for Emergency and Faulted Condition Loadings

The stress analyses used for Faulted Condition Loadings are discussed in Section 5.2.1.10. These are no emergency conditions specified.

5.2.1.15 Stress Levels in Category 1 Systems

For Class 1 components and piping, the loading combinations used in the analyses are supplied in Table 5-3. The design and analysis for normal and upset conditions are in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The stress limits for faulted conditions are given in Table 5-4. The design bases and loading conditions for the reactor internals are provided in Section 4.2.2. The dynamic analysis for reactor internals are provided in Section 3.9.1.

The analysis of the reactor coolant system is described in detail in Section 5.2.1.10. The results of the system analysis give loads on the component nozzles and at the component/support interface locations. These nozzle and component/support interface loads are either: (1) applied to the component in a detailed stress analysis to demonstrate structural adequacy, or (2) used to demonstrate that the multi-plant umbrella loads used in the component analysis are adequate for the McGuire units. The design allowable stress criteria for component supports is defined in Section 5.5.14. Detailed analyses are also performed on the component pressure boundaries to determine loads and stresses for seismic conditions, accident

conditions, pressure, deadweight, and transients. These stresses are combined to demonstrate the adequacy of the Class 1 pressure boundary. All stresses in the component are shown to be less than those allowed by the Codes and faulted criteria mentioned previously.

The results of the stress evaluation of the reactor coolant loop piping are summarized below. Presented are summary results for the design and faulted condition primary stress intensity calculations, which were performed in accordance with NB-3650 (Section III of the ASME Code). Summary results for all other loading combinations are also provided.

PRIMARY STRESS EVALUATION

Design Conditions

The stress intensities due to primary loadings of design pressure, weight, and OBE are combined using Equation 9 of NB-3650 of the Code. The resultant moment (m_i) in the Equation calculations combines loads from weight and OBE.

The maximum stress intensities due to design pressure, weight, and OBE in the reactor coolant loop piping and the Code allowable stress intensity values are presented in Table 5-8.

Emergency Conditions

There are no emergency conditions or emergency transients for McGuire.

Faulted Conditions

The maximum pressure in all faulted transients is less than the operating pressure at 100% power. This result indicates that the permissible pressure of 2.0 P, where P is the design pressure defined in the Code, is not exceeded.

The stress intensities due to primary loadings of maximum operating pressure, weight, SSE, and pipe rupture as identified in Section 3.6.2.1.1 are combined using Equation 9 of the Code. The resultant moment loading Mi) in the Equation 9 calculations combines loads from weight, pipe rupture, and SSE. Pipe rupture loads include the effects of reactor pressure vessel motion due to cavity pressure and internals hydraulics loads.

Table 5-8 summarizes the faulted condition piping stress evaluation.

FATIGUE EVALUATION (Normal, Upset and Test Conditions)

Piping evaluation for the reactor coolant loop, in accordance with the rules of the Code, requires fatigue evaluation for normal, upset, and test conditions. These design transient conditions are described in Section 5.2.1.5. The analysis presented in NB-3653 of the Code provides a simplified conservative method of assuring that piping stress intensities and fatigue usage factors are in conformance with the requirements of the Code.

The moments resulting from the reactor coolant loop seismic stress analyses are used as part of the input in the fatigue evaluation of the system. Separate static analyses were performed for seismic anchor movements to generate the moment responses from this load source.

A transient thermal analysis was performed for each of the normal, upset and test conditions specified in Section 5.2.1.5 to determine the time-history temperature distribution at selected radial positions across the pipe wall. Stress-producing items ΔT_1, ΔT_2, and &alpha$_a T_a$ - $\alpha_b T_b$ (as defined in the Code) were calculated from the time history temperature distribution across the wall. Each transient was described by at least two load sets representing the maximum and the minimum stress states.

Following the initiation of a thermal transient, the average temperature of the pipe wall varies as a function of time. The average temperature contributes to the moment loads through general thermal expansion of the reactor coolant loop. Two thermal expansion analyses, corresponding to the maximum

and the minimum pipe wall average temperatures or fluid temperatures were conservatively performed to generate the maximum and minimum moment sets for stress and fatigue evaluations. Thermal anchor movements were included in these analyses. The combination of these moments and the associated peak transient responses with the other required load sets produced conservative stress intensity ranges and fatigue usage factors.

The results of the reactor coolant loop piping fatigue analysis are shown in Table 5-8 for maximum stress points in the hot, cold, and crossover legs. The following paragraphs summarize these results:

The Code limit on the primary-plus-secondary stress intensity range, S_n, defined by Equation 10 of the Code, was exceeded at various locations because of the conservative approach adopted in generating the load sets for transient loadings. At all of these locations, the simplified elastoplastic discontinuity analysis outlined in the Code was performed to show the satisfaction of the Code stress and fatigue requirements.

The Code limit on stress intensity range S_e (Equation 12) was satisfied at all locations where the $3S_m$ limit on the primary-plus-secondary stress intensity range S_n (Equation 10) was exceeded. The maximum (Equation 12) stress intensity range is shown for each leg in Table 5-8.

The Code limit on the range of the primary-plus-secondary stress intensity, excluding thermal bending and thermal expansion stresses (Equation 13) was satisfied at all locations where the $3S_m$ limit on the primary-plus-secondary stress intensity range, S_n, was exceeded. The maximum stress intensity range obtained in accordance with Equation 13 is shown for each leg in Table 5-8.

Code limits on fatigue, as measured by cumulative usage factors, are satisfied at all locations on the reactor coolant loop piping. The maximum cumulative usage factor obtained for each leg is shown in Table 5-8.

Conclusions

The structural stress analysis of the reactor coolant loop piping performed for the McGuire Nuclear Station demonstrates the design adequacy and structural integrity under specified design loading conditions.

The primary stress evaluations for the design, normal, and upset conditions show the stress intensities to be below the allowable limits established in the ASME Code.

Reactor coolant loop piping stress intensities and fatigue usage factors are in conformance with the requirements of the Code for the fatigue evaluation performed under all normal, upset and test conditions. Therefore, the piping system is adequate for all design transient conditions described in Section 5.2.15.

The reactor coolant loop piping stress evaluation for the faulted condition shows the stress intensities for the unbroken legs of the broken loop and the unbroken loops are within the faulted condition allowable limits, $3S_m$, of Equation 9 (NB-3652 and Appendix F, F-1360) of the Code for the postulated breaks described in Section 3.6.2.1.1.

Primary plus secondary stress intensity ranges and fatigue cumulative usage factors confirm that breaks other than those identified in Section 3.6.2.1.1 need not be postulated. More detailed discussions on this are provided in Section 3.6.2.1.1.

In summary, the reactor coolant loop piping is adequate, and will maintain its structural integrity and meet all safety-related design requirements.

5.2.1.16 Analytical Methods for Stresses in Pumps and Valves

Pumps and valves within the Reactor Coolant System boundary are designed to meet the stress limits for faulted conditions given in Table 5-4. Analytical methods and limits for normal and upset conditions are in accordance with the applicable codes described in Table 3-4.

All pumps and valves in the RCPB are designed, built and analyzed according to Section III of the ASME Code. For those components within Westinghouse scope of supply, fabricators are required to meet the quality assurance requirements set forth in the Code and additionally, those requirements of our procurement contracts which assure that the necessary method and procedure for the design and construction of the components are followed. In addition, Westinghouse independently reviews and approves the component stress analysis reports.

All balance of plant Safety Class 1, 2, and 3 pumps and valves are designed to codes and for seismic conditions are listed in Table 3-4. All vendors of balance of plant ANS Safety Class 1, 2 or 3 equipment are verified to be on the approved vendor list prior to placement of the order for equipment. Additionally, the specification supplied to the vendor is reviewed and approved by Duke as well as the manufacturer's process procedures. The equipment is also seismically analyzed as noted in Table 3-4 and in accordance with Sections 3.7.2.1.1.8 and 3.7.2.1.1.9 for valves and pumps respectively. The seismic analysis reports are reviewed by Duke or our consultant for correctness and code compliance.

5.2.1.17 Analytical Methods for Evaluation of Pump Speed and Bearing Integrity

Reactor Coolant Pump overspeed evaluations are covered in Section 5.5.1.3.

5.2.1.18 Operation of Active Valves Under Transient Loadings

The analytical methods used in evaluating active components are described in Sections 3.7.2.1, 5.2.1.7 and 5.2.1.16. The analytical procedures used in designing Category 1 equipment includes consideration for the dynamic effects of seismic events, accidents and operation. No dynamic testing is anticipated.

5.2.1.19 Field Run Piping

Refer to Section 3.9.2.7 for a discussion of field run piping.

5.2.2 Overpressurization Protection

5.2.2.1 Location of Pressure Relief Devices

Pressure relief devices for the reactor coolant system comprise the three pressurizer safety valves and three power operated relief valves shown on Figure 5-1; these discharge to the pressurizer relief tank by common header. Other relief valves that discharge to the pressurizer relief tank are itemized in Table 5-9.

5.2.2.2 Mounting of Pressure Relief Devices

Refer to Section 3.9.2.5 for a discussion of mounting of pressure relief devices.

5.2.2.3 Report on Overpressure Protection

The pressurizer is designed to accommodate pressure increases (as well as decreases) caused by load transients. The spray system condenses steam to prevent the pressurizer pressure from reaching the setpoint of the power-operated relief valves during a step reduction in power level of up to ten percent of load.

The spray nozzle is located in the top of the pressurizer. Spray is initiated when the pressure controlled spray demand signal is above a given setpoint, 2260 psig. The spray rate increases proportionally (2%/psig) with increasing pressure rate and pressure error until it reaches a maximum value at 2310 psig.

The pressurizer is equipped with power-operated relief valves which limit system pressure for a large power mismatch and thus prevent actuation of the fixed high pressure reactor trip. The relief valves are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power operated relief valves if excessive leakage occurs. The relief valves are designed to limit the pressurizer pressure to a value below the high pressure reactor trip set point for all design transients up to and including the design percentage step load decrease with steam dump.

Power operated relief valve NC-34A operates on a compensated pressure deviation signal between pressurizer pressure and a nominal reference pressure of 2235 psig. The compensated pressure deviation signal is subjected to a proportional + integral controller. A lift signal is generated for NC-34A on a 100 psi or greater pressure difference. A reseat signal for NC-34A is generated on a pressure difference of 80 psi or less. Power relief valves NC-32B and NC-36B operate on actual pressurizer pressure. The lift and reseat setpoints for both NC-32B and NC-36B are 2335 psig and 2327 psig, respectively. The controller for NC-34A is subjected to a one second electronic lag in the actual pressurizer pressure signal during modes 1, 2, and 3. The PORVs have a two (2) second valve stroke time from the full closed to full open position.

The power operated relief valves are also used to provide protection against exceeding 10CFR50 Appendix G limits (including ASME Code Case N-514), as defined by McGuire Tech Spec Figures 3.4.3-1 and 3.4.3-2 during periods of water solid operation. Analyses have shown that one PORV is sufficient to prevent violation of these limits due to anticipated mass and heat input transients. However, redundant protection against such over-pressurization events is provided through the addition of low pressure setpoints to two PORV's, NC34A and NC32B. Since this protection is required only during low temperature water solid operation, the low pressure setpoint is enabled by the operator at reactor coolant loop temperature below 300°F. The low pressure setpoint is interlocked with reactor coolant loop temperature to minimize the possibility of inadvertent actuation. Refer to Section 7.6.17.1 for a more detailed description of the control system for the PORV's during water solid operation.

Each of the two PORV's is supplied with an independent, seismically designed supply of nitrogen which is sized to assure that no operator action is required to terminate the transient for a period up to 10 minutes. The sources of nitrogen are the Safety Injection System cold leg accumulators which are normally isolated from the reactor coolant system during periods when the low pressure setpoint is enabled. High pressure nitrogen from the accumulators will be regulated down to the required operating pressure for the PORV actuators. Relief valves will provide protection against over-pressurizing the actuators due to regulator failure. An isolation valve in each supply line will be normally closed when the low pressure setpoint is disabled in order to protect the associated accumulators against loss of pressure. Normally, the source of motive fluid for PORV actuation will be the Instrument Air System.

Administrative control is exercised to prevent inadvertent overpressurization when the Reactor Coolant System is water solid during startup or shutdown. Operating procedures for startup and shutdown are written such that the sequence of operations assures that the unit is maintained within the Technical Specifications. Additionally, these procedures contain precautions and limitations which are specified to emphasize sequences or combinations of unit conditions which are critical to the control of pressure in the Reactor Coolant System.

Duke participated in the PWR Safety and Relief Valve testing program. The data collected was used to verify the valve operability and the adequacy of the discharge piping, support integrity and nozzle loads. All portions of piping and supports in the Class I primary coolant boundary were shown to have acceptable stress levels per ASME Section III (NUREG 0737, Item II.D.1).

Isolated output signals from the pressurizer pressure protection channels are used for pressure control. These are used to control pressurizer spray and heaters and power operated relief valves. Pressurizer pressure is sensed by fast response pressure transmitters with a rapid time response.

In the event of a complete loss of heat sink, i.e., no steam flow to the turbine, protection of the Reactor Coolant System against overpressure is afforded by pressurizer and steam generator safety valves along with any of the following reactor trip functions:

1. Reactor trip on turbine trip (if the turbine is tripped)
2. High pressurizer pressure reactor trip
3. Overtemperature ΔT reactor trip
4. Low-low steam generator water level reactor trip.

Continued integrity of the reactor Coolant System during the maximum transient pressure is assured by design within the applicable codes as discussed in Reference 4. The code safety limit is 110 percent of the 2485 psig design limit.

A detailed functional description of the process equipment associated with the high pressure trip is provided in Reference 5.

The upper limit of overpressure protection is based upon the positive surge of the reactor coolant produced as a result of turbine trip under full load, assuming the core continues to produce full power. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2500 psia and a total accumulation of 3 percent. Note that no credit is taken for the relief capability provided by the power operated relief valves during this surge, but credit is taken for the self-actuated steam generator safety valves.

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray setpoints and the protection system setpoint pressures are listed in Table 5-10.

System components whose design pressure and temperature are less than the Reactor Coolant System design limits are provided with overpressure protection devices and redundant isolation means. System discharge from overpressure protection devices is collected in the pressurizer relief tank in the Reactor Coolant System. Isolation valves are provided at all auxiliary systems connections to the Reactor Coolant System.

5.2.3 General Material Considerations

5.2.3.1 Material Specifications

The material specifications used for the principal pressure retaining applications in each component comprising the Reactor Coolant System boundary are listed in Table 5-11 for Class 1 Primary Components and Table 5-12 for Class 1 and 2 Auxiliary Components. These materials are procured in accordance with the material specification requirements and include special requirements of the applicable ASME Code Rules.

The welding materials used for joining the ferritic base materials of the reactor coolant boundary, conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18, 5.20, and 5.30. They are tested and qualified to the requirements of ASME Section III rules.

The welding materials used for joining the austenitic stainless steel base materials of the reactor coolant boundary conform to ASME Material Specifications SFA 5.4, 5.9, and 5.30.

They are tested and qualified according to the requirements stipulated in Section 5.2.5 of this safety analysis report.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination of the reactor coolant boundary conform to ASME Material Specifications SFA 5.11, 5.14, and 5.30. They are tested and qualified to the requirements of ASME Section III rules and are used only in procedures which have been qualified to these same rules.

5.2.3.2 Compatibility With Reactor Coolant

Materials used in components within the reactor pressure boundary are listed in Table 5-11, Table 5-12, and Table 5-13. All of the ferritic low alloy and carbon steels which are used in principal pressure retaining applications are provided with corrosion resistant cladding on all surfaces that are exposed to reactor coolant. This cladding material has a chemical analysis which is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy. The other base materials which are used in principal pressure retaining applications which are exposed to the reactor coolant are austenitic stainless steel, nickel-chromium-iron alloy, martensitic stainless steel. Ferritic low alloy and carbon steel nozzles are safe ended with nickel-chromium-iron alloy weld F-Number 43 using weld buttering techniques followed by a post weld heat treatment. The buttering material requires further safe ending with austenitic stainless steel base material after completion of the post weld heat treatment.

The cladding of ferritic type base materials receives a post weld heat treatment.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials are used in the solution anneal heat treat condition. The heat treatments are as required by the material specifications. During subsequent fabrication, these pressure retaining materials are not heated above 800°F other than instantaneously and locally by welding operations. The solution annealed surge line material is subsequently formed by hot bending followed by a resolution annealing heat treatment. Corrosion tests are performed in accordance with ASTM A393 or Practice E of ASTM A262 and the accompanying screening test, Practice A.

Deleted paragraph(s) per 2002 revision.

5.2.3.3 Compatibility With External Insulation and Environmental Atmosphere

In general, all of the materials listed in Table 5-11 and Table 5-12 which are used in principal pressure retaining applications and which are subject to elevated temperature during system operation are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the reactor coolant boundary is specified to be either reflective stainless steel type, mass type, or to be made of compounded materials which yield low leachable chloride and/or fluoride concentrations. Mass type insulation is required to be in compliance with USNRC Regulatory Guide 1.36 to ensure the insulation or components do not create or accelerate corrosion of stainless steel. The compounded materials in the form of block, boards, cloths, tapes, adhesives, cements, etc., are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage or other contamination from the environmental atmosphere. Each lot of insulation material is qualified and analyzed in accordance with procurement specifications to assure that all of the materials provide a compatible combination for the reactor coolant boundary.

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as; valve packing, pump seals, etc., materials which are compatible with the coolant are used. These are shown in Table 5-11 and Table 5-12. Ferritic materials exposed to coolant leakage can be observed as part of the in-service visual and/or nondestructive inspection program to assure the integrity of the component for subsequent service.

5.2.3.4 Chemistry of Reactor Coolant

The Reactor Coolant System chemistry specifications are given in Table 5-14.

The Reactor Coolant System water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

The Chemical and Volume Control System provides a means for adding chemicals to the Reactor Coolant System which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup, and control the oxygen level due to radiolysis during all power operations subsequent to startup. The oxygen content and pH limits for power operations are shown in Table 5-14.

The pH chemical employed is lithium-7 hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/Inconel systems. In addition, lithium is produced in solution from the neutron irradiation of the dissolved boron in the coolant. If lithium needs to be added to the reactor coolant system, lithium hydroxide is introduced into the Reactor Coolant System via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Reactor makeup water is then used to flush the solution to the suction manifold of the charging pumps. The concentration of lithium in the Reactor Coolant system is maintained in a range specified for pH control. If lithium needs to be removed from the reactor coolant system, this can be accomplished by placing a standby mixed bed or cation bed demineralizer in service for a short period of time. Diverting letdown to the recycle holdup tank and making up to the NV system is also a means of reducing lithium if the mixed bed and cation bed demineralizers are not available.

During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the Reactor Coolant System in the same manner as described above for the pH control agent.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the volume control tank such that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A self-contained pressure control valve maintains a minimum pressure in the vapor space of the volume control tank. This can be adjusted to provide the correct equilibrium hydrogen concentration.

Components with stainless steel sensitized in the manner expected during component fabrication and installation will operate satisfactorily under normal chemistry conditions in pressurized water reactor systems, because chlorides, fluorides, and particularly oxygen, are controlled to very low levels.

5.2.3.5 Compliance with Regulatory Guide 1.50

The guidelines established in Regulatory Guide 1.50 are complied with except for paragraph C.2, “maintain preheat until stress relief is performed.”

Quenching or rapid cooling is prevented. When welding has been completed the weldment may be brought to ambient temperature by the method described in PFI Standard ES-19 paragraphs 3.5 and 3.6. If the ambient temperature is below 32°F, preheat is maintained at 50°F minimum until stress relieving has been done. This assures no detrimental effects to the metal and at the same time allows NDE to be performed and the proper installation of post weld stress relief equipment.

For ASME Section III Class 1 components within Westinghouse scope, the Westinghouse practice was in agreement with the recommendation of Regulatory Guide 1.50 except for Regulatory Positions 1(b) and 2.

In the case of Regulatory Position 1(b), the welding procedures were qualified within the preheat temperature ranges required by Section IX of the ASME Code. Westinghouse experience has shown excellent quality of welds using the ASME qualification procedures.

With regard to Regulatory Position 2, Westinghouse did not consider it necessary to maintain the preheat temperature until a post-weld heat treatment had been performed. In the case of large components, code acceptable low-alloy steel welds have been made using Westinghouse specified procedures. In the case of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or post-weld heat treatment has been followed by Westinghouse. In all cases, the welds have shown high integrity.

BWI maintains full compliance with Regulatory Guide 1.50 in S/G fabrication.

5.2.3.6 Compliance with Regulatory Guide 1.71

Duke does not comply with the specific requirements of Regulatory Guide 1.71. Performance qualifications, for personnel welding under conditions of limited accessibility, are conducted and maintained in accordance with the requirements of ASME B&PV Code Sections III and IX. A requalification is required when (1) any of the essential variables of Section IX are changed or (2) when authorized personnel have reason to question the ability of the welder to satisfactorily perform to the applicable requirements. Production welding is monitored for compliance with the procedure parameters and welding qualification requirements are certified in accordance with Sections III and IX. Further assurance of acceptable welds of limited accessibility is afforded by the welding supervisor assigning only the most highly skilled personnel to these tasks. Finally, weld quality, regardless of accessibility, is verified by the performance of the required nondestructive examination.

Westinghouse practice did not require qualification or requalification of welders for areas of limited accessibility as described by Regulatory Guide 1.71. Limited accessibility qualification or requalification, which is additional to ASME Section III and IX requirements, is an unduly restrictive requirement for shop fabrication, where the welders' physical position relative to the welds is controlled and did not present any significant problems. In addition, shop welds of limited accessibility were repetitive due to multiple production of similar components, and such welding was closely supervised. Experience has shown that Westinghouse shop practices produce high quality welds. The performance of required non-destructive evaluations provided further assurance of acceptable weld quality.

BWI maintains full compliance with Regulatory Guide 1.71 in S/G fabrication.

5.2.4 Fracture Toughness

5.2.4.1 Compliance with Code Requirements

Assurance of adequate fracture toughness of the ferritic materials in the Unit 1 Reactor Coolant System pressure boundary is provided by compliance with Section III of the 1971 ASME Boiler and Pressure Vessel Code, plus applicable Addenda and Code Cases to Winter 1971. The location and orientation of the impact specimens for Units 1 and 2 are in accordance with paragraph NB-2300 of ASME Section III. In addition, the reactor vessel materials meet the fracture toughness requirements of 10CFR 50, Appendix G, to the extent possible. The pressure-temperature limitations on reactor operation, as well as leak and hydrostatic test conditions are determined in accordance with Appendix G to Section XI of the ASME B&PV Code (1995 through 1996 Addendum) and Appendix G, 10CFR50. Since the fracture toughness testing performed on vessel material from Units 1 and 2 did not include all of the tests necessary to determine RT_{NDT} in the manner prescribed in NB-2300 of ASME III, Summer 1972 Addenda, the necessary properties were estimated using the procedures provided in Branch Technical Position MTEB 5-2, “Fracture Toughness Requirements for Older Plants.”

A summary of the fracture toughness data for the Unit 1 and Unit 2 reactor pressure vessel material are given in Table 5-15 and Table 5-17.

With regard to fracture toughness, the BWI steam generators are designed in compliance with the requirements of 10 CFR 50, Appendix G, Fracture Toughness Requirements and paragraph NB-2300 or NC-2300 of the ASME Code Section III for primary and secondary ferritic pressure boundary materials. Appropriate test are required to qualify the steam generator for primary and secondary hydrotests at temperatures as low as 70°F.

The BWI steam generators exceed the requirement as actual test results showed RT_{NDT} equal to 0°F by drop weight determination. The subsequent Charpy test results met the 50 ft-lb absorbtion 35 mil lateral expansion criteria of ASME Section III at 60°F.

Additional analysis justifies pressurization of the primary and secondary side of the steam generators at temperatures and pressures per Table 16.10 1-1.

5.2.4.2 Acceptable Fracture Energy Levels

Upper shelf fracture energy levels for the reactor vessel beltline materials (including welds) are determined by Charpy V-notch tests for the vessel irradiation surveillance test programs for Units 1 and 2. Testing on base metal specimens is performed in an orientation transverse to the principal rolling direction (the "weak" orientation). 10 CFR 50, Appendix G requires that the beltline materials demonstrate an unirradiated, or initial, upper shelf energy of no less than 75 ft-lb. It is further required that the beltline materials maintain an upper shelf energy of no less than 50 ft-lb through end-of-life. Should any of beltline material upper shelf energy decrease to a value less than 50 ft-lb before end-of-life, an equivalent margins analysis must be performed.

Charpy V-notch data for unirradiated beltline material specimens are in Table 5-19 through Table 5-29. Upper shelf energy projections through end-of-license are in Table 5-53 and Table 5-54 for Units 1 and 2, respectively.

5.2.4.3 Operating Limitations During Startup and Shutdown

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82 and in accordance with other requirements discussed in Section 5.2.4.1. These properties are then evaluated in accordance with Appendix G of the 1996 Addenda to Section XI of the ASME Boiler and Pressure Vessel Code and methods described in WCAP-14040 Rev 2 (Reference 6) to derive the heatup and cooldown restrictions for the reactor pressure vessel. The calculation of allowable pressure temperature relationships for various temperature heatup and cooldown rates is discussed in detail in WCAP-14040 Rev 2.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2 T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section XI as the reference flaw, amply exceed the current detection capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach (as modified by Code Case N-641) for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IC}, for the metal temperature at that time. K_{IC} is obtained from the reference fracture toughness curve, defined in Appendix A to the ASME Code. The K_{IC} curve is given by the equation:

$$K_{IC} = 33.20 + 20.734 \exp[0.0200(T - RT_{NDT})] \text{ (Equation 1)}$$

Where: K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$CK_{IM} + K_{It} \leq K_{IC} \text{ (Equation 2)}$$

Where:

K_{IM} = is the stress intensity factor caused by membrane (pressure) stress,

K_{It} = is the stress intensity factor caused by the thermal gradients,

K_{IC} = is provided by the code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test conditions during which the reactor core is not critical.

At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, (see discussion below), and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated. These curves define the allowable pressure at the actual indicated temperature as a function of the rate of temperature change. Heatup and cooldown curves for Units 1 and 2 are shown in Technical Specification Figures 3.4.3-1 through 3.4.3-6. Allowances for instrumentation error in the measurements of Reactor Coolant System temperature and pressure are included in the application of the heatup and cooldown curves.

Deleted paragraph(s) per 2003 update.

The RT_{NDT} values are derived using the methods outlined in Regulatory Guide 1.99 Revision 2 "Radiation Embrittlement of Reactor Vessel Materials", in accordance with the guidance in NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations" (Reference 29). The operating curves are calculated using the most limiting value of RT_{NDT} for the reactor vessel at the 1/4 T (thickness of the vessel at the beltline region) and 3/4 T locations. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the pre-service reactor vessel material fracture toughness properties, estimating the radiation-induced shift (ΔRT_{NDT}), and inclusion of an appropriate margin for uncertainties. The values of RT_{NDT} calculated at the 1/4 T and 3/4 T locations are also known as Adjusted Reference Temperature (ART) values.

Generally, the initial RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (T_{NDT}) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F. The detailed procedure for determination of the initial RT_{NDT} is provided in Article NB-2331 of Section III in the ASME Code.

The radiation induced shift, ΔRT_{NDT}, determined by the combined effects of alloy composition (copper and nickel contents), and fast neutron fluence, (E > 1 MeV) per Regulatory Guide 1.99, Revision 2. Data from the reactor vessel surveillance program (RVSP) are used to refine the predictions of the radiation-induced shift, ΔRT_{NDT} and establish appropriate margins for uncertainty associated with this estimate.

The appropriate margin is established per Regulatory Guide 1.99, Revision 2. Margins are lowest when credible surveillance data is available. In cases where the data is deemed non-credible or surveillance data is not available, larger margins are required.

Deleted paragraph per 2003 update.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material, inherently provides additional conservatism for the non-irradiated regions. Therefore, the steam generators, pressurizer, flanges, nozzles, and other regions not significantly affected by radiation are favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

The bases of the current heatup and cooldown curves for the MNS reactor vessels are documented in WCAP-15192, Revision 2 (Reference 30) and WCAP-15201, Revision 2 (Reference 31) for Units 1 and 2, respectively. In the basis documents, the curves were calculated for 34 EFPY. An applicability evaluation has been performed in WCAP-17455 (Reference 32) using updated fluence and materials data. The applicability evaluation concludes that the heatup and cooldown curves are applicable beyond 34 EFPY for McGuire Units 1 and 2. However, the heatup and cooldown curves in the Technical Specifications conservatively remain valid through 34 EFPY. The bases for the heatup and cooldown curves (from References 30 and 31) remain unchanged and are as follows:

Unit	Limiting Material	Loc	Initial RT_{NDT}	ΔRT_{NDT} (34 EFPY)	Margin	ART (34 EFPY)	Estimated Copper
1	Lower Shell Longitudinal Welds (Seams 3-442A and C	¼ T	-50°F	196°F	56°F	**202°F**	0.21%
		¾ T	-50°F	140°F	56°F	**146°F**	0.21%
2	Lower Shell Forging 04	¼ T	-30°F	119°F	34°F	**123°F**	0.15%
		¾ T	-30°F	87°F	34°F	**91°F**	0.15%

5.2.4.4 Compliance with Reactor Vessel Material Surveillance Program Requirements

Changes in fracture toughness of the core region plates, forgings, weldments, and associated heat affected zones due to radiation damage are monitored by a surveillance program which conforms with ASTM E-185-82, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" and 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance". Irradiation effects are quantified through pre-irradiation and post-irradiation testing of Charpy V-notch, and tensile specimens and post-irradiation testing of 1/2T compact tension specimens carried out during the lifetime of the reactor vessel. The surveillance specimens are loaded into capsules that are suspended in baskets attached to the neutron shield pads, which are located near the core mid-height region just outside the core barrel. By being situated closer to the core, the surveillance capsules are exposed to higher neutron flux and acquire fluences that lead the actual vessel fluence The capsules are removed at periodic intervals for testing and analysis. For additional details of the irradiation surveillance program refer to Section 5.4.3.7 and Reference 7.

5.2.4.5 Reactor Vessel Annealing

See Section 5.4.3.8 for a discussion of reactor vessel annealing.

5.2.5 Austenitic Stainless Steel

The unstabilized austenitic stainless steel material specifications used for the (1) Reactor Coolant System Boundary, (2) systems required for reactor shutdown, and (3) systems required for emergency core cooling are listed in Table 5-11 and Table 5-12.

The unstabilized austenitic stainless steel material specifications used for the reactor vessel internals which are required for emergency core cooling for any mode of normal operation or under postulated accident conditions, and for core structural load bearing members are listed in Table 5-13.

All of the above tabulated materials are procured in accordance with the specification requirements and include special requirements of the applicable ASME Code Rules.

5.2.5.1 Cleaning and Contamination Protection Procedures

It is required that all austenitic stainless steel materials used in the fabrication, installation and testing of nuclear steam supply components and systems be handled, protected, stored and cleaned according to recognized and accepted methods and techniques. The rules covering these controls are stipulated in the following Westinghouse Electric Corporation process specifications. These process specifications supplement the equipment specification and purchase order requirements of every individual austenitic stainless steel component or system which Westinghouse procures for a nuclear steam supply system, regardless of the ASME Code Classification. They are also given to Duke for use within their scope of supply and activity.

To assure that manufacturers and installers adhere to the rules in these specifications, surveillance of operations by Westinghouse personnel is conducted either in residence at the manufacturer's plant and the installer's construction site or during periodic engineering and quality assurance visitations and audits at these locations.

The process specifications which establish these rules and which are in compliance with the more current American National Standards Institute N-45 Committee specifications are as follows:

Process Specification Number	
82560HM	Requirements for Pressure Sensitive Tapes for use on Austenitic stainless Steels.
83338KA	Requirements for Thermal Insulation Used on Austenitic Stainless Steel Piping and Equipment.
83860LA	Requirements for Marking of Reactor Plant Components and Piping.
84350HA	Site Receiving Inspection and Storage Requirements for Systems, Material and Equipment.
884351NL	Determination of Surface Chloride and Fluoride on Austenitic Stainless Steel Materials.
85310QA	Packaging and preparing Nuclear Components for Shipment and Storage.
292722	Cleaning and Packaging Requirements of Equipment for Use in the NSSS.
597756	Pressurized Water Reactor Auxiliary Tanks Cleaning Procedures.
597760	Cleanliness Requirements During Storage, Construction, Erection and Start-up Activities of Nuclear Power Systems

With respect to the BWI steam generators compliance with Regulatory Guide 1.37 is applicable only to tubing. The requirements of Regulatory Guide 1.37 are fully imposed on the tubing supplier through the BWI tubing specification with the minor exception that the 1980 edition of ANSI 45.2.1 is used rather than the 1973 edition referenced in the Regulatory Guide.

During fabrication of the steam generators, BWI maintained cleanliness (including loose parts accountability and foreign material exclusion) in accordance with written procedures which as a minimum satisfy the applicable requirements of ASME NQA-2 and ANSI N45.2.1 Cleanliness Class B for primary side surfaces and tube OD and Class C for secondary side surfaces.

5.2.5.2 Solution Heat Treatment Requirements

All of the austenitic stainless steels listed in Table 5-11, Table 5-12 and Table 5-13 are procured from raw material producers in the final heat treated condition required by the respective ASME Code Section II material specification for the particular type or grade of alloy.

5.2.5.3 Material Inspection Program

All of the wrought austenitic stainless steel alloy raw materials which require corrosion testing after the final mill heat treatment are tested in accordance with ASTM A 393 or ASTM A 262 using material test specimens obtained from specimens selected for mechanical testing. The material are obtained in the solution annealed condition.

5.2.5.4 Unstabilized Austenitic Stainless Steels

The unstabilized austenitic stainless steels used in the reactor coolant pressure boundary and components are listed in Table 5-11 and Table 5-12. These materials are used in the as-welded condition as discussed in Section 5.2.5.2. The control of the water chemistry is stipulated in Section 5.2.3.4.

5.2.5.5 Avoidance of Sensitization

The unstabilized austenitic stainless steels used for core structural load bearing members and component parts of the reactor coolant pressure boundary are processed and fabricated using the most practicable and conservative methods and techniques to avoid partial or local severe sensitization.

The following paragraphs describe how the NSSS supplier (Westinghouse) avoided sensitization of austentic stainless steels prior to McGuire receiving its operating license. It is included here for historical reference only. Duke has developed its own program to control the use of sensitized stainless steel as later discussed in this section.

Westinghouse recognizes that the heat affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 900°F to 1600°F. However, severe sensitization, i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion, can still be avoided by control of welding parameters and welding processes. The heat input[1] and associated cooling rate through the carbide precipitation range are of primary importance, as shown by a recent Westinghouse study.

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[1] Heat input is calculated according to the formula: $H = \frac{(E)(I)(60)}{S}$ where: H = joules/in; E = volts; 1 = Amperes; and S = Travel Speed in in./min.

Of 25 production and qualification weldments tested, representing all major welding processes, and a variety of components, and incorporating base metal thicknesses from 0.10" to 4.0", only portions of 2 were severely sensitized. Of these, one involved a heat input of 120,000 joules, and the other involved a heavy socket weld in relatively thin walled material. In both cases, sensitization was caused primarily by high heat inputs relative to the section thickness. However, in only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment. The welding procedure for this joint will be revised and requalified to preclude this condition.

Westinghouse controls the heat input in all austenitic pressure boundary weldments by:

1. prohibiting the use of block welding
2. limiting the maximum interpass temperature of 350°F
3. exercising approval rights on all welding procedures.

To assure that these controls are effective in preventing sensitization, Westinghouse will, if necessary, conduct additional intergranular corrosion tests of qualification mock-ups of primary pressure boundary and core internal component welds, including the following:

Reactor Vessel Safe Ends

Pressurizer Safe Ends

Surge Line and RCP Nozzles

CRDM Head Adaptors

CRDM Seal Welds

Control Rod Extensions

Lower Instrumentation Penetration Tubes.

Primary boundary weldments which do not pass ASTM 393 and/or ASTM 262 Practice E as modified by Westinghouse Process Specification 84201 MW, will be requalified utilizing either low heat inputs or a material substitution.

The Westinghouse position concerning Regulatory guide 1.44, “Control of the Use of Sensitized Stainless Steel,” is based on the fact that unstabilized austenitic stainless steels are subject to intergranular attack (1GA) provided that three conditions are present simultaneously. These are:

1. An aggressive environment, e.g., an acidic aqueous medium containing chlorides or oxygen.
2. A sensitized steel.
3. A high temperature.

If any one of the three conditions described above is not present, intergranular attack will not occur. Since high temperatures cannot be avoided in all components in the Nuclear Steam Supply System, Westinghouse relies on the elimination of conditions 1 and 2 to prevent intergranular attack on wrought stainless steel components.

The water chemistry in the reactor coolant system of a Westinghouse PWR is rigorously controlled to prevent the intrusion of aggressive species. In particular, the maximum permissible oxygen and chloride concentrations were 0.10 ppm and 0.15 ppm respectively. WCAP-7477-L plus addendum and WCAP-7735, “Sensitized Stainless Steel in Westinghouse PWR Nuclear Supply Systems,” describe the precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage. The use of a hydrogen over pressure precludes the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long time exposure of severely sensitized stainless in early plants to PWR coolant environments has

not resulted in any sign of intergranular attack. The WCAP's describe the laboratory experimental findings and the Westinghouse operating experience. The five additional years of operations since the issuing of the WCAP's have provided further confirmation of the earlier conclusions. Severely sensitized stainless steels do not undergo any intergranular attack in Westinghouse PWR coolant environments.

In spite of the fact there never has been any evidence that PWR coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the Nuclear Steam Supply components. Accordingly measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. All wrought austenitic stainless steel stock is purchased in the solution treated and water quenched state. If, during the course of fabrication, the steel is heat treated in the sensitizing temperature range, 900°F to 1600°F, the component is resolution annealed and water quenched. It is generally accepted that these practices will prevent sensitization. Westinghouse has verified this by performing corrosion tests (ASTM 393) on as-received wrought material.

In addition to the above information concerning unstabilized wrought austenitic stainless steels, the following methods and material techniques are used to avoid partial or local severe sensitization in nozzle safe ends.

1. Weld deposit with Inconel (Ni-Cr-Fe weld metal F number 43) then attach safe end after final post weld heat treatment, which was used for the reactor vessel, pressurizer and accumulators.
2. Use of a stainless steel weld metal analysis A-7 containing more than 5 percent ferrite, which was used for the steam generator.

To summarize, Westinghouse has a four point program designed to prevent intergranular attack of austenitic stainless steel components.

1. Control of primary water chemistry to ensure a benign environment.
2. Procurement of raw materials in the solution treated and water quenched condition and the prohibition of subsequent heat treatments in the 900°F to 1600°F temperature range.
3. Control of welding processes and procedures to avoid HAZ sensitization.
4. Confirmation that the welding procedures used for the manufacture of components in the primary pressure boundary and of reactor internals do not result in the sensitization of heat affected zones.

Both operating experience and laboratory experiments in primary water have conclusively demonstrated that this program is 100% effective in preventing intergranular attack in Westinghouse Nuclear Steam Supply Systems utilizing unstabilized austenitic stainless steel.

Duke's program for the control of the use of sensitized stainless steel, in lieu of that outlined in Regulatory Guide 1.44, is set forth below.

All austenitic stainless steels and the fabrication thereof conforms to the requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1971 or later Edition and/or later contractual addenda. All austenitic stainless steel starting materials referred to in Section 5.2.5.2 are procured from raw material suppliers in the final heat treated condition required by the respective ASME Code Section II material specification for the particular type or grade of alloy in accordance with Regulatory Position C.2.

Sensitized stainless steel is defined as those types of unstabilized austenitic stainless steels (typically types 304 and 316) which have been exposed to elevated temperatures (800°F to 1500°F) for extended periods of time. “Severe” sensitization is defined as the condition of a component of highly sensitized (long term exposure to elevated temperatures) material exposed to high stress in a highly corrosive environment.

When highly sensitized stainless steel is to be used in nuclear components, an engineering evaluation of the magnitude of stress, corrosiveness of environment and degree of sensitization is performed to determine if the sensitization is "severe." The degree of sensitization is determined by metallurgical evaluation of material exposed to the same time-temperature variables. Components exposed only to the reactor coolant (such as reactor internals) are not of concern as the corrosiveness of this environment is controlled in accordance with the chemistry requirements of Table 5-14. Weldments which do not receive a sensitizing post weld stress relieve are not of concern as the welding parameters are controlled as stated below.

Control is exercised to prevent excessive exposure of stainless steel to halogens during manufacturing and construction. Components are cleaned in accordance with Duke Energy cleaning procedures. Pickling of highly sensitized stainless steel is avoided. Components are protected and stored in accordance with Duke Energy and/or Vendor specifications.

Weld joints are cleaned before and after welding in accordance with Duke Energy welding specifications.

Weldments are not post weld heat treated within the temperature range of 800°F to 1500°F unless an engineering evaluation of the magnitude of stress, corrosiveness of environment and degree of sensitization is performed.

Voltage and amperage ranges, size and type of welding electrode are specified in welding procedures and monitored to assure compliance. To prevent excessive time of exposure to sensitizing temperatures, a maximum interpass temperature of 350°F is specified and monitored on production weldments.

The design and fabrication of the BWI steam generator is accomplished in full compliance with Regulatory Guide 1.44 as applicable. Sensitized stainless steels are only used in the cladding of the primary head assembly, and gasket and diaphragm seating surfaces.

In cladding applications the sensitized stainless steel material does not serve a pressure retaining function and is L grade material on all wetted primary system surfaces.

In the fabrication of the BWI steam generators, the following requirements are imposed by the Certified Design Specification to prevent IGA on unstabilized austenitic stainless steels:

All austenitic stainless steels are to be procured in the solution annealed condition.

Wrought or cast austenitic stainless steels should not be subjected to fabrication processes or conditions which cause sensitization. If exposure to conditions which cause sensitization are unavoidable, the effects shall be mitigated by:

Specification of a stabilized or low carbon grade of the subject material

AND

Performance of a solution anneal treatment after exposure to conditions conducive to sensitization or performance of ASTM A 262 Practices A and E on coupons of the same material exposed to the same sensitizing conditions to demonstrate the extent of senzitization.

Austenitic stainless steels should not be subjected to manufacturing conditions which result in outer fiber strain on wetted surfaces greater than 2% If these conditions are unavoidable, the effects shall be mitigated by:

Performance of a solution anneal treatment after exposure to conditions which induce greater than 2% strain.

AND

Conduct ASTM A 262 Practice A and E on coupons of the same material exposed to the same conditions to demonstrate that neither the manufacturing process nor the solution anneal treatment results in a sensitization of the material.

All austenitic stainless steel castings shall have a ferrite content of 5 - 20 FN and be solution annealed.

5.2.5.6 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitizing Temperatures

In general, it is not feasible to remove samples from fabricated production components to prepare specimens for retest to determine the susceptibility to intergranular attack. These tests are only performed on test welds when meaningful results would predicate production material performance. No intergranular tests are planned because of satisfactory service experience (see Section 5.2.5.5).

5.2.5.7 Control of Delta Ferrite

Regulatory Guide 1.31, Control of Stainless Steel Welding, describes a method for implementing General Design Criteria 1 of Appendix A to 10CFR Part 50 and Appendix B 10CFR Part 50 with regard to control of welding austenitic stainless steel components and systems. The interim Regulatory position on this guide, March 1974, describes an alternative method of control. The following paragraphs describe the methods to be used and the verification of these methods for austenitic stainless steel welding on this application.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there are insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0 and 3 percent delta ferrite.

The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated or stamped in accordance with ASME B&PV Code, Section III Class 1, 2, and CS components. Delta ferrite control is appropriate for the above welding requirements except where no filler metal is used or for other reasons such control is not applicable. These exceptions include electron beam welding, autogenous gas shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedure and welder qualification accordance with Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5 percent delta ferrite[2] as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, the following examinations are performed in addition to the requirements of Section III.

1. As necessary delta ferrite determination is made for information on an undiluted weld deposit using calibrated magnetic measuring devices conforming to AWS A4.2-7A, "Calibrating Magnetic Instrument to Measure the Delta Ferrite Content of Austenitic Stainless Steel Weld Metal," for the

[2] The equivalent ferrite number may be substituted for percent delta ferrite.

welding procedure qualification records and comparison to the previously calculated delta ferrite value as described above for the "starting" welding material.

2. A visual examination is performed on the procedure-qualification-tested bend specimen using 5 to 10X magnification. In addition to determining the absence of open defects exceeding the rules of Section III, the bend specimen is examined to evaluate whether or not fissure-type discontinuities are discernible in the deposited weld metal. If the latter are discovered and the bend specimen otherwise satisfies the requirements of Section III, the number of fissure type discontinuities discernible per unit of area and the range of their length dimension if reported for information in the procedure qualification records. The bend specimen is rejected when fissure type discontinuities are present to the extent that they result in failure of the tested specimen according to Section III rules.

3. In addition to the essential elements required by Section III, the nonessential elements that determine energy input during welding as described in Section 5.2.5.5 are included in the procedure qualification record.

   The results of all the destructive and non-destructive tests are reported in the procedure qualification record in addition to the information required by Section III.

The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7, type 308 for all applications except type 308L weld metal analysis may be substituted for consumable inserts when used for weld root closures. Bare weld filler metal, including consumable inserts, used in inert gas welding processes conform to ASME SFA-5.9, and are procured to contain not less than 5 percent delta ferrite according to Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA-5.4 or SFA-5.9 and are procured in a wire-flux combination to be capable of providing not less than 5 percent delta ferrite in the deposit according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.

Combinations of approved heats and lots of "starting" welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding materials by lots and heats as appropriate. All of the weld processing is monitored according to approved inspection programs which include review of "starting" materials, qualification records and welding parameters. Welding systems are also subject to quality assurance audit including calibration of gages and instruments; identification of "starting" and completed materials; welder and procedure qualifications; availability and use of approved welding and heat treating procedures; and documentary evidence of compliance with materials, welding parameters and inspection requirements. Fabrication and installation welds are inspected using nondestructive examination methods according to Section III rules.

To further assure the reliability of these controls, Westinghouse has initiated a verification program to last for at least one year. Reference 9 describes the Westinghouse position on control of delta ferrite and the delta ferrite verification program.

The following requirements are used for austenitic stainless steel welding of nuclear safety related systems performed by Duke. These requirements are in lieu of those in Regulatory Guide 1.31.

All austenitic stainless steel welding conforms to the fabrication requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1971 Edition and later contractual addenda. All new welding procedures and welding procedure qualifications conform to the requirements of the ASME, Boiler and Pressure Vessel Code, Section IX, 1971 Edition and later addenda as specified on the welding procedure qualification records.

All bare welding filler material, including consumable inserts, meet the requirements of ASME SFA-5.9 and contain a minimum of 5% delta ferrite, as determined by chemistry. All flux shielded welding filler

material meets the requirements of ASME SFA-5.4 of SFA-5.9 and contains a minimum of 5% delta ferrite. All austenitic stainless steel filler material, used for joining type 304 or 316 base materials, is type 308, 308L, 316 or 316L, none of which exceeds 15% delta ferrite and meets chemical requirements of the applicable specification.

Actual chemical analysis of each heat or lot of filler material is performed and includes the percentage delta ferrite as determined by the "Schaeffler Constitution Diagram for Stainless Steel Weld Metal" or its equivalent. Chemical analysis of bare wire is by undiluted weld deposit or from the wire melt itself. Chemical analysis of flux shielded welding filler material is by undiluted weld deposits only.

A certified chemical test report accompanies each heat or lot of material which is verified to meet the above requirements prior to issuance of the material for welding. This documentation is retained at the job site.

Each lot and heat of filler material is readily identifiable and traceable to the specific joint for which it was used, by actual field documentation.

All new welding procedure qualifications utilize wire meeting the requirements above. Bend specimen is visually examined to determine the ductility and soundness of the test specimen for acceptability by Code standards. Areas which have questionable discontinuities present are inspected by PT and/or magnification to determine acceptability by Code standards.

All welding procedures specify the voltage and amperage ranges for specific welds, and restrict heat input time at temperature, by specifying a maximum of 350°F interpass temperature. Specific welding procedures are identified for each joint to be welded.

All welds are visually inspected for cracks and other unacceptable defects. Welds which have questionable defects are examined under magnification. The root pass of all full penetration welds, not welded against a backing strip, is similarly inspected.

RT and/or PT, as required by the Code, is performed on all ASME Section III Class 1&2 full penetration butt weldments. Microfissuring of the magnitude considered to be detrimental to the structural integrity of weldments is within the sensitivity levels of the NDE methods employed. Weldments with this microfissuring are rejected and treated as other similar types defects in accordance with the Code's acceptance criteria.

Other "in process" weld inspections are performed (such as verification of welding procedure parameters, welder qualification, joint identification, etc.) in accordance with ASME Section III code requirements and additional requirements of the Duke Energy Quality Assurance Procedures.

Control of welding in the BWI steam generators is as follows:

The requirements of Regulatory Guide 1.31 shall be imposed for welding of austenitic stainless steel. All ASME Code welds performed between austenitic stainless steel and ferritic steels or nickel-base alloys shall be performed with ASME II, Part C SFA 5.14 ERNiCR-3 filler metal. Stainless steel filler material used to join austenitic steel to itself shall conform to Regulatory Guide 1.31 with a delta-ferrite requirement for the deposit of δ 5-15 FN. The maximum limit for carbon content in austenitic stainless steel filler material is 0.02%.

5.2.6 Pump Flywheel

The integrity of the reactor coolant pump flywheel is assumed on the basis of the following design and quality assurance procedures.

5.2.6.1 Design Basis

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to produce high energy missiles in the event of structural failure. Conditions which may result in overspeed of the reactor coolant pump increase both the potential for failure and the kinetic energy of the flywheel. Structural integrity of the flywheel is ensured by a range of actions as recommended by NRC Regulatory Guide 1.14 or an NRC-approval alternative as described in Section 5.2.6.3. These actions include conservative stress analyses, inspections and tests.

5.2.6.2 Fabrication and Inspection

The flywheel consists of two plates, approximately five inches and eight inches thick, bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties, such as vacuum-melting, or electroslag remelting. Each plate is fabricated from SA533, Grade B, Class 1 steel. Supplier certification reports are available for all plates and demonstrate the acceptability of the flywheel material on the basis of the requirements of AEC Regulatory Guide 1.14.

Flywheel blanks are flame-cut from the A533 Grade B, Class 1 plates with at least 1/2 inch of stock left on the outer and bore radii for machining to final dimensions. The finished machined bores are subjected to magnetic particle or liquid penetrant examinations. The finished flywheels are subjected to 100 percent volumetric ultrasonic inspection per Paragraphs NB-2532.1 and NB-2532.2 of the ASME Section III Boiler and Pressure Vessel Code.

5.2.6.3 Acceptance Criteria and Compliance with NRC Regulatory Guide 1.14

The reactor coolant pump motor flywheel shall conform to the following material acceptance.

1. The Nil-Ductility Transition Temperature (NDTT) of the flywheel material shall be obtained by two (2) drop weight tests (DWT) which will exhibit “no-break” performance at 20°F in accordance with ASTM E-208. The above drop weight test demonstrate that the NDTT of the flywheel material is no higher than 10°F.

2. A minimum of three (3) Charpy V-notch impact specimens shall be tested at ambient (70°F) temperature in accordance with the specification ASTM-E-23. The Charpy V-notch (Cv) energy in both the parallel and normal orientation with respect to the rolling direction of the flywheel material shall be at least 50 ft-lbs at 70°F to demonstrate compliance with Regulatory Guide 1.14. A lower bound Kl reference curve (see Figure 5-17) has been constructed from dynamic fracture toughness data generated in SA533, Grade B, Class 1 steel. All data points are plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static, together represents a large degree of conservatism. Reference of this curve to the guaranteed Ni1 Ductility Transition Temperature of +10°F indicates that, at the predicted flywheel operating temperature of 110°F, the minimum fracture toughness is in excess of 100 KS1-in$^{1/2}$. This conforms to Regulatory Guide 1.14 requirements that the dynamic stress intensity factor must be at least 100 KS1-in$^{1/2}$.

   Precautionary measures taken to preclude missile formation from primary coolant pump components, assure that the pumps do not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy case.

   The flywheels are tested to a speed of 125% above the normal operating speed of the motor but they are not tested to the anticipated overspeed if a break occurs in the reactor coolant piping in either the

suction or discharge side of the pump. However, the integrity of the flywheel under the worst overspeed condition during a piping break in the reactor coolant system is demonstrated by analysis and model testing. The ductile analysis is performed using the faulted condition criteria (Appendix B) of Section III of the ASME Boiler and Pressure Vessel Code. Compliance with the limits given in the code for the faulted condition assures that the flywheel can withstand the worst overspeed condition with sufficient margin.

Thus, it is concluded that flywheel plate materials are suitable for use and can meet Regulatory Guide 1.14 acceptance criteria on the bases of suppliers certification data.

An inservice inspection program is maintained for the reactor coolant pump flywheels. This program provides for the inspection of each reactor coolant pump flywheel, as stated below, per the actions of Regulatory Guide 1.14 or the recommendations of Westinghouse Topical Report WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination," transmitted by letter dated August 24, 2001. The NRC accepted WCAP-15666 for referencing in license applications in a Safety Evaluation dated May 5, 2003. The acceptability for referencing this topical report in lieu of Positions C.4.b (1) and C.4.b (2) of Regulatory Guide 1.14 was per NRC approval of LAR 223/205 dated August 5, 2004, which revises TS 5.5.7, "Reactor Coolant Pump Flywheel Inspection Program" to a frequency of 20 years.

Twenty Year Inspection Requirement:

In lieu of Position C.4.b (1) and C.4.b (2) of Regulatory Guide 1.14, a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at less than or equal to 20 year intervals.

5.2.7 Reactor Coolant Pressure Boundary Leakage Detection Systems

5.2.7.1 Leakage Detection Methods

Two types of Reactor Coolant System leakage are considered for purposes of leakage detection - (1) leakage to other systems and (2) leakage to the Containment. Functional redundancy is provided by the use of diverse monitoring methods. The sensitivities indicated below are typical for the various types of instruments planned.

Reactor coolant leakage to the Main Steam and Feedwater Systems via steam generator tube leaks is detected by activity monitors located on steam generator steam lines and condenser air ejector effluent lines (refer to Sections 11.4.2.2.14, and 11.4.2.2.2, for discussion on these monitors). Table 5-30 presents information about these detectors and their leak detection sensitivities. Positive indication of secondary system activity is provided in the Control Room. Leakage into the Component Cooling System via the Residual Heat Removal System is detected by activity monitors on the component cooling heat exchangers (see Section 9.2.2.3).

A method of detecting leakage into the Containment is measurement of the Containment floor and equipment sump level. These are small sumps (6' x 6' x 18") located on either side of the containment outside the crane wall. Any leakage would fall to the floor inside the crane wall and run by a sump drain line to one of the two sumps. Any leakage outside the crane wall would fall to the floor and gravity drain to the sumps. The sump level change, measured by a sump level detector, would indicate the leakage rate. This method of detection would indicate in the Control Room a water leak from either the Reactor Coolant System or the Main Steam and Feedwater Systems. The sensitivity of this method is presented in Table 5-30. Figure 5-18 provides a method of determining the systems leakage rate in gpm using the sump level change. Table 5-55 provides a discussion of compliance with Regulatory Guide 1.45.

The environmental conditions during plant power operations and the physical configuration of lower containment will obstruct the total reactor coolant system leakage (including steam) from directly entering the CFAE sump and subsequently, will lengthen the sump's level response time. Therefore, reactor coolant system pressure boundary leakage detection by the CFAE sump will typically occur following other means of leakage detection. Operating experience with high enthalpy primary and secondary water leaks indicates that flashing of high temperature liquid produces steam and hot water mist that is readily absorbed in the Containment air. Much of the hot water that initially hits the containment floor will evaporate in a low relative-humidity environment as it migrates towards a sump. Local low points along the Containment floor provide areas for water to form shallow pools that increase transport time to one or more building sumps. The net effect is only a fraction of any high enthalpy water leakage will eventually collect in a sump and early leak detection may rely on alternate methods.

The incore instrument sump level alarm offers another means of detecting leakage into the containment. This enhances the diversity of the leakage detection function as recommended in Regulatory Guide 1.45. The incore tunnel sump is located under the reactor vessel in the tunnel area, where no leakage is expected under normal conditions. The incore instrument sump is nominally 5 feet x 5 feet x1 foot deep, which corresponds to a capacity of approximately 186 gallons. The setting for the alarm on the plant computer is at sump HI level, which is approximately 11 inches above the sump floor. The incore tunnel sump pump starts at HI level and stops on a LOW Level (approximately 3 inches of water). For an initial condition prior to the development of a primary system leak, it is conservative to assume that the incore sump is empty (i.e., dry) due to evaporation. Once a leak develops, the plant computer provides an alarm in the control room where the sump pump starts at the HI level. This volume of water at the HI level is approximately 172 gallsons. As such, the plant computer will alert the control room operators to a primary system leak of 1 gallon per minute into the incore tunnel sump in less than 4 hours. The incore instrument sump leakage detection system is an exception to positions C2, C5, C7, and C8 of Regulatory Guide 1.45 as described in Table 5-55. However, diversity in leakage detection for the incore tunnel sump is available through the CVUCDT level change and Containment atmosphere radioactivity monitors.

Another method is the Containment particulate air activity monitor (refer to Section 11.4.2.2.4 for discussion of this monitor). The sensitivity of this detector is dependent upon the reactor coolant activity and, therefore, is relatively insensitive to leaks during the initial period of unit operation when the coolant activity is low. Table 5-30 presents further information about the sensitivity of this instrument. Positive indication of leakage is provided in the Control Room.

In a license amendment request dated July 27, 2005, (reference 25) as supplemented (references 26 and 27) Duke clarified the capabilities of the Reactor Coolant System leakage detection instrumentation in regard to Regulatory Guide 1.45 as summarized in Table 5-55. In a letter and Safety Evaluation dated September 30, 2006 (reference 28), the NRC approved this license amendment request.

The Containment particulate radioactivity monitor is only a reliable leakage detection method for Mode 1. Radioactivity can be dispersed into the containment atmosphere from numerous sources, including natural products (primarily radon daughter products), airborne loose surface contamination and reactor coolant leakage. Modes 2, 3 & 4 which are transitory in nature result in changes within containment that typically result in the dispersal of additional concentrations of radioactivity into the containment atmosphere. This additional radioactivity within the containment atmosphere results in effectively masking any collected radioactivity from an active RCS leak, and limits the particulate monitor's ability to adequately detect an RCS leak. In addition, for the particulate monitor to detect RCS leakage, the RCS leakage must flash to steam. However, the temperature of the RCS is reduced for Modes 3 and 4 which results in less dispersal of radioactivity into the containment atmosphere from an active RCS leak. In Mode 1, water activation products build into eqjuilibrium rapidly within the RCS, and provide a significant source term for rapid detection of an RCS leak. In Modes 2, 3 and 4 only corrosion products are available in the RCS for detection of an RCS leak. All of these factors render the particulate monitor limited in its ability to detect

an RCS leak, and detection of a 1 GPM active RCS leak by the particulate monitor cannot be assured until the unit enters Mode 1.

The particulate and gaseous radioactivity monitors obtain their sample from three possible locations, incore area, lower containment and upper containment, which are selectable from the control room. For the purpose of RCS leakage detection, a sample from the lower containment region is required, because the RCS is physically located within the lower containment region. The incore area and lower containment samples are both obtained from the lower containment region. The containment atmosphere particulate monitor collects airborne particulate activity on a fixed filter monitored by a gross beta detector. The collected activity is referenced to as background and is displayed as gross counts per minute (CPM). Background is a combination of collected beta activity from various sources that may include natural decay products, airborne contamination and any small RCS leakage. To detect changes in the containment airborne activity, the particulate monitor utilizes a differential algorithm that calculates an increasing accumulation of particulate activity. The control room readout module displays this increasing accumulation rate as counts per minute accumulating each minute (CMM). The alarm for leakage detection is based upon a positive accumulation rate above background activity on the fix filter.

The Containment particulate alarm setpoint is set as low as practicable, considering the actual concentration of radioactivity in the RCS and the containment background radiation concentration. As low as practicable alarm setpoint is a balance between sufficiently high enough above typical background radiation variations to preclude spurious alarms while sufficiently low enough to assure reasonable sensitivity for early detection of an RCS leak. The alarm setpoint is based upon detected increasing accumulate rate of containment particulate activity above background. Variations in background of containment radiation do occur, and the particulate detector compensates for these changes once the background radiation reaches equilibrium. At the background threshold of collected containment particulate activity that affects detector operability, a failure alarm is actuated for high background on the detector. The alarm setpoint (for detector operability) will be less than or equal to the projected containment activity accumulation rate following a one GPM leak.

In addition several indirect leakage detection methods are employed. Two humidity detectors (one each at the inlets to the upper and lower air handling units) are installed within each containment. The humidity detector system will provide changes in dewpoint or relative humidity.

The Containment gaseous radioactivity monitor has limited sensitivity due to a reduced noble gas source term in the absence of failed fuel, and is considered only as a diverse means of leakage detection method and is not a reliable method for leakage detection.

The ventilation unit condensate drain tank level change offers another means of detecting leakage into the containment. Level change would indicate removal of moisture from the containment by the containment air coolers. The sensitivity of this method is presented in Table 5-30.

Leakage between the double O-ring of the reactor vessel main flange is sensed in the leakoff line by a temperature detector.

Liquid effluent monitors (Refer to Section 11.4.2.1.5 for discussion of these monitors) continuously monitor downstream of each of the two component cooling heat exchangers for activity levels indicative of a reactor coolant leak from the Residual Heat Removal System.

Another indirect method for leakage detection is volume control tank level change, which uses as its basis the makeup demand for the Reactor Coolant System. This method takes into account the reactor coolant pump seal water injection rate, the letdown rate, and the charging rate into the Reactor Coolant System. The leakage detection sensitivity of the volume control tank level sensor is about a factor of two less than that for the sump level method. Measurement of volume control tank level change offers another means of detecting leakage into containment. This enhances the diversity of the leakage detection function as recommended in Regulatory Guide 1.45. The volume tank level detector can be used to differentiate

between Reactor Coolant System and Main Steam and Feedwater Systems leakage because the detector input comes from the former. Information on this method is presented in Table 5-30.

These indirect leakage detection methods are provided as indications and/or alarms to the Control Room, alerting the operators that possible corrective action may be required.

The above detection methods provide information indicative of the integrity of the Reactor Coolant System. This information can be supplemented by laboratory analyses of samples such as Containment sump fluid, Containment air, and steam generator secondary fluid.

Regulator Guide 1.45 is implemented in the design of the Reactor Coolant Pressure Boundary Leakage Detection System with the following clarifications: Regulatory Position C.6, Seismic Qualification, is interpreted as follows: The Leakage Detection System is capable of performing its function following seismic events that do not require plant shut down. The airborne particulate radioactivity monitoring equipment is not seismically qualified to function through the safe shutdown earthquake. The particulate monitor is credited in Mode 1 to meet the requirements of Regulator Guide 1.45.

Position C.6 of Regulatory Guide 1.45 recommends that the Containment air particulate radiation monitor should be designed to remain functional during and following a safe shutdown earthquake (SSE). The basis identified in this position is that it is important for the operators to quickly assess the conditions within the containment following an earthquake comparable to an SSE. In a license amendment request dated March 4, 1996, Duke proposed alternative methods to meet the basis for Position C.6 which include other instrumentation and revised earthquake procedures. These alternative methods include, but are not limited to the following:

- narrow range containment pressure instrumentation,
- wide range containment pressure instrumentation,
- wide range containment sump level instrumentation,
- high range containment radiation monitors, and
- acquisition and analysis of containment atmosphere grab samples.

In addition, an inspection of the plant would be conducted following an earthquake pursuant to the steps in the earthquake response procedures. The conditions of the reactor coolant system would be assessed during a walkdown.

In the Safety Evaluation dated July 30, 1996, the NRC concluded that Duke has demonstrated an acceptable alternative (alternate to seismic Category I) to Position C.6 of Regulatory Guide 1.45 by showing that adequate instrumentation and procedures will be available to assess conditions inside containment following a seismic event comparable to an SSE.

5.2.7.2 Indication in Control Room

Some of the above methods have readouts in the Control Room as noted in Table 5-30 and Table 1-6.

5.2.7.3 Limits For Reactor Coolant Leakage

The maximum allowable leakage rates from identified and unidentified sources are presented in Technical Specifications. The bases for these leakage rates and the criteria for shutdown of the reactor (in the event these rates are exceeded) are also presented in Technical Specifications.

5.2.7.4 Unidentified Leakage

The total leakage from the Reactor Coolant System Pressure Boundary that was not recycled was expected to be on the order to 20 gallons per day per original design. Current practice is to maintain total

leakage from the Reactor Coolant System Pressure Boundary as low as possible. Radiological consequences are discussed in Chapter 11.

An analysis of the pipe cracks related to the size and type of piping used in the Reactor Coolant System is presented in WCAP-7503, Rev. 1, "Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System." This topical report contains mathematical models and experimental data for pipe rupture locations, crack growth, detectable leakage cracks, and critical through-wall cracks. Appendix B to that document presents an analysis and a curve of flow through a crack vs. the ratio of wall thickness and crack length. The length of a crack leaking at a given flow rate for a particular wall thickness and crack width can be determined from this curve for accumulator and primary system piping. Detectable leakage rates and critical through-wall cracks (based on principles of fracture mechanics) are discussed in Sections 4 and 5, respectively, of WCAP-7503, Revision 1.

5.2.7.5 Maximum Allowable Total Leakage

Ratio of the maximum allowable leakage rates to the normal makeup rate and Containment sump pump removal rates are as follows:

$$R_{im} = \frac{Li}{M} = \frac{10gpm}{398gpm} 0.025$$

$$R_{um} = \frac{Lu}{M} = \frac{1gpm}{398gpm} 0.0025$$

Note: This formula was revised during 1998 update.

$$R_{is} = \frac{Li}{S} = \frac{10gpm}{200gpm} 0.05$$

$$R_{us} = \frac{Lu}{S} = \frac{1gpm}{200gpm} 0.005$$

Note: The above equation was revised in the 1998 update.

where

Li = Maximum allowable identified leakage rate, gpm

Lu = Maximum allowable unidentified leakage rate, gpm

M = normal makeup rate, gpm

S = sump pump removal rate, gpm

R_{im} = ratio of identified leakage rate to makeup rate

R_{um} = ratio of unidentified leakage rate to makeup rate

R_{is} = ratio of identified leakage rate to sump removal rate

R_{us} = ratio of unidentified leakage rate to sump removal rate

5.2.7.6 Differentiation Between Identified and Unidentified Leaks

The pressurizer relief tank (PRT) and the reactor coolant drain tank (RCDT) collect reactor coolant pressure boundary (RCPB) leakage from all identifiable sources other than leakage to the Main Steam and Feedwater Systems and the Component Cooling System which is described above.

Figure 5-1 shows the various inputs to the PRT including the pressurizer safety relief valves. Inputs to the RCDT are shown on Figure 11-1 and include the reactor vessel head gasket, reactor coolant pump seals, excess letdown heat exchanger drain, and valve stem leakoff. The criteria for RCPB valve stem leakoff is given in Section 5.5.12.2 thus, these tanks collect all anticipated RCPB leakage not entering another system. Level indication in the PRT and RCDT plus leakage to other systems described in Section 5.2.7.1 above provide the measure of identified leakage. Volume control tank level and pressurizer level along with charging pump flow provide a measure of total system leakage. Containment sump level is a conservative measure of unidentified leakage, since both the Reactor Coolant System and other Containment systems are collected. Subtracting the identified leakage from the total leakage gives a conservative measure of unidentified leakage since leakage from the CVCS system is not differentiable from total system leakage.

5.2.7.7 Sensitivity and Operability Tests

All components used for leakage detection are calibrated, and operational tested before initial use. Many of the detectors (e.g., level detectors and activity monitors) are in frequent use during normal operation, thus verification of their operability is assured. Visual inspections and periodic calibration and maintenance are performed to assure that sensitivities and operability are maintained.

5.2.8 Inservice Inspection Program

Class 1, 2, and 3 components are those components classified as Duke Energy Class A, B, and C, respectively, and are equivalent to Quality Group A, B, and C, respectively, of Regulatory Guide 1.26. Class MC components are metal containment pressure retaining components and their integral attachments, as specified in 10 CFR 50.55a(g). Class 1, 2, 3, and MC components shall be examined in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI in effect as specified in 10 CFR 50.55a(g) to the extent practical. Requests for relief from inservice inspection requirements determined to be impractical will be submitted to the NRC for review in accordance with NRC guidelines for submitting such requests.

Classes 1, 2, and 3 system pressure testing complies with Section XI Article IWB-5000, IWC-5000, and IWD-5000.

Augmented inservice inspection to protect against postulated piping failures will be inspected in accordance with SRG-78-01 (Augmented Inservice Inspection for Pipe Rupture Protection).

5.2.8.1 Provisions for Access to Reactor Coolant Coolant Pressure Boundary

The various components of the reactor coolant system, associated auxiliary systems, and emergency core-cooling systems have been designed with provisions for access as required by Section XI of the ASME Code to the extent practical. The access to specific areas of the reactor vessel are described in 5.4.4.4. The examinations are performed to the extent practical as specified in 10CFR 50.55a(g).

5.2.8.2 Equipment for Inservice Inspections

For all examinations, both remote and manual, specific procedures will be prepared describing the equipment, inspection technique, operator qualification, calibration standards, flaw evaluation and records. These techniques and procedures shall meet the requirements of the Section XI edition in effect as stated in Section 5.2.8. The procedures and equipment used in performing the reactor vessel and nozzle inservice inspections are prepared in accordance with inspection requirements per Code and industry standards as specified in McGuire Nuclear Station Inservice Inspection Plan.

5.2.8.3 Recording and Comparing Data

Examination results will be compared with baseline inspection data and evaluated by NDE inspector level III per McGuire Inservice Inspection procedures. Recording and evaluation methods used are manual or approved electronic media. Data is retained for the service life of the component or system.

5.2.8.4 Reactor Vessel Acceptance Standards

Acceptance standards for evaluation of examination results, including Reactor Vessel examination results, will be in accordance with the edition and addenda of Section XI in effect and as stated in Section 5.2.8. Examinations for which evaluation standards are not in Section XI will be evaluated in accordance with the original construction code.

5.2.8.5 Coordination of Inspection Equipment with Access Provision

Appropriate equipment is specified in inspection specifications and procedures used for specific examination areas of components. Modification of obstructions to provide access to the examination areas will be considered to the extent practical.

5.2.9 References

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5.3 Thermal Hydraulic System Design

5.3.1 Analytical Methods and Data

The thermal and hydraulic design bases of the Reactor Coolant System are described in Sections 4.3 and 4.4 in terms of core heat generation rates, DNBR, analytical models, peaking factors, and others relevant aspects of the reactor.

5.3.2 Operating Restrictions On Pumps

The minimum NPSH and minimum seal injection flow rate must be established before operating the reactor coolant pumps. With the minimum 6 gpm seal injection flow rate established, the operator has to verify that the system pressure satisfies NPSH requirements. NPSH required for the NCP is listed in Table 5-35. The information to make this determination is provided in the operating procedure for the reactor coolant pumps.

5.3.3 Temperature-Power Operating Map

The relationship between Reactor Coolant System temperature and power is shown in Figure 5-19.

The above referenced figure is for general information. Calculational sources should be consulted for actual predicted behavior and/or operational limits. In addition, the figure does not reflect operation under a reduced T-average coastdown scheme.

The effects of reduced core flow due to inoperative pumps is discussed in Sections 5.5, 15.2, and 15.3.

Natural circulation capability of the system is shown in Section 15.2.

5.3.4 Load Following Characteristics

The Reactor Coolant System is designed on the basis of steady state operation at full power heat load. The reactor coolant pumps utilize constant speed drives as designed in Section 5.5 and the reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in Section 7.7. Operation with one pump out of service requires adjustment only in Reactor Protection setpoints as discussed in 7.2.

5.3.5 Transient Effects

Transient effects are evaluated as follows: Complete Loss of Forced Reactor Coolant Flow (15.3), Partial Loss of Forced Reactor Coolant Flow (15.3), Startup of an Inactive Loop (15.4), Loss of Load (15.2), Loss of Normal Feedwater (15.2), Loss of Offsite Power (15.2), and Accidental Depressurization of the Reactor Coolant System (15.2).

5.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal and hydraulic characteristics are given in Sections 4.3 and 4.4.

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5.4 Reactor Vessel and Appurtenances

In support of the McGuire Unit 1 and Unit 2 Measurement Uncertainty Recapture (MUR) power uprate, the reactor vessel material discussions and values presented in affected tables have been re-performed to reflect the uprated power operation level for the reactor core of 3469 MWt (1.02 times the original licensed power level (3411 MWt), minus measurement uncertainty (0.3%)). The updated reactor vessel materials discussions/tables re-performed at 3469 MWt bound the original licensed power operation for the reactor core at 3411 MWt.

Section 5.4 has been divided into four principal sections viz., design bases, description, evaluation and test and inspections for the reactor vessel and its appurtenances consistent with the requirements of the introductory paragraph of Section 5.4 of the Standard Format and Content Guide Revision 1. The following specific information required by the guide is cross referenced below.

Guide Reference	FSAR Section
5.4.1 Protection of Closure Studs	5.4.2.2
5.4.2 Special Processes for Fabrication and Inspection	5.4.2.1, 5.4.4
5.4.3 Features for Improved Reliability	5.4.2.1
5.4.4 Quality Assurance Surveillance	5.4.2, 5.4.4
5.4.5 Materials and Inspections	5.2.3, 5.4.4
5.4.6 Reactor Vessel Design Data	Table 5-32

5.4.1 Design Basis

5.4.1.1 Codes and Specifications

The reactor vessel and closure head are Safety Class 1. Design and fabrication of the vessel were carried out in strict accordance with ASME Section III, Class 1. Material specifications are in accordance with the ASME code requirements and are given in Sections 5.2.3 and 5.2.5.

5.4.1.2 Design Transients

Cyclic loads are introduced by normal power changes, reactor trip, startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Vessel analysis result in a usage factor that is less than 1.0.

With regard to the thermal and pressure transients involved in the loss of coolant accident, the reactor vessel and closure head are analyzed to confirm that the delivery of cold emergency core cooling water to the vessel following a loss of coolant accident does not cause a loss in integrity of the vessel and head.

The design specifications require analysis to prove that the vessel is in compliance with the fatigue limits of Section III of the ASME Boiler and Pressure Vessel code. The loadings and transients specified for the analysis are based on the most severe conditions expected during service.

A control rod housing failure does not cause propagation of failure to adjacent housing or to any other part of the Reactor Coolant system boundary.

Design transients are discussed in detail in Section 5.2.1.5.

5.4.1.3 Protection Against Non-Ductile Failure

Protection against non-ductile failure is discussed in Section 5.2.4.

5.4.1.4 Inspection

The internal surface of the reactor vessel is capable of periodic inspection using visual and/or non-destructive techniques over the accessible areas. During refueling, the vessel cladding is capable of being inspected in certain areas between the closure flange and the primary coolant inlet nozzles, and, if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

The closure head is examined visually during each refueling. Optical devices permit a selective inspection of the cladding, control rod drive mechanism adaptor and the gasket seating surface. The head flange to shell full penetration welded transition section is accessible on the outer surfaces for visual inspection, dye penetrant or magnetic particle, and ultra-sonic testing. The closure studs can be inspected periodically using visual, magnetic particle and/or ultrasonic techniques.

The design of the reactor vessel and appurtenances complies with the requirements of the ASME Section XI, Rules for Inservice Inspection of the Nuclear Reactor Coolant Systems.

5.4.2 Description

The reactor vessel for Unit 1 was fabricated by Combustion Engineering and for Unit 2 was fabricated by De Rotterdamsche Droogdok Mij. N.V. (The Rotterdam Dockyard Company). Both vessels are cylindrical with a welded hemispherical bottom head and a removable, bolted flanged and gasketed, hemispherical upper head. The reactor vessel closure region is sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leakoff communications; one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains a CRDM (See Figure 5-20). This head adaptor is a tubular member, attached by partial penetration welds to the underside of the closure head. The upper end of this adaptor contains acme threads for the assembly of control rod drive mechanisms. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. Inlet and outlet nozzles are spaced evenly around the vessel. Outlet nozzles are located on the vessel to facilitate optimum layout of the Reactor Coolant System equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The bottom head of the vessel contains penetrations for connection and entry of the nuclear in-core instrumentation. Each penetration consists of a tubular member made of either an Inconel or an Inconel-stainless steel composite tube. Each tube is attached to the inside of the bottom head by a partial penetration weld.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 0.156 inch minimum of stainless steel or Inconel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets, except for the upper head which is insulated with thermal wrap blanket and microtherm fully encapsulated stainless steel panels. The insulation is contoured to enclose the top, sides and bottom of the vessel. Provisions for removability of the insulation is made for the portions covering the closure head and bottom head to provide access for inservice inspection.

5.4.2.1 Fabrication Processes

The following paragraphs describe how the NSSS supplier (Westinghouse) avoided using sensitized stainless steel and low alloy welds as a pressure boundary material for the reactor vessel prior to McGuire receiving a operating license. It is included here for historical reference only. Duke Power has their own

programs to control the use of sensitized stainless steel and to provide inservice inspection as discussed in UFSAR Sections 5.2.5.5 and 5.2.8 respectively.

1. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited and has been eliminated by either a select choice of material or by programming the method of assembly. This restriction on the use of sensitized stainless steel has been established to provide the primary system with preferential materials suitable for:
   a. Improved resistance to contaminants during shop fabrication, shipment, construction and operation.
   b. Application in critical areas.
2. Minimum preheat requirements have been established for pressure boundary welds using low alloy weld material. Special preheat requirements have been added for stainless steel cladding of low stressed areas. Preheat must be maintained until post weld heat treatment, except for overlay cladding where it may be lowered to ambient temperature under restrictive conditions. The purpose of placing limitations on preheat requirements is to provide additional precautionary measures that decrease the probabilities of weld cracking by decreasing temperature gradients, lower susceptibility to brittle transformation, prevent hydrogen embrittlement and reduce peak hardness.
3. The threads of the control rod drive mechanism adaptor as well as the surfaces of the guide studs are chrome plated to prevent possible galling of the mated parts.
4. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining beads are Inconel weld metal in order to prevent cracking.
5. The location of full penetration weld seams in the upper closure head and bottom head are restricted to areas that permit accessibility during inservice inspection.

Principal design parameters of the reactor vessel are given in Table 5-32.

5.4.2.2 Compliance with Regulatory Guide 1.65

Regulatory Guide 1.65 was published after procurement of the McGuire Units 1 and 2 reactor vessel bolting material. The bolting material qualification tests were performed per the ASME Section III Code and Addenda in effect at the time of material procurement, which required meeting an average of 35 ft-lbs impact energy with neither lateral expansion or maximum ultimate tensile strength limitation required. The bolting material meets the ASME Code requirements, but in some cases does not conform with regulatory position C.1.b.(2) of Regulatory Guide 1.65 as described below.

Unit 1

The stud bolts for Unit 1 were fabricated from approximately 18 bars of 7½" diameter produced from two heats of SA 540 Grade B24 material. The nuts and washers were made from approximately 7 tubes of 10.723" diameter by 2.224" wall thickness produced from one heat of SA540, Grade B23 material. Tests were performed at 10°F on specimens from each end of 7 bars and 7 tubes as required by the ASME Code. The required three impact tests on each end of the bars and tubes tested showed impact energy values that ranged from a low of 44, 46 and 44 ft-lbs to a high of 52, 52 and 52 ft-lbs for the bars, and from a low of 42,46 and 46 ft-lbs to a high of 54, 56 and 54 for the tubes. Three bars and one tube showed impact values that were less than the minimum 45 ft-lb energy criterion of the guide.

All the bars and tubes tested on Unit 1 met the ultimate tensile strength criterion of Regulatory Guide 1.65.

Unit 2

The stud bolts for Unit 2 were fabricated from 12 bars of 7½" diameter produced from six heats of SA 540, Grade B24 material. The nuts were made from 12 tubes of 10.82" diameter by 2.16" thickness produced from one heat of SA 540, Grade B24 material while the washers were made from 3 tubes of 10.82" diameter by 1.97" thickness made from one heat of SA 540, Grade B24 material. Tests were performed at 10°F on specimens from each end of the 12 bars and showed impact energy ranging from a low of 41, 44 and 44 ft-lb to a high of 58, 60.5 and 60.5 ft-lbs. Three bars did not meet the minimum 45 ft-lb energy imposed by the Guide. The tests results at 10°F for the tubes conformed with the requirements of the Guide.

All bars and tubes tested on Unit 2 met the ultimate tensile strength criterion of Regulatory Guide 1.65.

For the bars and tubes showing 10°F impact data averaging below 45 ft-lbs, the intent of Regulatory Guide 1.65 is met, inasmuch as sufficient fracture toughness is expected at the specified preload temperature or at the lowest service temperature, both of which are significantly above the 10°F Charpy test temperature. Also, the impact energies at the preload or lowest service temperature will be higher than was obtained at the 10°F Charpy test temperature.

The inspection of the Units 1 and 2 reactor vessel bolting material achieves the same purpose as the Regulatory Guide position C.2 as described in the following (also see Table 5-34).

The bolting material was ultrasonicly examined according to an approved Combustion Engineering procedure which requires that:

1. The 100% examination is conducted after heat treatment and prior to threading.
2. The material is scanned in both the radial and axial directions.
3. The calibration for the radial examination is based on a standard back reflection established in an indication-free area of each stud.
4. The calibration for the axial scan is based on a distance corrected reference level established on the responses from 3/8 in. diameter flat bottomed holes in a representative calibration block.
5. For radial testing, material containing discontinuities that produce an indication exceeding 20% of the calibration back reflection amplitude, or that cause a 50% or greater loss in back reflection is unacceptable. For axial testing, material containing a discontinuity or discontinuities producing an indication or indications, equal to or greater than the primary DAC reference line is unacceptable.

The studs and nuts were magnetic particle tested after heat treatment and threading.

The protection of closure studs and stud bolt holes against corrosion nuts meets Regulatory Guide 1.65 position C.3. Protection against the possibility of incurring corrosion effects is assured by:

1. Decrease in level of tensile strength compatible with the requirement of fracture toughness.
2. Design of the reactor vessel studs, nuts, and washers, allowing them to be completely removed during each refueling permitting visual and nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the inservice inspection described in Section 5.4.4. Refueling procedures require that each stud be removed, inspected, and placed in a rack. After the studs are removed, the stud holes in the vessel flange are sealed with a special plug. The studs are lifted and moved to a storage area before the water level is raised in the refueling cavity. Thus, the bolting materials and stud holes should not be exposed to the borated refueling cavity water.
3. Protection of the bolting materials by use of a manganese phosphate surface treatment.

Augmented examination of the reactor vessel stud bolting meets position C.4. The inservice examinations of the reactor vessel stud bolting are performed in accordance with ASME Section XI Code rules. In addition to the Section XI examinations, supplemental surface examinations are performed in accordance with the requirements and acceptance limits of ASME III, paragraphs NB-2545 or NB-2546.

These surface examinations are performed to satisfy the requirements of Regulatory Guide 1.65 and are considered as augmented inspections to the normal ASME Section XI inspections.

5.4.3 Evaluation

Pressurized Thermal Shock Evaluation for License Renewal:

The requirements of 10 CFR 50.61 are to protect against pressurized themal shock transients in pressurized-water reactors. The screening criterion established by §50.61 is 270ºF for plates, forgings, and axial welds. The screening criterion is 300ºF for circumferential welds. According to this regulation, if the calculated RT_{PTS} for the limiting reactor beltline materials is less than the specified screening criterion, then the vessel is acceptable with regard to the risk of vessel failure during postulated pressurized thermal shock transients. The regulations require updating of the pressurized thermal shock assessment upon a request for a change in the expiration date of the facility operating license. The RT_{PTS} calculations are time-limited aging analyses because all six of the criteria contained in 10 CFR 54.3 are met. The RT_{PTS} values have been projected to the end of the period of extended operation using the methods provided in §50.61.

The RT_{PTS} results for all beltline materials are presented in Table 5-51 for McGuire Unit 1 and in Table 5-52 for McGuire Unit 2. All the beltline materials in the McGuire reactor vessels have RT_{PTS} values below the screening criteria of 270ºF for plates, forgings or longitudinal welds and 300ºF for circumferential welds at 54 EFPY. The lower shell plate longitudinal welds 3-442 A, B, and C, using Diablo Canyon Unit 2 surveillance data, are the most limiting material for McGuire Unit 1 with a 54 EFPY PTS value of 203°F. The lower shell forging 04 is the most limiting material for McGuire Unit 2 with a 54 EFPY PTS value of 148°F.

Upper Shelf Energy Evaluation for License Renewal:

Appendix G of 10 CFR Part 50 requires that reactor vessel beltline materials must have a Charpy Upper Shelf Energy (USE) of no less than 75 ft-lb and must maintain a Charpy USE of no less than 50 ft-lb throughout the life of the reactor vessel, unless it is demonstrated, in a manner approved by the Director, Office of Nuclear Reactor Regulation (NRR), that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The USE calculations are time-limited aging analyses because all six of the criteria contained in 10 CFR 54.3 are met. The USE analyses for each vessel have been projected to the end of the period of extended operation using the guidance provided in Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials.

The USE values for McGuire Units 1 and 2 reactor vessel beltline materials at 54 EFPY are presented in Table 5-53 for McGuire Unit 1 and Table 5-54 for McGuire Unit 2. All of the beltline materials in the McGuire reactor vessels have USE above the 50 ft-lb limit. The intermediate shell plate longitudinal welds 2-442 A, B, and C, using surveillance data, are the most limiting material for McGuire Unit 1 with a 54 EFPY USE value of 60.5 ft-lbs. The bottom head ring 03 is the most limiting material for McGuire Unit 2 with a 54 EFPY USE value of greater than 61.8 ft-lbs.

Pressure – Temperature Limits for License Renewal:

Appendix G of 10 CFR Part 50 requires heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program. Normally, the pressure-temperature limits are calculated for several years into the future and remain valid for an established period of time not to exceed the current operating license expiration. For McGuire Unit 1 and Unit 2, the heatup and cooldown limit curves for normal operation at 34 EFPY provide a predicted operating window that is sufficient to conduct heatups and cooldowns. Prior to their

expiration, the current McGuire Units 1 and 2 heatup and cooldown limit curves must be replaced by curves that are valid through 60 years (54 EFPY).

5.4.3.1 Steady State Stresses

Evaluation of steady state stresses is discussed in Section 5.2.1.2.

5.4.3.2 Fatigue Analysis Based on Transient Stresses

Fatigue analysis on transient stresses is discussed in Section 5.2.1.2.

5.4.3.3 Thermal Stresses Due to Gamma Heating

The stresses due to gamma heating in the vessel wall are calculated by the vessel vendor and combined with the other design stresses. They are compared with the code allowable limit for mechanical plus thermal stress intensities to verify that they are acceptable. The gamma stresses are low and thus have a negligible effect on the stress intensity in the vessel.

5.4.3.4 Thermal Stresses Due to Loss Of Coolant Accident

The following paragraphs describe a one-time Westinghouse analysis that was performed to show thermal stress on the reactor vessel resulting from a loss of coolant accident to be within allowable limits at the time the analysis was performed. This analysis was performed prior to McGuire receiving its operating license and is historical in nature. Duke Power has programs in place to ensure integrity of the vessel under all expected modes of operation including all anticipated transients.

In the event of a large loss of coolant accident, the Reactor Coolant System rapidly depressurizes, and the loss of coolant may empty the reactor vessel. If the reactor is at normal operating conditions before the accident, the reactor vessel and closure head temperatures are approximately 550°F. If the unit has been in operation for some time, part of the reactor vessel is irradiated. At an early stage in the depressurization transient, the Emergency Core Cooling System rapidly injects cold coolant into the reactor vessel and closure head. This results in thermal stress in the vessel wall and closure head. To evaluate the effect of the stress, three possible modes of failure are considered in the vessel; ductile yielding, brittle fracture and fatigue.

Ductile Mode - The failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Nuclear Power Plant Components Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at the times during the safety injection transient. The results of the analyses showed that local yielding may occur only in approximately the inner 18 percent of the base metal and in the vessel cladding, complying with the above criterion.

Brittle Mode - The beltline region of the reactor vessel was chosen for analysis because the material adjacent to the centerline of the reactor core is subjected to the highest irradiation level and thus has the lowest end-of-life fracture resistance in the vessel. This analysis is performed assuming the variation effects of water temperature, heat transfer coefficients and fracture toughness as a function of time, temperature and irradiation. Both a local crack effect and a continuous crack effect have been considered with the latter requiring the use of a rigorous finite element axisymmetric code. It is concluded from the analysis that if the NSSS sustains a large loss of coolant accident the integrity of the reactor pressure vessel would be maintained and the unit could be shutdown in an orderly manner.

Fatigue Mode - From the standpoint of fatigue, the in-core instrumentation tube attachment welds to the vessel bottom head is the most sensitive region of the reactor vessel during a loss of coolant accident.

This location has the highest usage factor. The failure criterion used for the failure analysis is that of Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the core allowable usage factor of one. As a worst case assumption, the in-core instrument tubes and attachment penetration welds are considered to be quenched to the cooling water temperature while the vessel wall maintains its initial temperature before the start of the transient. The maximum possible pressure stress during the transient is also taken into account. This method of analysis is quite conservative and yields calculated stresses greater than would actually be experienced. The resulting usage factor for the instrument tube welds considering all the operating transients and including the safety injection transient occurring at the end of the unit life is below 0.2 which compares favorably with the code allowable usage factor of 1.0.

Since the closure head receives insignificant irradiation, it is evaluated in a ductile manner for the loss of the coolant accident. This analysis shows the closure head meets the applicable ASME code allowable limits.

It is concluded from the results of these analyses that the delivery of cold emergency core cooling water to the reactor vessel following a loss of coolant accident, does not cause any loss of integrity of the vessel.

5.4.3.5 Deleted

5.4.3.6 Heatup and Cooldown

Heatup and cooldown requirements for the reactor vessel material are discussed in Section 5.2.4.

5.4.3.7 Reactor Vessel Material Surveillance Program Requirements

The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Technical Specification Figures 3.4.3-1 through 3.4.3-6.

In the surveillance program the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile and 1/2 T (thickness) compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and is in accordance with ASTM-E-185-73, “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,” in all aspects except the factor by which four of the capsules lead the vessel maximum fast neutron exposure. New methods for calculating fast neutron fluence were developed after the reactor vessel internals were designed. These new calculations indicate that capsules are in locations which factors exceed the maximum lead factor 3.0 recommended by ASTM-185-73. However, the factors are within the recommend maximum lead of 5.0 in ASTM-185-82.

The intent of setting a limit on the lead factor is to position capsules as near to the vessel wall as possible. In the design of the vessel internals, the capsules are positioned as near to the vessel wall as possible. The test results from the encapsulated specimens will represent the actual behavior of the material in the vessel, and therefore the evaluation of the effects of radiation on the actual vessel material will not be influenced by the higher lead factor.

The reactor vessel surveillance program uses six specimen capsules. The capsules are located in holder tubes attached to the neutron shield pads and are positioned directly opposite the center portion of the core. Sketches of an elevation and plan view showing the location and dimensional spacings of the capsules with relation to the core, neutron shield pads, vessel and weld seams are shown in Figure 5-21, Figure 5-22 and Figure 5-23. The capsules can be removed when the vessel head is removed and can be

replaced when the internals are removed. The six capsules contain reactor vessel steel specimens oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region of the Unit 1 reactor vessel and specimens oriented both parallel and normal to the major working direction of the limiting core region shell forging of the Unit 2 vessel. Associated weld metal and weld heat affected zone metal specimens are also included in each capsule. The six capsules contain 54 tensile specimens, 360 Charpy V-notch specimens (which include metal and weld heat affected zone material) and 72 CT specimens.

Dosimeters including Ni, Cu, Fe, Co-Al, shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are place in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak tested. Archive vessel material sufficient for at least 2 capsules is kept in storage should the need arise for additional replacement test capsules in the program. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent is made for surveillance material and as deposited weld metal.

Each of the six capsules contains the following specimens:

Material	No. of Charpys	No. of Tensile	No. of CTs
Limiting Plate or Forging Material[1]	15	3	4
Limiting Plate or Forging Material[2]	15	3	4
Weld Metal	15	3	4
Heat Affected Zone	*15	-	-

Note:

1. Specimens oriented parallel to the principal rolling direction of plate for Unit 1 or major working direction for forging for Unit 2.
2. Specimens oriented normal to the principal rolling direction of plate for Unit 1 or major working direction of forging for Unit 2.

The following dosimeters and thermal monitors are included in each of the six capsules:

Dosimeters

Iron
Copper
Nickel
Cobalt-Aluminum (0.15% Co)
Cobalt-Aluminum (Cadmium shielded)
U-238 (Cadmium shielded)
Np-237 (Cadmium shielded)

Thermal Monitors

97.5% Pb, 2.5% Ag (579°F Melting Point)
97.5% Pb, 1.75% Ag, 0.75% Sn (590°F Melting Point)

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition

temperature shift measurements are representative of the vessel at a later time in life. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

The calculated maximum fast neutron (E > 1 MeV) exposures at the vessel wall cladding / base metal interface are approximately 2.57 x 10^{19} n/cm^2 at 54 EFPY for Unit 1 and 2.48 x 10^{19} n/cm^2 at 54 EFPY for Unit 2. The reactor vessel surveillance capsules are located at 56º and 58.5º from the major cardinal axes.

Correlations between the calculations and the measurements on the irradiated samples in the capsules are described in Section 5.4.3.7.1 of this FSAR and have indicated good agreement. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by use of data on all capsules withdrawn. The lead factors and schedule for removal of the capsules for post-irradiation testing are as shown in Table 5-33.

Deleted paragraph(s) per 2002 revision.

5.4.3.7.1 Measurement of the Integrated Fast Neutron Exposure of the Irradiation Specimens

The use of passive neutron sensors such as those included in LWR internal surveillance capsules does not yield a direct measure of the energy-dependent neutron flux at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy- dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the desired exposure rates averaged over the irradiation period and, hence, the time integrated exposures experienced by the sensor sets may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

1 - The measured reaction rate for each sensor,
2 - The energy response of each sensor,
3 - The neutron energy spectrum at the sensor set location,
4 - The physical characteristics of each sensor,
5 - The operating history of the reactor.

Procedures applicable to the evaluation of the neutron sensor sets contained in individual surveillance capsules are described in ASTM Standard E853, "Standard Practice for Analysis and Interpretation of Light Water Reactor Surveillance Results". This umbrella practice relies on, and ties together, the application of several supporting ASTM standard practices, methods, and guides dealing with the general areas of activation measurements, neutron transport calculations, and dosimetry data interpretation.

The determination of individual reaction rates for the sensors comprising the multiple foil neutron dosimeter sets involves laboratory counting procedures, decay corrections to account for the operating history of the reactor, and corrections for competing reactions within the sensor materials. Following withdrawal from the reactor, the specific activity of each of the irradiated radiometric sensors is determined using the latest version of ASTM counting procedures for each reaction of interest. In particular, the following standards are applicable to the radiometric sensors utilized in LWR programs:

E523 Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Copper

E263 Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Iron

E264 Standard Test Method for Measuring Fast Neutron Reaction Rates by Radioactivation of Nickel

E704 Standard Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238

E705 Standard Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237

E481 Standard Test Method for Measuring Neutron Fluence Rate by Radioactivation of Cobalt and Silver

E1005 Standard Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance

E181 Standard General Methods for Detector Calibration and Analysis of Radionuclides

Following sample preparation and weighing, the specific activity of each sensor is determined by means of a high purity germanium, HPGe, gamma spectrometer. In the case of these multiple foil sensor sets, these analyses are performed by direct counting of each of the individual sensors, or, as is sometimes the case with U-238 and Np-237 fission monitors, by direct counting preceded by dissolution and chemical separation of cesium from the sensor.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full-power operation can be determined from the following equation:

$$R = \frac{A}{N_0 F Y \sum \frac{P_j}{P_{\text{ref}}} C_j \left[1 - e^{-\lambda t_j}\right]\left[e^{-\lambda t_{d,j}}\right]}$$

Note: This equation created per 2014 update.

where:

A = Measured specific activity (dps/g)

R = Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus)

N_0 = Number of target element atoms per gram of sensor

F = Weight fraction of the target isotope in the sensor material

Y = Number of product atoms produced per reaction

P_j = Average core power level during irradiation period j (MW)

P_{ref} = Maximum or reference power level of the reactor (MW)

C_j = Calculated ratio of ϕ (E > 1.0 MeV) during irradiation period j to the time-weighted average ϕ (E > 1.0 MeV) over the entire irradiation period

λ = Decay constant of the product isotope (sec^{-1})

t_j = Length of irradiation period j (sec)

t_d = Decay time following irradiation period j (sec)

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the above equation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variation of core power level within any given fuel cycle as well as over multiple fuel cycles. For the sensor sets utilized in surveillance capsule dosimetry programs, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections.

The ratio C_j, which can be calculated for each fuel cycle using the neutron transport methodology described in Section 5.4.3.7.2, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single-cycle irradiation, $C_j = 1.0$. However, for multiple-cycle irradiations, particularly those employing low-leakage fuel management, the additional C_j correction must be utilized in order to provide accurate determinations of the decay-corrected reaction rates for the dosimeter sets contained in the surveillance capsules.

Prior to using the measured reaction rates in dosimetry evaluation procedures, additional corrections are made to the U-238 foil measurements to account for the presence of U-235 impurities in the sensors as well as to address the effects of build-in of plutonium isotopes over the course of the irradiation. These corrections are location- and fluence- dependent and can be derived from the plant-specific transport calculations described in Section 5.4.3.7.2.

In addition to the corrections made for the presence of U-235 in the U-238 fission sensors, corrections are also made to both U-238 and Np-237 sensors to account for gamma ray-induced fission reactions occurring over the course of the irradiation. These photo-fission corrections are, likewise, location-dependent and are based on plant-specific calculations described in Section 5.4.3.7.2.

The derivation of fast neutron exposure rates from a set of measured reaction rates has historically proceeded along one of two avenues. One common method, referred to as the spectrum-averaged cross section approach, employs a calculated neutron energy spectrum at the sensor set locations to determine a spectrum-averaged cross section for each sensor included in the dosimetry set. These calculated spectrum averaged cross-sections are, in turn, used to compute appropriate exposure rates from individual sensors; and, an evaluation of the desired exposure rates characteristic of the irradiation is obtained via an average of the individual sensor results. The uncertainties associated with the exposure rates derived using this approach are usually determined from elementary statistics as the standard deviation of the mean.

The second common approach used in the evaluation of multiple foil dosimetry sets utilizes a least-squares adjustment procedure to produce a best fit of the calculated spectrum at the sensor set location to the measured reaction rates from all sensors. In this methodology, uncertainties in the derived exposure rates are dependent on the resultant fit of the calculated spectrum to the measured data; and include a combination of the uncertainties in measured reaction rates, sensor cross-sections, and the trial spectrum. As in the case of the spectrum-averaged cross section approach, best results are generally achieved when the trial spectrum closely approximates the actual spectrum experienced by the sensors. However, when foil coverage is sufficient, the impact of differences between the trial spectrum and the actual spectrum on derived exposure rates is normally less severe when the adjustment method is employed.

The use of the least-squares adjustment methodology in the evaluation of light water reactor dosimetry is addressed in ASTM Standard E944 "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance". In that guide, the recommended approach to be used in the application of adjustment methods to determine best estimates of neutron exposure parameters and their associated uncertainties is described and a list of several available computer codes capable of performing the adjustment function is provided.

In the overall dosimetry evaluation, these two approaches to sensor set analysis are viewed as complementary. Since the least-squares adjustment approach results in reduced uncertainties in the final exposure estimates, this avenue is considered to be the prime methodology for the determination of exposure rates and associated uncertainties from the sensor set reaction rates. However, evaluations using spectrum-averaged cross sections are also considered as an additional check on the adjustment results as well as an indicator of the appropriateness of the trial spectrum used as input to the adjustment procedure.

In the measurement uncertainty recapture uprate evaluation, the least-squares adjustment method has been used. Least-squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best-estimates for key exposure parameters such as fast flux Iϕ(E > 1.0 MeV)I or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum.

In general, the least-squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{Ri} = \sum \left(\sigma_{ig} \pm \delta_{\sigma ig}\right)\left(\phi_g \pm \delta_{\phi_g}\right)$$

relates a set of measured reaction rates R_i to a single neutron spectrum ϕ_g through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the surveillance capsule dosimetry, the FERRET code (Reference 24) was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters (fast fluence Iϕ(E > 1.0 MeV)I and dpa) along with associated uncertainties.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.
3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor test.

For a given application, the calculated neutron spectrum is obtained from the results of plant-specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. For the current measurement uncertainty recapture uprate application, the calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations described in Section 5.4.3.7.2.

The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross sections and uncertainties were obtained from the SNLRML dosimetry cross-section library (Reference 21). The dosimetry reaction cross sections and uncertainties that are utilized in LWR evaluations comply with ASTM Standard E1018, Application of ASTM Evaluated Cross-Section Data File, Matrix E706 (IIB).

The uncertainties associated with the measured reaction rates, dosimetry cross sections, and calculated neutron spectra are input to the least-squares procedure in the form of variances and covariances. The assisgnment of the input uncertainties also follows the guidance provided in ASTM Standard E944.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks, and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g \, x \, R_{g'} \, x \, P_{gg'}$$

Note: This equation created per 2014 update.

where R_a specifies an overall fractional normalization uncertainty, and the fractional uncertainties R_g and $R_{g'}$ specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta_{\theta e}^{-H}$$

Note: This equation created per 2014 update.

where

$$H = \frac{(g-g')^2}{2\gamma^2}$$

Note: This equation created per 2014 update.

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short range correlations over a group range y (0 specifies the strength of the latter term). The value of δ is 1.0 when $g = g'$ and is 0.0 otherwise.

5.4.3.7.2 Calculation of Integrated Fast Neutron (E > 1.0 MeV) Exposure of the Irradiation Specimens and Reactor Vessel Wall

Discrete ordinates transport calculations are performed on a fuel-cycle-specific basis to determine the neutron and gamma ray environment within the reactor geometry. The specific methods applied have been benchmarked according to the guidelines of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001 (Reference 22) and have been approved by he NRC staff for general application to PWR analysis. A description of the transport methodology along with the SER documenting NRC staff approval of the method and computer codes are provided in Reference 23.

In the application of this methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, a series of two-dimensional plant-specific transport calculations are carried out and then synthesized to generate a three-dimensional neutron flux distribution., ϕ(r,0,z), throughout the geometry of interest using the procedures outlined in Regulatory Guide 1.190. These three-dimensional mappings of the neutron environment are completed for each operating fuel cycle and then integrated to determine the neutron fluence experienced by the surveillance test specimens and the pressure vessel wall. In particular, the three-dimensional synthesized flux is calculated using the following techniques as described in Reference 22.

Thus,

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)] / [\phi(r)]$$

Note: This equation was created on 2014 update.

where ϕ (r,0,z) is the synthesized three-dimensional neutron flux distribution, ϕ(r,0) is the transport solution in r,0 geometry, ϕ(r,z) is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution and ϕ(r) is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,0 two-dimensional calculation.

In the approved analysis methodology, the transport calculations are completed using the DORT discrete ordinates code (Reference 25 and the BUGLE-96 cross-section library (Reference 26. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering is treated with a P_5 legendre expansion, and the angular discretization is modeled with an S_{16} order of angular quadruture.

Energy- and space-dependent core power distributions as well as system operating temperatures are treated on a fuel-cycle-specific basis. The spatial variation of the neutron source is obtained from a burnup-weighted average of the respective power distributions from individual fuel cycles including pinwise gradients for all fuel assemblies located on the periphery of the core. The energy disribution of the source is determined on a fuel-assembly-specific basis and includes the effects of fissioning in both uranium and plutonium isotopes.

The results of the transport calculations are validated on a plant-specific basis by comparison with the results of surveillance capsule dosimetry developed using the procedures described in Section 5.4.3.7.1. These comparisons are used to demonstrate that the plant-specific application is consistent with the uncertainty evaluations provided in WCAP-14040 (Reference 23) and to establish that the 20% uncertainty criterion listed in Regulatory Guide 1.190 is met. These comparisons are not used to modify or bias the results of the transport calculations.

In recognition of the crucial role played by reactor physics computations, ASTM Standard Practice E853 "Analysis and Interpretation of Light-Water Reactor Surveillance Results" requires that the transport methodology used in the performance of these calculations be benchmarked and qualified for application to LWR configurations. These benchmarking and qualification studies are generally based on a series of calculation/measurement comparisons for reactor configurations exhibiting increased levels of complexity. Examples of facilities available for these studies are the PCA benchmark facility, the VENUS benchmark facility, and power reactor surveillance capsule and cavity dosimetry data bases.

The PCA (Pool Critical Assembly) experiments documented in References 9, 10, and 11 provide a well characterized, clean geometry benchmark against which neutron transport techniques may be tested. The nature of the PCA configuration permits the benchmarking of basic discrete ordinates modeling techniques and neutron transport cross-sections in a water/steel environment similar to that observed within a light water power reactor.

The VENUS experiments described in Reference 12 also qualify as a controlled benchmark. However, in contrast to the slab geometry of the PCA, the VENUS core consists of pin-type fuel assemblies arrayed in a fashion designed to simulate the irregular shape of an LWR core. In addition, the VENUS mockup includes cylindrical stainless steel components external to the core. Thus, along with the test of basic nuclear data, comparisons of calculations and measurements for the VENUS facility provide the additional benefit of a verification of the R,Θ modeling approach used in LWR analyses.

Final verification of the analytical approach used in neutron exposure evaluations occurs via direct comparison with measurements obtained from power reactor surveillance capsule and reactor cavity dosimetry data bases. These comparisons define the effects of long-term irradiations with multiple core power distributions as well as provide insight into biases and uncertainties that may exist due to construction and operational variables characteristic of a commercial plant.

The validation of the transport calculational methodology used in support of the measurement uncertainty recapture uprate program for McGuire Units 1 and 2 is provided in WCAP-14040-A (Reference 23).

The Ex-Vessel Neutron Dosimetry Measurement Program at McGuire Unit 1 and 2 provides a verification of fast neutron exposure distribution within the reactor vessel wall beltline region and establishes a mechanism to enable long-term monitoring of this portion of the reactor vessel. This neutron measurement system is located external to the reactor vessel which allows for ease of dosimetry removal and replacement. The program assists in the evaluation of radiation damage of the reactor vessel beltline region by measuring the fluence to this region which can be used to predict the shift in the reference nil ductility transition temperature (RT_{NDT}). When used in conjunction with dosimetry from internal surveillance capsules and with the results of neutron transport calculations, the reactor cavity neutron measurements allow the projection of embrittlement gradients through the reactor vessel wall with minimum uncertainty. Comprehensive sensor sets including radiometric monitors are employed at discrete locations within the reactor cavity to characterize the neutron energy spectrum variations axially and azimuthally over the beltline region of the reactor vessel. In addition, stainless steel gradient chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations.

The reactor cavity neutron dosimetry is installed in the annular air gap between the reactor vessel insulation and the primary concrete shield wall in both Units 1 and 2. The ex-vessel neutron dosimetry consists of aluminum dosimeter capsules connected to and supported by 4 stainless steel bead chains, which are supported by tubular brackets attached to a support bar. The support bar is suspended by 2 support chains that are connected to plates welded to the reactor cavity liner plate. The bead chains are mechanically secured to the concrete wall below the reactor vessel. The ex-vessel neutron dosimetry measures fluence for approximately 1/8 of the vessel wall circumference relative to well-known reactor features. Neutron transport calculations then determine the fluence for all the vessel beltline wall.

5.4.3.8 Capability for Annealing the Reactor Vessel

There are not special design features which would prohibit the onsite annealing of the vessel. If the unlikely need for an annealing operation was required to restore the properties of the vessel material opposite the reactor core because of the neutron irradiation damage, a metal temperature greater than 650°F for a period up to 168 hours would be applied. Various modes of heating may be used depending on the temperature.

The reactor vessel material surveillance program is adequate to accommodate the annealing of the reactor vessel. Sufficient specimens are available to evaluate the effects of the annealing treatment.

5.4.4 Tests and Inspections

The reactor vessel quality assurance program is given in Table 5-34.

5.4.4.1 Ultrasonic Examinations

1. During fabrication angle beam inspection of 100 percent of plate material is performed to detect discontinuities that may be undetected by longitudinal wave examination, in addition to the design code straight beam ultrasonic test.
2. The reactor vessel is examined after hydro-testing to provide a base line map for use as a reference document in relation to later inservice inspections.

5.4.4.2 Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adaptors are inspected by dye penetrant after the first layer of weld metal, after each 1/4 inch of weld metal, and the final surface. Bottom instrumentation tubes are inspected by dye penetrant after each layer of weld metal. Core support block attachment welds are inspected by dye penetrant after first layer of weld metal, and after each 1/2

inch of weld metal. This is required to detect cracks or other defects, lower the weld surface temperatures, cleanliness and prevent microfissures. All austenitic stainless steel clad surfaces are 100 percent dye penetrant tested after the hydrostatic test.

5.4.4.3 Magnetic Particle Examination

1. All surfaces of quenched and tempered materials have the inside diameter inspected prior to cladding and the outside diameter 100 percent inspected after hydro-testing. This serves to detect possible defects resulting from the forming and heat treatment operations.
2. The attachment welds for the vessel supports, lifting lugs and refueling seal ledge are inspected after the first layer of weld metal and after each 1/2 inch of weld thickness. Where welds are back chipped, the areas are inspected prior to welding.

5.4.4.4 Inservice Inspection

The welds in the following areas of the installed irradiated reactor vessel are available for ASME Section XI required inspections:

1. Vessel shell - The inside surface.
2. Primary coolant nozzles - The inside surface.
3. Closure head - The inside and outside surface.

   Bottom head - The outside surface.
4. Closure studs, nuts and washers.
5. Field welds between the reactor vessel, nozzles and the main coolant piping.
6. Vessel flange seal surface.
7. CRDM - Note the exception under Section 5.2.8.6.

The design considerations which have been incorporated into the system design to permit the above inspections are as follows:

1. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.
2. The closure head is stored dry on the reactor operating deck with the insulation capable of being temporarily removed during refueling to facilitate the inspection.
3. All reactor vessel studs, nuts and washers are removed to dry storage during refueling.
4. Removable plugs are provided in the primary shield. The insulation covering the nozzle welds may be removed.
5. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.
6. A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

The reactor vessel presents access problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for the periodic non-destructive tests which are required by the ASME inservice inspection code. These are:

1. Shop ultrasonic examinations are performed on all internally clad surfaces to acceptance and repair standards to assure an adequate cladding bond to allow later ultrasonic testing of the base metal from inside surface. The size of cladding bonding defect allowed is 3/4 of an inch in diameter.
2. The design of the reactor vessel shell in the core is a clean, uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.
3. After the shop hydrostatic testing, selected areas of the reactor vessel are ultrasonic tested and mapped to facilitate the inservice inspection program. Vessel design data are in Table 5-32. Transients and anticipated number of cycles are as indicated in Table 5-49. The vessel fabricator quality surveillance information is in Table 5-34.

5.4.5 References

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5.5 Component and Subsystem Design

5.5.1 Reactor Coolant Pumps

5.5.1.1 Design Bases

The reactor coolant pump ensures an adequate core cooling flow rate and hence sufficient heat transfer, to maintain a DNBR greater than the analysis limit within the parameters of operation. The required net positive suction head is by conservative pump design always less than that available by system design and operation.

Sufficient pump rotation inertia is provided by a flywheel, in conjunction with the impeller and motor assembly, to provide adequate flow during coastdown. This flow following an assumed loss of pump power provides the core with adequate cooling.

The pump is capable of operation without mechanical damage at overspeeds up to and including 125 percent of normal speed.

The reactor coolant pump is shown in Figure 5-24. The reactor coolant pump design parameters are given in Table 5-35.

5.5.1.2 Design Description

The reactor coolant pump is a vertical, single stage, centrifugal, shaft seal pump designed to pump large volumes of main coolant at high temperatures and pressures.

The pump consists of three areas from bottom to top. They are the hydraulics, the shaft seals, and the motor.

1. The hydraulic section consists of an impeller, diffuser, casing, thermal barrier, heat exchanger, lower radial bearing, main flange, motor stand, and pump shaft.
2. The shaft seal section consists of three devices. They are the number 1 controlled leakage, film riding face seal and the number 2 and number 3 rubbing face seals. These seals are contained within the main flange and seal housings.
3. The motor section consists of a vertical solid shaft, squirrel cage induction type motor, an oil lubricated double Kingsbury type thrust bearing, two oil lubricated radial bearings, and a flywheel.

Attached to the bottom of the pump shaft is the impeller. The reactor coolant is drawn up through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is a thermal barrier heat exchanger which limits heat transfer between hot system water and seal injection water on loss of seal injection water flow.

High pressure seal injection water is introduced through the thermal barrier wall between the pump bearing and the thermal barrier heat exchanger. A portion of this water flows upward by the radial bearing and into the seals; the remainder flows down the shaft through the thermal barrier labyrinth and past the cooling coils where it acts as a buffer to prevent NC system water from entering the radial bearing and seal section of the unit. The heat exchanger provides a means of cooling system water to an acceptable level in the event that seal injection flow is lost. The water lubricated journal-type pump bearing, mounted above the thermal barrier heat exchanger, has a self-aligning spherical seat.

The reactor coolant pump motor bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury type bearings. All are oil

lubricated. The lower radial bearing and the thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner.

The motor is an air-cooled, (minimum NEMA) Class B insulated, squirrel cage induction motor. The rotor and stator are of standard construction and are cooled by air. Six resistance temperature detectors are located throughout the stator to sense the winding temperature. The top of the motor consists of a flywheel and an anti-reverse rotation device.

Each reactor coolant pump is equipped with displacement vibration sensors located at the upper motor bearing (flywheel), lower motor bearing, and pump seal locations and seismic vibration sensors on the motor frame. Signals from these sensors are continuously monitored by the vibration monitoring system located in the control room. The amplitude of the vibration signal can be read on the vibration monitoring system. Displacement sensor vibration levels of greater than or equal to 15 mils but less than 20 mils will send an alert alarm to the control room operators. Displacement sensor vibration levels of greater than or equal to 20 mils will send a danger alarm to the control room operators. Seismic sensor vibration levels of greater than or equal to 4.5 mils but less than 5 mils will send an alert alarm to the control room operators. Seismic sensor vibration levels of greater than or equal to 5 mils will send a danger alarm to the control room operators. Vibration data is also collected and stored for spectral analysis biweekly.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings and special parts. Component cooling water is supplied to the two oil coolers on the pump motor and to the pump thermal barrier heat exchanger.

The pump shaft, seal housing, thermal barrier, main flange and motor stand can be removed from the casing as a unit without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The performance characteristic, shown in Figure 5-25, is common to all of the fixed speed mixed flow pumps, and the "knee" at about 45 percent design flow introduces no operational restrictions, since the pumps operate at full speed.

5.5.1.3 Design Evaluation

5.5.1.3.1 Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flow rates. Initial Reactor Coolant System tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial unit operation.

The Reactor Protection System ensures that pump operation is within the assumptions used for loss of coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests were conducted on less than full scale prototype seals as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the number 1 seal ("seal ring") is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The "spring-rate" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the number 1 seal entirely removed (full reactor pressure on the number 2 seal) shows that relatively small leakage rates would be maintained for a short period of time (30 mins. max) even if the number 1 seal fails entirely. The operator is warned of this condition by the increase in number 1 seal leakoff and has time to close this line, and to conduct a safe unit shutdown without significant leakage of reactor coolant to the Containment. Thus, it may be concluded that gross leakage from the pump does not occur, even if the No. 1 seals were to suffer physical damage.

The effect of loss of off-site power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically due to loss of off-site power so that component cooling flow is automatically restored. Seal water injection flow is also restored by the automatic restart of a centrifugal charging pump on diesel power.

The reactor coolant pumps are not required to operate without seal water injection during activation of a Safety Injection Signal (SIS). The centrifugal charging pumps operate during SIS activation, supplying seal injection water in addition to the safety injection flow. The seal water supply line from the charging pumps to the reactor coolant pumps contains no containment isolation valves which close automatically during SIS activation ("S" signal).

5.5.1.3.2 Coastdown Capability

It is important to reactor operation that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a blackout condition, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided in the figures in Section 15.3.

The pump is designed for the design basis earthquake at the site and the integrity of the bearings is described in Section 5.5.1.3.4. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with the safe shutdown earthquake. Core flow transients and figures are provided in Sections 15.3.1 and 15.3.2.

5.5.1.3.3 Flywheel Integrity

Demonstration of integrity of the reactor coolant pump flywheel is discussed in Section 5.2.6.

5.5.1.3.4 Bearing Integrity

The design requirements for the reactor coolant pump bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. To this end, the surface-bearing stresses are held at a very low value, and even under the most severe seismic transients do not begin to approach loads which cannot be adequately carried for short periods of time.

Because there are no established criteria for short time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

Oil levels of the motor bearings are continuously monitored and signal an alarm in the Control Room and may require shutting down of the pump if the low level alarm cannot be cleared in a short period of time. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, is indicated and alarmed in the Control Room as a high bearing temperature. This, again, requires pump motor shutdown. Even if these indications are ignored, and the bearing proceeds to failure,

the low melting point of Babbitt metal on the pad surfaces ensures that no sudden seizure of the bearing occurs. In this event the motor continues to drive, as it has sufficient reserve capacity to operate, even under such conditions. However, it demands excessive currents and at some stage is shut down because of high current demand.

The reactor coolant pump shaft is designed so that its critical speed is well above the operating speed.

5.5.1.3.5 Locked Rotor

It was hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft. This constitutes a loss of coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, as it is still supported on a shaft with two bearings. Flow transients and figures are provided in Section 15.3.3.

There are no credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions are initially high temperature signals from the bearing water temperature detector, and excessive number 1 seal leakoff indications respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shut down for investigation.

5.5.1.3.6 Critical Speed

It is considered desirable to operate below first critical speed, and the reactor coolant pumps are designed in accordance with this philosophy. This results in a shaft design which, even under the most severe postulated transient, gives very low values of actual stress.

5.5.1.3.7 Missile Generation

Each component of the pump is analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

5.5.1.3.8 Pump Cavitation

The minimum net positive suction head required by the reactor coolant pump at running speed is approximately 245 ft. head (approximately 106 psi). In order for the controlled leakage seal to operate correctly it is necessary to have a differential pressure of approximately 200 psi across the seal. This is taken into consideration in the operating instructions. At this pressure the net positive suction head requirement is exceeded, and no limitation on pump operation occurs from this source.

5.5.1.3.9 Pump Overspeed Consideration

For the turbine trips actuated by either the Reactor Protection System or the turbine protection system the reactor coolant pumps are maintained connected to the external network to prevent any pump overspeed condition.

A loss of off-site power resulting in isolation of the generator from the external network could result in an overspeed condition. The turbine control system limits the overspeed to less than 120 percent by actuation of the turbine control and intercept valves. As additional backup, the turbine protection system has redundant and diverse overspeed protection as describe in Section 10.2.2.

5.5.1.3.10 Anti-Reverse Rotation Device

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

As the motor comes to a stop and begins to rotate in the opposite direction, one pawl engages the ratchet plate and the motor and ratchet plate also begin to rotate until stopped by the spring return shock absorbers. The rotor remains in this position until the motor is energized again. After the motor is energized and begins to rotate, the ratchet plate is returned to its original position by the spring return shock absorbers.

When the motor is started, the pawls drag over the ratchet plate until the motor reaches approximately 70 revolutions per minute. At this time, centrifugal forces acting on the pawls are sufficient to overcome the force of gravity and hold the pawls in the running position until the speed falls below the above value. Considerable shop testing and plant experience with the design of these pawls have shown high reliability of operation.

5.5.1.3.11 Shaft Seal Leakage

Leakage along the reactor coolant pump shaft is controlled by three shaft seals arranged in series such that reactor coolant leakage to the Containment is essentially zero. Charging flow is directed to each reactor coolant pump via a seal water injection filter. It enters the pumps through a connection in the thermal barrier flange. The flow is then directed to a cavity between the main flange and thermal barrier where the flow enters the bearing chamber. Here the flow splits and a portion enters the Reactor Coolant System via the thermal barrier cooler cavity. The remainder of the flow flows up the pump shaft (cooling the lower bearing) and leaves the pump via the number 1 seal where its pressure is reduced to that of the volume control tank. Leakoff from the number 1 seal assembly from each pump is piped to a common manifold and then via a seal water filter through a seal water heat exchanger where the temperature is reduced to about that of the volume control tank. Leakage past the number 1 seal provides a constant pressure on the number 2 seal and leakage past the number 2 seal provide constant pressure on the number 3 seal. A standpipe is provided to assure a backpressure of at least 7 feet of water on the number 2 seal and warn of excessive number 2 seal leakage flow to the reactor coolant drain tank; via a second overflow connection.

5.5.1.3.12 Seal Discharge Piping

Discharge pressure from the number one seal is reduced to that of the volume control tank. Water from each pump number one seal is piped to a common manifold, through the seal water return filter and through the seal water heat exchanger where the temperature is reduced to that of the volume control tank. The number 2 and number 3 leak off lines permit normal number 2 and 3 seal leakage to flow to the reactor coolant drain tank.

5.5.1.3.13 Spool Piece

A removable spool piece in the Reactor Coolant Pump shaft facilitates the inspection and maintenance of the pump seal system without breaking any of the fluid, electrical or instrumentation connections to the motor and without removal of the motor. See Figure 5-26.

Thus it serves to reduce unit downtime for pump maintenance, and also to reduce personnel radiation exposure due to the reduced time in the proximity of the primary coolant loop.

5.5.1.3.14 Enclosed Self-Ventilated Motors with Air Coolers

These motors are enclosed and have an integrally mounted air-to-water heat exchanger or air cooler. The ventilating air is recirculated within the motor and cooled by the heat exchanger.

By NEMA standards, this type of enclosure is defined in two parts; the general description of an enclosed machine and the more specific definition of the water-cooled air feature.

An enclosed machine is one so enclosed as to prevent the free exchange of air between the inside and outside of the case but not sufficiently enclosed to be termed air tight.

Incorporating the above description, an enclosed motor with air coolers is defined as an enclosed motor which is cooled by circulating air which in turn is cooled by circulating water through a heat exchanger.

5.5.1.4 Tests and Inspections

Support feet are cast integral with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for usual access to the internal surface of the pump casing. Inservice inspection is discussed in Section 5.2.8.

The reactor coolant pump quality assurance program is given in Table 5-36.

5.5.1.5 Radiological Considerations

Personnel radiological exposure associated with operation of the Reactor Coolant Pumps (RCP) is limited, since the pumps are controlled remotely from the Control Room. The RCPs located inside the Reactor Building crane-wall are normally not accessible during normal power operation due to radiological conditions. Pump maintenance or inspections are performed during refueling outages, utilizing routine radiological controls.

5.5.2 Steam Generator

5.5.2.1 Design Bases

Steam generator design data are given in Table 5-37. The design sustains transient conditions given in Section 5.2.1. Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation, and the bases for the estimates are given in Chapter 11. Rupture of a steam generator tube is discussed in Chapter 15.

The internal moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.25 percent by weight under the following conditions:

1. Steady-state operation up to 125 percent of full load steam flow, with water at the normal operating level for original licensed thermal power (3411 MWt). For operation at 3469 MWt (Measurement Uncertainty Recapture (MUR) power uprate thermal power), resultant moisture carryover does not exceed 0.25 percent by weight (Reference 53).
2. Loading or unloading at a rate of five percent of full power steam flow per minute in the range from 15 percent to 100 percent of full load steam flow.
3. A step load change of ten percent of full power in the range from 15 percent to 100 percent full load steam flow.

Codes and materials requirements of the steam generator are given in Sections 3.2, 5.2.3, and 5.2.5. Also see Section 5.2.1.10.

The steam generator design maximizes integrity against hydrodynamic excitation and vibration failure of the tubes for unit life. Refer to Section 5.5.2.3.5.

The water chemistry in the reactor side is selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces. The water chemistry requirements for the secondary side are discussed in Section 10.4.7.

5.5.2.2 Design Description

The steam generator shown in Figure 5-27 is a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The unit is primarily low alloy steel. The heat transfer tubes and the divider plate are Inconel 690 and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel (304 equivalent). The primary side of the tube sheet is weld clad with Inconel.

Feedwater flows directly into a downcomer section and is mixed with saturated recirc flow before entering the boiler section. Subsequently, water-steam mixture flows upward through the tube bundle and into the steam drum section. Centrifugal moisture separators, located above the tube bundle, remove most of the entrained water from the steam. Steam dryers are employed to increase the steam quality to a minimum of 99.75 percent (0.25 percent moisture). The moisture separators recirculate flow through the annulus formed by the shell and tube bundle wrapper.

The steam generators are provided with two 21" diameter primary manways which allow access to each channel of the primary head and one 21" diameter secondary manway on the steam drum dome to permit access to the steam drum, moisture separation equipment, feedring and top of the tube bundle. Eight 6" diameter handholes are provided at the top (secondary side) of the tubesheet. One 2" inspection port provides access to the tube free lane just above the first two tube supports closest to the tubesheet. From the third support toward the top of the tube bundle each support, except the last or eighth, has two 2" ports positioned on either end of the tube free lane just above the support. A 6" handhole is provided on the transition cone to facilitate inspection of the feedring.

Main feedwater can be introduced to the steam generator through the auxiliary feed nozzle located in the shell of the steam generator by means of valving in a cross connection between main and auxiliary feedwater lines. For certain modes of operation, one typically at low power, which involve the addition of relatively cold feedwater to the steam generator, it is desirable to utilize the auxiliary feed nozzle to preclude the potential thermal hydraulic transients. Such transient may be caused by large temperature differences between saturated water in the steam generator and feedwater. A description of the auxiliary feedwater flow path is given in Section 10.4.7.2.

5.5.2.3 Design Evaluation

The following highlights critical steam generator failure modes and design improvements implemented by BWI in the steam generators to address the problems:

Tube to tubesheet crevice IGA is avoided by selection and control of the tube alloy and the development and implementation of tube expansion tooling and procedures which minimize the crevice at the tubesheet secondary face.

Tube to tubesheet crevice and primary side stress corrosion cracking is avoided by using tube expansion techniques which minimize residual stresses.

Tube sensitization is avoided by stress relieving the pressure boundary of the steam generator, including the primary head to tubesheet weld (but excluding the steam drum to heat exchanger closing seam) prior to tubing the generator. Stress relief of the closing seam weld is performed locally and the tube bundle is insulated to maintain the bundle well below sensitization temperatures.

The tubesheet sludge pile is minimized through achievement of a high circulation ratio in the generator, creating high volume cross flow which is evenly distributed on the tubesheet secondary face, high capacity blowdown capability, water chemistry limits and provision of multiple access ports for sludge lancing.

Tube support crud accumulation and consequent undesirable increases in pressure drop across tube supports is avoided through the use of 'open-flow' lattice grids.

Denting at tube support locations is precluded by open-flow lattice grid supports, line contact between tubes and supports, high circulation flows and selection of 410S tube support material which resists corrosion.

Tube vibration fretting wear at lattice grid and U-bend supports is avoided by maintaining optimum tube to support contact/clearance, installing U-bend supports as the tubing process proceeds, applying conservative analytical predictive techniques and selecting tube support material that resists wear with the Inconel 690 interface.

U-bend cracking of inner row tubes is avoided by use of large minimum radius bends and application of stress relief in the tightest bends.

5.5.2.3.1 Forced Convection

The limiting case for heat transfer capability is the “Nominal 100 Percent Design” case. The steam generator effective heat transfer coefficient is based on the coolant conditions of temperature and flow for this case, and includes a conservative allowance for tube fouling. Adequate tube area is selected to ensure that the full design heat removal rate is achieved.

5.5.2.3.2 Natural Circulation Flow

The steam generators which provide a heat sink are at a higher elevation than the reactor core which is the heat source. Thus natural circulation is assured for the removal of decay heat.

5.5.2.3.3 Tube and Tube Sheet Stress Analyses

Tube and tube sheet stress analyses of the steam generator are discussed in Section 5.2.1.10.

5.5.2.3.4 Corrosion

Pressurized-water reactor (PWR) steam generators have experienced primary side stress corrosion cracking (SCC) in the small-radius U-bends and in the expanded zones of tubes.

Heat treatment of Alloy 690 for optimum SCC resistance involves mill annealing at temperatures sufficient to put all the carbon into solid solution, followed by a thermal treatment to precipitate carbides on the grain boundaries in the tube metal microstructure. Resistance to SCC is greatest when the grain boundaries are well decorated with carbides.

The SCC testing has demonstrated that Alloy 690 is highly resistant to cracking in primary side water environments. Alloy 690 resists SCC as well as or better than Alloy 600 or Alloy 800 in secondary side water environments. Alloy 690 has somewhat greater SCC resistance than Alloy 600 in concentrated caustic environments. Alloy 690 resists pitting and general corrosion as well as or better than Alloy 600 or Alloy 800.

Many tests have been performed which compare the PWSCC behavior of candidate steam generator tubing. These results indicate that Alloy 690 reverse U-Bend specimens do not exhibit PWSCC in the 12,000 hour test.

In statically loaded tube tensile specimens tested in 680°F primary water, Alloy 690 does not exhibit PWSCC after 7,000 hours.

Additional results, which are collected on highly stressed Alloy 690 specimens tested in a variety of pure and primary water environments for times up to 31,000 hours, indicate that Alloy 690 is highly resistant to PWSCC.

In steam tests which are performed in 760°F steam produced from hydrogenated pure water, Alloy 690 displays no PWSCC after exposure times up to 6,000 hours.

The above results indicate that PWSCC of Alloy 690 has not been evidenced in tests reported in open literature.

Thermally treated Alloy 690 is the best choice for steam generator tubing based on the resistance of Alloy 690 to PWSCC and the superior resistance to secondary side SCC, intergranular attack and pitting.

5.5.2.3.5 Flow Induced Vibration

In the design of the BWI steam generators, consideration has been given to the possibility of vibratory failure of tubes due to mechanical or flow induced excitation. This consideration includes detailed analysis of the tube support system.

The primary cause of tube vibratory failure in heat exchanger components is that due to hydrodynamic excitation by the fluid outside the tube. The dominant source of hydrodynamic excitation is fluid cross flow and therefore analyses focus on the two regions where the tube bundle is subject to cross flow. These areas are at the entrance of the downcomer feed to the tube bundle and in the curved tube section of the U-bend.

Analysis of the steam generator tubes indicates the flow velocities to be sufficiently below that which is required for damaging fatigue or impacting vibratory amplitudes. The support system, therefore is deemed adequate to preclude excessive tube motion.

In the analyses, all three known potential flow-induced vibration mechanisms were taken into account: fluid-elastic instability, vortex shedding resonance and random turbulence excitation. Of these mechanisms, fluid-elastic instability is the most significant. As a result, the evaluation io this mechanism was performed with highly conservative analysis parameters drawn from published empirical data bases.

Summarizing the results of analyses and tests of steam generator tubes and various support structures for flow induced vibration, it can be stated that an evaluation of support adequacy has been completed using all published techniques believed to be applicable to heat exchanger tube support design. In addition, the tube support system used is consistent with accepted standards of heat exchanger design utilized throughout the industry (spacing, clearance, etc.). Furthermore, the design techniques are supplemented with a continuing literature search effort to maintain current understanding of the complex mechanism of concern.

Further consideration is given to the possibility of mechanically excited vibration, in which resonance of external forces with tube natural frequencies must be avoided. Evidence indicates that the transmissibility of external forces either through the structure or from fluid within the tubes is negligible and provides little cause for concern.

5.5.2.4 Tests and Inspections

The steam generator quality assurance program is given in Table 5-38.

Radiographic inspection and acceptance standards are in accordance with the requirements of Section III of the ASME code, 1986 Edition.

Liquid penetrant inspection is performed on weld deposited tube sheet cladding, channel head cladding, tube-to-tube sheet weldments, and weld deposit cladding.

Liquid penetrant inspection and acceptance standards are in accordance with the requirements of Section III of the ASME code, 1986 Edition.

The inspections of the Steam Generator Surveillance Program follow the requirements of Technical Specification 5.5.9, "Steam Generator (SG) Program". (Reference Section 18.3.2 Steam Generator Surveillance Program)

Magnetic particle inspection is performed on the tube sheet forging, channel head casting, nozzle forgings, and the following weldments:

1. Nozzle to shell
2. Support brackets
3. Instrument connections (primary and secondary)
4. Temporary attachments after removal
5. All accessible pressure containing welds after hydrostatic test

Magnetic particle inspection and acceptance standard are in accordance with requirements of Section III of the ASME code, 1986 Edition.

An ultrasonic test is performed on the tube sheet forging, tube sheet cladding, secondary shell and head plate and nozzle forgings.

The heat transfer tubing is subjected to eddy current test.

Hydrostatic tests are performed in accordance with Section III of the ASME Code, 1986 Edition.

In addition, the heat transfer tubes are subjected to a hydrostatic test per ASME Section II SB-163 prior to installation into the vessel. The test pressure is 3150 + 140/.

Manways provide access to both the primary and secondary sides.

Inservice inspection of steam generator tubes is discussed in Section 5.2.8.4. Supplementary Steam Generator tube inspection information, included in the Steam Generator Program, can be found in the Technical Specifications.

Deleted paragraph(s) per 2002 revision.

5.5.2.5 Radiological Considerations

The passive design of the steam generators does not present radiological consequences during routine operation. The steam generators located inside the Reactor Building crane-wall are normally not accessible during normal power operation due to radiological conditions. Steam generator maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.3 Reactor Coolant Piping

5.5.3.1 Design Bases

The Reactor Coolant System piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of unit operation or anticipated system interactions. Code and material requirements are provided in Sections 3.2 and 5.2.3 respectively. Section 5.2.5 discusses sensitization and its prevention, cleaning procedures, storage, etc., that prevent stress corrosion cracking.

Materials of construction are specified to minimize corrosion/erosion and ensure compatibility with the operating environment.

The piping in the Reactor Coolant System pressure boundary is Safety Class 1 and is designed and fabricated in accordance with ASME III.

The minimum wall thicknesses of the loop pipe and fittings are not less than that calculated using the ASME III Class 1 formula of Paragraph NB-3641.1(3) with an allowable stress value of 17,550 psi.

The pipe wall thickness for the pressurizer surge lines is Schedule 160.

The minimum pipe bend radius is 5 nominal pipe diameters; ovality does not exceed 6 percent.

All butt welds, nozzle welds, and boss welds are of a full penetration design.

The mechanical properties of representative material heats in the final heat treated condition are determined by test at 650°F design temperature per ASTM E-21 or equivalent. In particular, the hot yield strength, (0.2 percent offset) at 650°F equals or exceeds 19,850 psi.

Processing and minimization of sensitization are discussed in Section 5.2.5.

Flanges conform to ANSI B16.5.

Inservice inspection is discussed in Section 5.2.8.

5.5.3.2 Design Description

Principal design data for the reactor coolant piping are given in Table 5-39.

Pipe and fittings are case, seamless without longitudinal welds and electroslag welds, and comply with the requirements of ASME Section II, Parts A and C, Section III, and Section IX.

The Reactor Coolant System piping is specified in the smallest sizes consistent with system requirements. In general, high fluid velocities are used to reduce piping sizes. This design philosophy results in the reactor inlet and outlet piping diameters given in Table 5-39. The line between the steam generator and the pump suction is larger to reduce pressure drop and improve flow conditions to the pump suction.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. There is not electroslag welding on these components. All smaller piping which comprise part of the Reactor Coolant System boundary, such as the pressurizer surge line, spray and relief line, loop drains and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed at the spray and surge line connections to the pressurizer to provide protection against thermal fatigue.

All piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. Residual heat removal pump suction, which is 45° down from the horizontal centerline. This enables the water level in the Reactor Coolant System to be lowered in the reactor coolant pipe while continuing to operate the Residual Heat Removal System should this be required for maintenance.
2. Loop drain lines and the connection for temporary level measurement of water in the Reactor Coolant System during refueling and maintenance operation.
3. The differential pressure taps for flow measurement, which are downstream of the steam generators on the first 90° elbow.
4. The RTD leg bypass flow taps, which are located at 120° intervals around the hot legs to insure a representative temperature sample.

Penetrations into the coolant flow path are limited to the following:

1. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.
2. The reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.
3. The narrow range detectors are located in RTD wells that extend into the reactor coolant pipes.
4. The wide range temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant pipes.

Signals from the narrow range RTDs are used to compute the RCS ΔT (T_{hot} - T_{cold}) and $T_{avg.}$ The ΔT and $T_{avg.}$ for each loop are indicated on the Main Control Board.

The Reactor Coolant System piping includes those sections of piping interconnecting the reactor vessel, steam generator, and reactor coolant pump. It also includes the following:

1. Charging line and alternate charging line from the cold leg branch connections on the reactor coolant loops 1 & 4 respectively to the second check valve.
2. Letdown line and excess letdown line from the branch connections on reactor coolant loop 3 to the second downstream valve.
3. Pressurizer spray lines from the reactor coolant loops 1 & 2 cold legs to the spray nozzle on the pressurizer vessel.
4. Residual heat removal lines (via the NI system) from the Reactor Coolant loops 2 & 3 hot and loops 1 through 4 cold legs out to the second check valve and from the Reactor Coolant loop 3 hot leg out to the second valve.
5. Safety injection lines from the Reactor Coolant System hot and cold legs out to the second check valve.
6. Accumulator lines from the reactor coolant loop cold legs to the second check valve.
7. Loop fill, loop drain, sample, and instrument lines to or from the reactor coolant loops out to the second valve.
8. Pressurizer surge line from the reactor coolant loop 2 hot leg to the pressurizer vessel inlet nozzle.
9. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.
10. All branch connection nozzles attached to reactor coolant loops.
11. Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the power-operated pressurizer relief valves and pressurizer safety valves.
12. Seal injection water and labyrinth differential pressure lines to or from the reactor coolant pump inside reactor containment out to the second valve.
13. Auxiliary spray line from the pressurizer spray line header out to the second valve.
14. Sample lines from pressurizer to the isolation valve.

Details of the materials of construction and codes used in the fabrication of reactor coolant piping and fittings are discussed in Sections 5.2.3 and 5.2.5.

5.5.3.3 Design Evaluation

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown and seismic loads is discussed in Section 5.2.1.10.

5.5.3.3.1 Material Corrosion/Erosion Evaluation

The water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

An upper limit of about 50 feet per second is specified for internal coolant velocity to avoid the possibility of accelerated erosion. All pressure containing welds out to the second valve that delineates the reactor coolant pressure boundary are available for examination with removable insulation.

Components with stainless steel operate satisfactorily under normal chemistry conditions in pressurized water reactor systems, because chlorides, fluorides, and particularly oxygen, are controlled to very low levels.

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system desired reactor coolant water quality listed in Table 5-14. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and Sampling System which are described in Chapter 9.

5.5.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in Section 5.2.5.

5.5.3.3.3 Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, mercury and lead is prohibited. Permissible thread lubricants are listed in the McGuire Power Chemistry Materials Guide.

Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit of 0.0015 mg $C1/dm^2$ and 0.0015 F/dm^2.

5.5.3.4 Tests and Inspections

The Reactor Coolant System piping quality assurance program is given in Table 5-40.

Radiographic examination is performed throughout 100 percent of the wall volume of each pipe and fitting in accordance with NB-2573 of Section III of the ASME Code for all pipe 27½ inches and larger. All unacceptable defects are either eliminated or repaired in accordance with the requirements of Paragraph NB-2578 and NB-2579 of ASME III.

A liquid penetrant examination is performed on both the entire outside and inside surfaces of each finished fitting in accordance with the procedure of ASME III. Acceptance standards are in accordance with Paragraph NB-2546 of Section III, 1971 Edition.

The pressurizer surge lines conform to SA-376 Type 304, 304N (code case 1423-1), or type 316 with supplementary requirements S2 (transverse tension tests), and S6 (ultrasonic test). The S2 requirements apply to each length of pipe. The S6 requirements apply to 100 percent of the piping wall volume.

The end of pipe sections, branch ends and fittings are machined back to provide a smooth weld transition adjacent to the weld path. Butt welds are ground smooth either at the vendors for pre-fit assemblies or at the site to permit inspection in accordance with ASME Section XI. There are no welds within pipe sections and normally the supplied fittings do not contain welds.

5.5.3.5 Radiological Considerations

During normal operation, personnel radiological exposure associated with operation of the Reactor Coolant system piping is limited. The Reactor Coolant system piping is primarily located inside the Reactor Building crane-wall, and is normally not accessible during normal power operation due to radiological conditions. Reactor coolant system piping maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.4 Main Steam Line Flow Restrictor

5.5.4.1 Design Basis

Each steam generator is provided with 7 flow restrictor venturis assembled into the steam outlet nozzle. The flow restrictors are designed to limit steam flow rate consequent to the unlikely event of a steam line rupture, thereby reducing the cooldown rate of the primary system and limiting stresses of internal steam generator components.

The flow restrictor is designed to minimize unrecovered pressure loss coincident with limiting accident flow rate to an acceptable value.

Although it is not considered to be part of the pressure vessel boundary, the restrictor is constructed of material specified in Section III ASME Code.

5.5.4.2 Design Description

The flow restrictor is an assembly of seven smaller nozzles installed within the steam outlet nozzle of the steam generator. The venturi sleeves are constructed from SA 312-304L material and are retained with a SA 516 GR70 retainer plate. The flow restrictor assembly is attached to the main steam generator outlet nozzle forging by interference fit.

5.5.4.3 Design Evaluation

The equivalent throat diameter of the steam generator outlet is 15.87 inches and the resultant pressure drop through the restrictors at 100 percent steam flow is approximately 2.7 psi. The steam side weld to the outlet nozzle is in compliance with manufacturing and quality control requirements of ASME Code Section III.

5.5.4.4 Tests and Inspections

The restrictors are not a part of the steam system boundary. No tests or inspections of the restrictors are anticipated beyond those performed in the fabricator's shop.

5.5.4.5 Radiological Considerations

The passive design of the main steam flow restrictors does present radiological consequences during routine operation. The flow restrictors located inside the Reactor Building crane-wall are normally not accessible during normal power operation due to radiological conditions. SG flow restrictor maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.5 Main Steam Line Isolation System

Refer to Section 10.3 for a discussion of main steam line isolation.

5.5.6 Reactor Core Isolation Cooling System

This section is not applicable to Pressurized Water Reactors.

5.5.7 Residual Heat Removal System

The Residual Heat Removal System transfer heat from the Reactor Coolant System to the Component Cooling System to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal unit cooldown and maintains this temperature until the unit is started up again.

The Residual Heat Removal System also serves as part of the Emergency Core Cooling System during the injection and recirculation phases of a loss of coolant accident.

As a secondary function, the Residual Heat Removal System also is used to transfer refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of the refueling operations.

As part of the FLEX mitigation strategy in response to NRC Order EA-12-049, the ability to provide makeup from the RWST to the Reactor Coolant System is required following a postulated beyond design basis event. A Residual Heat Removal System piping connection is provided for this capability on the 750' Elevation of the Auxiliary Building.

5.5.7.1 Design Bases

Residual Heat Removal System design parameters are listed in Table 5-41.

The Residual Heat Removal System is designed to remove heat from the Reactor Coolant system during the second phase of unit cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the steam and power conversion systems through the use of the steam generators.

The Residual Heat Removal System is placed in operation approximately four hours after reactor shutdown when the temperature and pressure of the Reactor Coolant System are below 350°F and less than 450 psig, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with component cooling water at design flow and temperature, the Residual Heat Removal System is designed to reduce the temperature of the reactor coolant from 350°F to 200°F within 16 hours. The heat load handled by the Residual Heat Removal System during the cooldown transient includes residual and decay heat from the core and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 20 hours following reactor shutdown from an extended run at full power.

5.5.7.2 Design Description

The Residual Heat Removal System as shown in Figure 5-28 consists of two residual heat exchangers, two residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet line to the Residual Heat Removal System is connected to the hot leg of reactor coolant loop 3, while the return lines are connected to the cold legs of each of the reactor coolant loops. These return lines are also the Emergency Core Cooling System low head injection lines (see Section 6.3). The Residual Heat Removal System may also supply the Reactor Coolant System through the Chemical & Volume Control System Auxiliary Pressurizer Spray Connection.

The Residual Heat Removal System suction line is isolated from the Reactor Coolant system by two motor-operated valves in series, both located inside the Containment. Each discharge line is isolated from the Reactor Coolant System by two check valves located inside the Containment and by a normally open

motor-operated valve located outside the Containment. (The check valves and the motor-operated valve on each discharge line are not part of the Residual Heat Removal System; these valves are shown as part of the Emergency Core Cooling System - see Section 6.3.2.1.

During Residual Heat Removal System operation, reactor coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers, and back to the Reactor Coolant System. The heat is transferred to the component cooling water circulating through the shell side of the residual heat exchangers.

Coincident with operation of the Residual Heat Removal System, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the Chemical and Volume Control System low pressure letdown line for cleanup and/or pressure control. By regulating this letdown flow and the charging flow, the Reactor Coolant System pressure may be controlled when the pressurizer is water-solid. Pressure regulation is necessary to maintain the pressure in the range dictated by the fracture prevention criteria requirements of the reactor vessel and by the number 1 seal differential pressure and net positive suction head requirements of the reactor coolant pumps.

The Reactor Coolant system cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. A line containing a flow control valve bypasses the residual heat exchangers and is used to maintain a constant return flow to the Reactor Coolant System. Instrumentation is provided to monitor Residual Heat Removal System pressure, temperature and total flow.

The Residual Heat Removal can also be used for filling the refueling cavity before refueling. After refueling operations, water is pumped back to the refueling water storage tank until the water level is brought down to the flange of the reactor vessel. The remainder of the water is removed via a drain connection at the bottom of the refueling canal by the reactor coolant drain tank pumps (Liquid Waste Recycle System) or by the refueling water purification pump (Refueling Water System).

When the Residual Heat Removal System is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the Sampling System to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each residual heat removal train between the pump and heat exchanger.

The Residual Heat Removal System functions in conjunction with the high head portion of the Emergency Core Cooling System to provide injection of borated water from the refueling water storage tank into the Reactor Coolant System cold legs during the injection phase following a loss of coolant accident.

In its capacity as the low head portion of the Emergency Core Cooling System, the Residual Heat Removal System provides long-term recirculation capability for core cooling following the injection phase of the loss of coolant accident. This function is accomplished by aligning the Residual Heat Removal System to take fluid from the Containment sump, cool it by circulation through the residual heat exchangers, and supply it to the core directly as well as via the centrifugal charging pumps in the Chemical and Volume Control System and the safety injection pumps in the Emergency Core Cooling System.

The use of the Residual Heat Removal System as part of the Emergency Core Cooling System is more completely described in Section 6.3.

Deleted paragraph(s) per 2002 revision.

The NRC issued Generic Letter 98-02, "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Function While in a Shutdown Condition," on May 28, 1998. This generic letter was issued to alert licensees of a potential to drain down the RCS system when the reactor is in hot shutdown conditions. The McGuire system design has a common ECCS/RHR suction header that

can be connected to the FWST. If the FWST isolation valve is opened in these conditions, there is a potential for hot RCS water to drain to the FWST through the suction header. In addition, this hot water could flash to steam creating steam voiding that could adversely affect operation of the ECCS and RHR pumps. McGuire's administrative controls include engineering controls, training initiatives, scheduling controls, and operating and abnormal procedures that preclude alignments and conditions that would allow an inadvertent draindown event. The generic letter response was transmitted to the NRC in a letter from M.S. Tuckman to the NRC dated November 24, 1998, and the generic letter was closed out for McGuire by NRC letter dated March 27, 2000.

5.5.7.2.1 Component Description

The material used to fabricate Residual Heat Removal System components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion resistant material.

Component codes and classifications are given in Section 3.2, and component parameters are listed in Table 5-42.

Residual Heat Removal Pumps

Two pumps are installed in the Residual Heat Removal System. The pumps are sized to deliver reactor coolant flow through the residual heat exchangers to meet the unit cooldown requirements. The use of two separate residual heat removal trains assures that cooling capacity is only partially lost should one pump become inoperative.

The residual heat removal pumps are protected from overheating and loss of suction flow by miniflow bypass lines that assure flow to the pump suction. A control valve located in each miniflow line is operated based a measurement of pump discharge flow. Setpoints are chosen to ensure that the valves open before discharge flow falls below 500 gpm and close after flow rises above 1000 gpm.

A pressure sensor in each pump discharge header provides a signal for an indicator in the Control Room. A high pressure alarm is also actuated by the pressure sensor.

The two pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water existing twenty hours after reactor shutdown when the temperature difference between the two systems is small.

The installation of two heat exchangers in separate residual heat removal trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

Residual Heat Removal System Valves

Typically, valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

Relief valve ND3 (NC Loop 3 Disch. to ND System Safety Relief) has a minimum relief capacity of 900 GPM at a set pressure of 450 psig. The other 3 relief valves are supplied to protect the system from leakage through check valves from the Reactor Coolant System.

5.5.7.2.2 System Operation

Reactor Startup

Generally, while at cold shutdown condition, decay heat from the reactor core is being removed from the Reactor Coolant System by the Residual Heat Removal System. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At initiation of the unit startup, the Reactor Coolant System is completely filled. The Residual Heat Removal System is operating and is connected to the Chemical and Volume Control System via the low pressure letdown line to control reactor coolant pressure. The Residual Heat Removal System may supply flow to the pressurizer via the Auxiliary Pressurizer Spray line to equalize temperature. During this time, the Residual Heat Removal System acts as an alternate letdown path. The manual valves downstream of the residual heat exchangers leading to the letdown line of the Chemical and Volume Control System are opened. The pressure control valve in the line from the Residual Heat Removal System to the letdown line of the Chemical and Volume Control System is then manually adjusted in the Control Room to permit letdown flow. Failure of any of the valves in the line from the Residual Heat Removal System to the Chemical and Volume Control System has no safety implications, either during startup or cooldown.

After the reactor coolant pumps are started, the residual heat removal pumps are stopped, but pressure control via the Residual Heat Removal System and the low pressure letdown line is continued until the pressurizer steam bubble is formed. The pressurizer heaters are energized and indication of steam bubble formation is provided in the Control Room by pressurizer level indication. the Residual Heat Removal System is then isolated from the Reactor Coolant System, and the system pressure is controlled by normal letdown and the pressurizer spray and pressurizer heaters.

Power Generation and Hot Standby Operation

During power generation and hot standby operation, the Residual Heat Removal System is not in service but is aligned for operation as part of the Emergency Core Cooling System.

Reactor Shutdown

The initial phase of reactor cooldown is accomplished by transferring heat from the Reactor Coolant System to the Main Steam System through the use of the steam generators.

When the reactor coolant temperature and pressure are reduced to below 350°F and less than 450 psig, approximately four hours after reactor shutdown, the second phase of cooldown starts with the Residual Heat Removal System being placed in operation.

Startup of the Residual Heat Removal System includes a warmup period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the residual heat exchangers. By adjusting the control valves downstream of the residual heat exchangers the mixed mean temperature of the return flows is controlled. Coincident with the adjustment of flow through the heat exchanger, the heat exchanger bypass valve is regulated to give the required total flow.

The reactor cooldown rate is limited by Reactor Coolant System equipment cooling rates based on allowable stress limits, as well as the operating temperature limits of the Component Cooling System. As the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchangers is increased by adjusting the control valve in each heat exchanger's tube-side outlet line.

As cooldown continues, the pressurizer is filled with water and the Reactor Coolant System is operated in the water solid condition.

At this stage, pressure control is accomplished by regulating the charging flow rate and the rate of letdown from the Residual Heat Removal System to the Chemical and Volume Control System.

After the reactor coolant pressure is reduced and the temperature is 140°F or lower, the Reactor Coolant System may be opened for refueling or maintenance.

Provisions to avoid isolation of ND suction during shutdown modes (i.e., guarding against inadvertent closure of *ND-1B or *ND-2AC) are administered by procedure. After reaching Mode 5 during shutdown, breakers for both *ND-1B and *ND-2AC are opened to ensure inadvertent valve actuation is avoided. Provisions against ND suction line failure are provided by procedural means and interlocks (*ND-1B and *ND-2AC can only be opened if NC pressure is below nominally 385 psig). The ND suction piping design pressure limit is 450 psig, and relief valve *ND-3 provides protection with a lift setpoint of nominally 450 psig.

Refueling

Either residual heat removal pump can be utilized during refueling to pump borated water from the refueling water storage tank to the refueling cavity. One means of filling the refueling cavity is to pump refueling water from the FWST into the reactor vessel through the normal Residual Heat Removal System return lines and into the refueling cavity through the open reactor vessel.

During refueling, the Residual Heat Removal System is maintained in service with the number of pumps and heat exchangers in operation as required by the heat load. Residual Heat Removal System flow rates can also be adjusted as required by the heat load. With reduced water levels during refueling, such as occur during mid-loop operation, there is a potential for vortex formation at the connection of the Residual Heat Removal pump suction line to the Reactor Coolant System loop 3 hot leg. Vortexing has the potential for air entrainment which could result in inaccurate level instrumentation and could adversely affect the capabilities of removing decay heat from the reactor, as identified in NRC Generic Letter 87-12 (Reference 51). To reduce the likelihood for vortexing, Residual Heat Removal System flow rate requirements during refueling are reduced to 1000 gpm, provided that Reactor Coolant System temperatures can be maintained at or below 140°F. Higher flow rates may be required directly after shutdown to accommodate higher decay heat rates, but 1000 gpm will be sufficient for most of the refueling outage to maintain adequate cooling and to prevent boron stratification in the event of a boron dilution incident. (Reference 52).

Following refueling, the residual heat removal pumps can be used to drain the refueling cavity to the top of the reactor vessel flange by pumping the water from the Reactor Coolant System to the refueling water storage tank. The remainder of the water is removed via a drain connection at the bottom of the refueling canal by the reactor coolant drain tank pumps (Liquid Waste Recycle System) or by the refueling water purification pump (Refueling Water System).

5.5.7.3 Design Evaluation

5.5.7.3.1 System Availability and Reliability

The system is provided with two residual heat removal pumps and two residual heat exchangers arranged in separate flow paths. If one of the two pumps or one of the two heat exchangers is not operable, safe cooldown of the unit is not compromised; however, the time required for cooldown is extended.

The time required to cool the RCS from the hot standby condition to the cold shutdown condition (200°F) with one RHR train, assuming design values of component cooling water and service water temperatures, is less than 34 hours.

The two separate flow paths provide redundant capability of meeting the safeguards function of the Residual Heat Removal System. The loss of one Residual Heat Removal System flow path would not negate the capability of the Emergency Core Cooling System since the two flow paths provide full redundancy for safeguards requirements.

To assure reliability, the two residual heat removal pumps are connected to separate electrical buses so that each pump receives power from a different source. If a total loss of off-site power occurs while the system is in service, each bus is automatically transferred to a separate emergency diesel power supply. A prolonged loss of off-site power would not adversely affect the operation of the Residual Heat Removal System.

In response to Generic Letter 2008-01 "Managing Gas Accumulation in ECCS, Decay Heat Removal, and Containment Spray Systems", an evaluation concluded that system procedures and design are adequate to maintain the RHR system sufficiently full of water to ensure operability. Inadequate system fill and vent can result in pump cavitation, pump gas binding, or water hammer. The complete Duke response can be viewed via Reference 43.

5.5.7.3.2 Leakage Provisions and Activity Release

In the event of a loss of coolant accident, fission products may be recirculated through part of the Residual Heat Removal System exterior to the Containment. If the residual heat removal pump seal should fail, the water would spill out on the floor in a shielded compartment. Each pump is located in a separate, shielded room. If one of the rooms is flooded, this would have no effect on the other since there are no interconnections. In addition, in each room provisions are made for draining spillage into a sump which is provided with dual pumps and suitable level instrumentation so that the spillage can be pumped to the Liquid Waste Recycle System.

5.5.7.3.3 Overpressurization Protection

The inlet line to the Residual Heat Removal System is equipped with a pressure relief valve sized to relieve the combined nominal flow of all the centifugal charging pumps at the relief valve set pressure.

Each discharge line from the Residual Heat Removal System to the Reactor Coolant System is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the Residual Heat Removal System from the Reactor Coolant System.

The design of the Residual Heat Removal System includes two isolation valves in series on the inlet line between the high pressure Reactor Coolant System and the lower pressure Residual Heat Removal System. Each isolation valve is interlocked with one of the two independent Reactor Coolant System pressure signals. The interlocks prevent the valves from being opened when Reactor Coolant System pressure is greater than approximately 385.5 psig. If the valves are in the open or intermediate position, the interlocks actuate an alarm when the Reactor Coolant System pressure increases above 440 psig. The alarm notifies the operator that double barrier isolation between the Reactor Coolant System and the Residual Heal Removal System is not being maintained. These interlocks are described in more detail in Section 7.4.1.5.

5.5.7.3.4 Dual Function

The Emergency Core Cooling System (ECCS) function performed by the Residual Heat Removal System is not compromised by its Residual Heat Removal (RHR) function. The valves associated with the Residual Heat Removal System are normally aligned to allow immediate use of this system in its ECCS mode of operation. The system has been designed in such a manner that two redundant flow circuits are available, assuring the availability of at least one train for these purposes.

The RHR function of the Residual Heat Removal System is accomplished through a suction line arrangement which is independent of any ECCS function. The RHR return lines are arranged in parallel redundant circuits and are utilized also as the low head safety injection lines to the Reactor Coolant System. Utilization of the same return circuits for ECCS as well as for RHR lends assurance to the proper functioning of these lines for ECCS purposes.

5.5.7.3.5 Radiological Considerations

The highest radiation levels experienced by the Residual Heat Removal System are those which would result from a loss of coolant accident. Following a loss of coolant accident, the Residual Heat Removal System is used as part of the Emergency Core Cooling System. During the recirculation phase of emergency core cooling, the Residual Heat Removal System is designed to operate for up to a year pumping water from the Containment sump, cooling it, and returning it to the Containment to cool the core.

Since, except for some valves and piping, the Residual Heat Removal System is located outside the Containment, most of the system is not subjected to the high levels of radioactivity in the Containment post-accident environment.

The operation of the Residual Heat Removal System does not involve a radiation hazard for the operators since the system is controlled remotely from the Control Room. If maintenance of the system is necessary, the portion of system requiring maintenance is isolated by remotely operated valves and/or manual valves. The isolated piping is drained and flushed before maintenance is performed.

5.5.7.4 Tests and Inspections

Periodic visual inspections and preventive maintenance are conducted during unit operation according to normal industrial practice.

The instrumentation channels for the residual heat removal pump flow instrumentation devices are calibrated regularly per the McGuire Preventive Maintenance Program and are used during each refueling operation.

Due to the role the Residual Heat Removal has in sharing components with the Emergency Core Cooling System, the residual heat removal pumps are tested as a part of the Emergency Core Cooling System testing program (see Section 6.3.4).

5.5.8 Reactor Coolant Cleanup System

The Chemical and Volume Control System provides reactor coolant cleanup and is discussed in Chapter 9. Personnel radiological exposure associated with operation of the Chemical Volume and Control System (CVCS) is limited, since the system is normally controlled remotely from the Control Room. In the event system maintenance is required, the system has provisions for draining and flushing to reduce activity levels. Portions of the Chemical Volume and Control System (CVCS) inside the Reactor Building crane-wall are normally not accessible during normal power operation due to radiological conditions. CVCS components routinely expected to have high activity (e.g. letdown filters, demineralizers) are typically located within shielded pits or rooms.

Portions of the CVCS system which support Emergency Core Cooling System functions would experience the highest radiation levels subsequent to a loss of coolant accident ECCS sump recirculation operation.

5.5.9 Main Steam Line and Feedwater Piping

Refer to Sections 10.3 and 10.4.7 for a discussion of main steam line and feedwater piping.

5.5.10 Pressurizer

5.5.10.1 Design Bases

The general configuration of the pressurizer is shown in Figure 5-29. The design data of the pressurizer are given in Table 5-43. Codes and material requirements are provided in Sections 3.2 and 5.2.3.

5.5.10.1.1 Pressurizer Surge Line

The surge line is sized to limit the pressure drop between the Reactor Coolant System and the safety valves with maximum allowable discharge flow from the safety valves. Overpressure of the Reactor Coolant System does not exceed 110 percent of the design pressure.

The surge line and connection at each end are designed to withstand the thermal stresses resulting from volume surges which occur during operation.

The pressurizer surge line nozzle diameter is given in Table 5-43.

In response to NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, analyses were performed to confirm the adequacy of the existing surge line piping. These analyses provided McGuire specific data to augment the results obtained from the Westinghouse Owners Group Pressurizer Surge Line Thermal Stratification Generic Detailed Analysis (WCAP-12639).

The following applicability analyses were conducted: a specific review of operating records to ensure that system ΔT limits assumed in WCAP-12639 were not exceeded, a verification of operational methods to ensure that they were consistent with the methods assumed in WCAP-12639 (Limits on system ΔT for future operation are recommended), and a verification of applicability of seismic OBE bending moments used in the fatigue analysis and combined deadweight and OBE moments at the hot leg nozzle.

The following McGuire specific evaluations were performed: an evaluation of the adequacy of pipe support(s) for loads and displacements, an evaluation of the effects of stratification on stress and fatigue at integral welded attachments (lugs, plates, etc.), and an evaluation of the effects of stratification on stress and fatigue of the pressurizer nozzle.

In addition to the applicability and plant specific evaluations, the following was also evaluated: the new maximum pipe movements against available rupture restraint gaps, the effect of stratified movements on rupture restraint blowdown loads; and the effect of stratification on postulatetd break locations.

The results of all of the above analyses confirmed the adequacy of the existing design for McGuire. See references 40, 41 and 42 for detailed discussions of the analyses and the results.

5.5.10.1.2 Pressurizer

The volume of the pressurizer is equal to, or greater than, the minimum volume of steam, water, or total of the two which satisfies all of the following requirements:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.
2. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of ten percent at full power.
3. The steam volume is large enough to accommodate the surge resulting from the design step load reduction of load with reactor control and steam dump without the water level reaching the high level reactor trip point.
4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip.

5. The pressurizer does not empty following reactor and turbine trip.
6. The emergency core cooling signal is not activated during reactor trip and turbine trip.

5.5.10.2 Design Description

5.5.10.2.1 Pressurizer Surge Line

The pressurizer surge line connects the pressurizer to one reactor hot leg. The line enables continuous coolant volume pressure adjustments between the Reactor Coolant System and the Pressurizer.

5.5.10.2.2 Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant.

The surge line nozzle and removable electric heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge line nozzle. A screen at the surge line nozzle and baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/ water interface and assist mixing.

Spray line nozzles, relief and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves also can be operated manually by a switch in the Control Room.

A small continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge from the pressurizer, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the Reactor Coolant System, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power-operated relief valves for normal design transients. Heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop.

The reactor primary system (pressurizer) is designed to accept a load reduction of 10% without operation of the PORVs located on the pressurizer. A 40% steam dump to condenser to ameliorate the consequences of a step load reduction of 50% which effectively translates into a 10% load reduction for the primary system in the initial phase of the pressure transient.

Material specifications are provided in Table 5-11 for the pressurizer and the surge line. Design transients for the components of the Reactor Coolant system are discussed in Section 5.2.1.5. Additional details on the pressurizer design cycle analysis are given in Section 5.5.10.3.5.

Pressurizer Support

The skirt type support is attached to the lower head and extends for a full 360° around the vessel. The lower part of the skirt terminates in a bolting flange with bolt holes for securing the vessel to its foundation. The skirt type support is provided with ventilation holes around its upper perimeter to assure free convection of ambient air past the heater plug connector ends for cooling.

Pressurizer Instrumentation

Refer to Chapter 7 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

Power Sources For Pressurizer Equipment

Pressurizer is equipped with two groups of 416 Kw pressurizer heaters (nominal, initial capacity) each supplied from the redundant 600 VAC essential auxiliary power system, one heater group per power train. Power is available to each heater via offsite power system or from emergency power system. Each heater group has the capability to maintain natural circulation under hot standby conditions. Pressurizer heaters are automatically shed from the emergency power system upon actuation of a safety injection actuation signal. The electrical portions of the PORVs, and pressurizer level indications are provided power by the Vital Instrumentation and Control Power System. The PORV Block valves are provided power from the 600 VAC essential Auxiliary Power System. All these circuits are safety related (Reference 5).

Spray Line Temperatures

Temperatures in the spray lines from two loops are measured and indicated. Alarms from these signals are actuated by low spray water temperature. Alarm conditions indicate insufficient flow in the spray lines.

Safety and Relief Valve Discharge Temperatures

Temperatures in the pressurizer safety and relief valve discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage through the associated valve.

5.5.10.3 Design Evaluation

5.5.10.3.1 System Pressure

Whenever a steam bubble is present within the pressurizer, Reactor Coolant System pressure is controlled by the pressurizer spray, pressurizer heaters and normal letdown flow. Analyses indicate that proper control of pressure is maintained for the operating conditions.

A safety limit has been set to ensure that the Reactor Coolant System pressure does not exceed the maximum transient value allowed under the ASME code, Section III (1971 Edition), and thereby assure continued integrity of the Reactor Coolant System boundary.

Evaluation of unit conditions of operation which follow indicate that this safety limit is not reached.

During startup and shutdown, the rate of temperature change is controlled by the operator. When the reactor core is shutdown, the maximum heatup by pump energy is limited. The installed pressurizer electrical heating capacity provides additional controlled heatup energy.

When the pressurizer is filled with water, i.e., near the end of the second phase of unit cooldown and during initial system heatup, Reactor Coolant System pressure is controlled by regulating letdown flow to the Chemical and Volume Control System.

5.5.10.3.2 Pressurizer Performance

The pressurizer has a minimum free internal volume. The normal operating water volume at full load conditions is 60 percent of the free internal vessel volume. Under part load conditions, the water volume in the vessel is reduced for proportional reductions in unit load to 25 percent of free vessel volume at zero (0) power level. The various operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in Table 5-43.

5.5.10.3.3 Pressure Setpoints

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valves setpoints, and the protection system setpoint pressures are listed in Table 5-10. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics.

5.5.10.3.4 Pressurizer Spray

Two separate, automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open, and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the power relief valves during a step reduction in power level of ten percent of full load.

The pressurizer spray lines and valves are large enough to provide adequate spray using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force. The spray valves and spray line connections are arranged so that the spray will operate when one reactor coolant pump is not operating. The line may also be used to assist in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flow path from the Chemical and Volume Control System and the Residual Heat Removal System to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown if the reactor coolant pumps are not operating. This spray may be supplied by either the Chemical & Volume Control System or the Residual Heat Removal System. The thermal sleeves on the pressurizer spray connection and the spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

5.5.10.3.5 Pressurizer Design Analysis

The occurrences for pressurizer design cycle analysis are defined as follows:

1. The temperature in the pressurizer vessel is always, for design purposes, assumed to equal saturation temperature for the existing Reactor Coolant System pressure, except in the pressurizer steam space subsequent to a pressure increase. In this case, the temperature of the steam space exceeds the saturation temperature since an isentropic compression of the steam is assumed.

   The only exception of the above occurs when the pressurizer is filled solid during startup and cooldown.

2. The temperature shock on the spray nozzle is assumed to equal the temperature of the nozzle minus the cold leg temperature and the temperature shock on the surge nozzle is assumed to equal the pressurizer water space temperature minus the hot leg temperature.

3. Pressurizer spray is assumed to be initiated instantaneously to its design value as soon as the Reactor Coolant System pressure increases 40 psi above the nominal operating pressure. Spray is assumed to be terminated as soon as the Reactor Coolant System pressure falls 40 psig below the normal operating pressure.

4. Unless otherwise noted, pressurizer spray is assumed to be initiated once per occurrence of each transient condition. The pressurizer surge nozzle is also assumed to be subject to one temperature transient per transient condition, unless otherwise noted.

5. At the end of each transient, except the faulted conditions, the Reactor Coolant System is assumed to return to a load condition consistent with the plant heatup transient.

6. Temperature changes occurring as a result of pressurizer spray are assumed to be instantaneous. Temperature changes occurring on the surge nozzle are also assumed to be instantaneous.

7. Whenever spray is initiated in the pressurizer, the pressurizer water level is assumed to be at the no load level.

5.5.10.4 Tests and Inspections

The pressurizer is designed and constructed in accordance with ASME Section III, 1971 Edition.

To implement the requirements of ASME Section XI, the following welds are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The path is ground smooth for ultrasonic inspection.

1. Support skirt to the pressurizer lower head.
2. Surge nozzle to the lower head.
3. Nozzles to the safety, relief, and spray lines.
4. Nozzle-to-safe-end attachment welds.
5. All girth and longitudinal full penetration welds.
6. Manway attachment welds.

The liner within the safe end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer quality assurance program is given in Table 5-44.

5.5.10.5 Radiological Considerations

During normal operation, personnel radiological exposure associated with operation of the Pressurizer is limited, as the system is remotely operated from the Control Room. The Pressurizer is located inside the Reactor Building crane-wall, and is normally not accessible during normal power operation due to radiological conditions. Any required Pressurizer maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.11 Pressurizer Relief Tank

5.5.11.1 Design Bases

Design data for the pressurizer relief tank are given in Table 5-45. Codes and material of the tank are given in Sections 3.2 and 5.2.3.

The tank design is based on the requirement to absorb the pressurizer discharge during a step load decrease of 10 percent. This is equivalent to a discharge of pressure steam equal to 110 percent of the volume above the full power pressurizer water level setpoint. The tank is not designed to accept a continuous discharge from the pressurizer. The volume of water in the tank is capable of absorbing the heat from the assumed discharge, assuming an initial temperature of 120°F and increasing to a final temperature of 200°F. If the temperature in the tank rises above 120°F during unit operation, the tank is cooled by spraying in cool water and draining out the warm mixture, or by circulating the water through the reactor coolant drain tank heat exchanger. The spray rate is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

5.5.11.2 Design Description

The pressurizer relief tank condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from specific relief valves located inside or outside the Containment is also piped to the relief tank. The tank normally contains water and a predominantly nitrogen atmosphere.

Steam is discharged through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. A flanged nozzle is provided on the tank for the pressurizer discharge line connection.

5.5.11.2.1 Pressurizer Relief Tank Pressure

The pressurizer relief tank pressure transmitter provides pressurizer relief tank pressure indication and a high pressure alarm in the control room.

5.5.11.2.2 Pressurizer Relief Tank Level

The pressurizer relief tank level transmitter supplies a signal for an indicator with high and low level alarms.

5.5.11.2.3 Pressurizer Relief Tank Water Temperature

The temperature of the water in the pressurizer relief tank is indicated, and an alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

5.5.11.3 Design Evaluation

The volume of water in the tank is capable of absorbing heat from the pressurizer discharge during a loss of load from full power without a turbine trip scram. Water temperature in the tank is maintained at the nominal Containment temperature.

The rupture discs on the relief tank have a relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure is twice the calculated pressure resulting from the maximum design safety valve discharge described above. The tank and rupture discs holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

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

5.5.11.4 Radiological Considerations

During normal operation, personnel radiological exposure associated with operation of the Pressurized Relief Tank is limited, as the system is remotely operated from the Control Room. The Pressurizer Relief Tank is located inside the Reactor Building crane-wall, and is normally not accessible during normal power operation due to radiological conditions. Pressurizer Relief Tank maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.11.5 Tests and Inspections

There are no required tests or inspections performed on the Pressurizer Relief Tank.

5.5.12 Reactor Coolant System Pressure Boundary Valves

5.5.12.1 Design Bases

As noted in Section 5.2, all valves out to and including the second valve normally closed are capable of automatic or remote closure, larger than three-fourths inch, are ANS Safety Class 1, and ASME III, Code Class 1 valves[1] .

All three-fourths inch valves are Class 2 since the interface with the Class 1 piping is provided with suitable orificing for such valves. Valves identified as primary isolation valves are provided with the means to periodically assess backflow leakage. Reactor Coolant System Pressure Isolation Valves are listed in Table 5-50. For a check valve to qualify as the system boundary, it must be located inside the Containment system. Valves in the reactor pressure boundary and in other seismic Category 1 systems are tabulated in Table 6-113, and Table 5-5, Active and Inactive Valves in the Reactor Coolant System Boundary. Each valve is designed to withstand the most severe environmental conditions applicable to that valve. Valves may be subjected to various conditions such as post LOCA radiation, extreme temperatures and pressures. Each valves applicable conditions are specified in the valve specification.

Materials of construction are specified to minimize corrosion/erosion and to assure compatibility with the environment.

Valve leakage is minimized to the extent practicable by design.

Valve stresses are also maintained within the limits of ASME Section III and the requirements specified in Section 3.7.2.

Applicable code cases and addenda are determined by purchase date subject to the limitations of 10CFR 50, Section 55.55a.

5.5.12.2 Design Description

All valves in the Reactor Coolant System which are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant, such as for hard surfacing and packing, are special materials.

Typically, all manual and motor-operated valves of the Reactor Coolant System which are three inches and larger are provided with double-packed stuffing boxes and stem intermediate lantern gland leakoff connections. Typically, all throttling control valves, regardless of size, are provided with double-packed stuffing boxes and with stem leakoff connections. Leakoff connections are piped to a collection system as required.

Gate valves at the Engineered Safety Features interface are either wedge design or parallel disc and are essentially straight through. The wedge may be either split or solid. All gate valves have backseat and outside screw and yoke. Globe valves, “T” and “Y” style, are full ported with outside screw and yoke construction. Check valves are either swing type or spring loaded, lift piston type for sizes two inches and smaller and swing type or tilting disc type for sizes two and one-half inches and larger. No stainless steel check valves have body penetrations other than the inlet, outlet and bonnet. The check hinge is serviced through the bonnet.

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[1] Valve closure time must be such that for any postulated component failure outside the system boundary, the loss of reactor coolant would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup system. Normal makeup systems are those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, operation or cooldown

Valves at the Residual Heat Removal System interface are provided with interlocks that meet the intent of IEEE-279. These interlocks are discussed in detail in Sections 5.5.7 and 7.4.1.5.

The isolation valves between the accumulators and the Reactor Coolant System are normally open; however, these valves are provided with controls to assure opening (if closed for testing purposes) on a safety injection signal. In that the subject valves are normally open and do not serve as an active device during LOCA, IEEE 279 (1971) is not applicable in this situation. Therefore, the subject valve control circuit is not designed to this standard. The controls are discussed in detail in Section 6.3.

Design parameters for reactor coolant boundary valves are given in Table 5-46.

5.5.12.3 Design Evaluation

Stress analysis of the Reactor Coolant Loop/Support System, discussed in Sections 3.7 and 5.2 assures acceptable stresses for all valves in the reactor coolant pressure boundary under every anticipated condition.

Reactor coolant chemistry parameters are specified to minimize corrosion. Periodic analyses of coolant chemical composition, discussed in the Technical Specifications assure that the reactor coolant meets these specifications. The upper-limit coolant velocity of about 50 feet per second precludes accelerated corrosion.

Valve leakage is minimized by design features as discussed above.

The valves are designed and fabricated to meet the requirements of ASME XI.

All Reactor Coolant System boundary valves required to perform a safety function, during the short term recovery from transients or events considered in the respective operating condition categories, operate in less than ten seconds.

5.5.12.4 Tests and Inspections

Initial hydrostatic seat leakage and operation tests are performed on reactor coolant boundary valves as required by ASME III and Technical Specifications. Subsequent tests for repairs or replacements are performed as required by ASME, Section XI and periodic tests are performed in accordance with the technical specifications.

There are not full-penetration welds within valve body walls. Valves are accessible for disassembly and internal visual inspection.

5.5.12.5 Radiological Considerations

During normal operation, personnel radiological exposure associated with operation of the Reactor Coolant System Pressure Boundary Valves is limited, as the valves are passive checks or can be remotely operated from the Control Room. The Reactor Coolant System Pressure Boundary Valves are located inside the Reactor Building crane-wall, and are normally not accessible during normal power operation due to radiological conditions. Reactor Coolant System Pressure Boundary Valve maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.13 Safety And Relief Valves

5.5.13.1 Design Bases

The combined capacity of the pressurizer safety valves is designed to accommodate the maximum surge resulting from complete loss of load. This objective is met without reactor trip or any operator action

provided that the steam safety valves open as designed when steam pressure reaches the steam-side setting.

The power-operated pressurizer relief valves are designed to limit pressurizer pressure to a value below the fixed high pressure reactor trip setpoint.

5.5.13.2 Design Description

The pressurizer safety valves are the totally enclosed pop type. The valves are spring loaded self-activated and with back pressure compensation features.

The six-inch pipe connecting the pressurizer nozzles to their respective code safety valves, are shaped in the form of a loop seal. Condensate, as a result of normal heat losses to the ambient, will be drained off by a continuous drain back to the pressurizer. This line is located in the bottom of each loop. The valves are equipped with flex-i-disc inserts that seal on steam, and will prevent the leakage of hydrogen gas or steam through the valve seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they start lifting.

The relief valves are quick-opening, operated automatically or by remote control. Remotely operated stop valves are provided to isolate the power operated relief valves if excessive leakage develops.

Temperatures in the pressurizer safety and relief valve discharge lines are measured and indicated. An increase in a discharge line temperature is an indication of leakage through the associated valve.

Design parameters for the pressurizer spray control, safety and power relief valves are given in Table 5-47.

5.5.13.3 Design Evaluation

The pressurizer safety valves prevent Reactor Coolant System pressure from exceeding 110 percent of system pressure, in compliance with the ASME Nuclear Power Plant Components Code, Section III.

The pressurizer power relief valves prevent actuation of the fixed high-pressure trip for all design transients up to and including the design step load decrease, with steam dump but without reactor trip. The relief valves also limit in a desirable manner opening of the spring-loaded safety valves.

Force time histories were generated for the Safety and Relief valve discharges. An analysis of the piping and components was performed using a direct integration force time history procedure. The results of this analysis are used to show that the requirements of subsection 5.5.13.4 are met.

Nozzle design of the pressurizer is in accordance with Section 3.9.2.

5.5.13.4 Tests and Inspections

Testing performed on safety and relief valves consists of operational and hydrostatic tests.

The safety valves are periodically tested for operability with steam. Inservice testing of the safety valves is performed in accordance with ASME Code, Section XI, Section IWV.

There are no full penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

Pursuant to requirements of NUREG 0737; the McGuire Station PORVs and Safety relief valves have been qualified to operating conditions and design basis transients. Duke Power participated in an EPRI testing program verified that the valves used at EPRI test facility opened and reclosed as expected for the design basis flow conditions. The data generated by these tests was used for evaluation of the adequacy of the discharge piping and supports (Reference 4). By letter dated July 27, 1995, Duke informed the NRC that modifications were being prepared to change the seal configuration for the code safety valves to

seal with a steam medium and to install continuous drains for the existing loop piping. The implementation of these design solutions permanently resolved the concerns regarding NUREG-0737, Item II.D.1.

NRC Generic Letter (GL) 90-06, Resolution of Generic Issue 70, "Power Operated Relief Valve and Block valve reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," was issued on June 25, 1990. In response to the issues identified in the GL, McGuire evaluated the treatment of the Pressurizer PORVs and Block Valves. The results of that study are enumerated in Reference 44, "Generic Letter 90-06". The pressurizer PORVs and Block Valves at McGuire and associated testing programs meet or exceed the minimum requirements suggested by the NRC as an acceptable response to Generic Letter 90-06.

5.5.13.5 Radiological Considerations

During normal operation, personnel radiological exposure associated with operation of the Safety and Relief Valves is limited, the system can be operated automatically or remotely from the Control Room. The Safety and Relief Valves are located inside the Reactor Building crane-wall, and are normally not accessible during normal power operation due to radiological conditions. Safety and Relief Valve maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.14 Component Supports

5.5.14.1 Design Bases

The equipment supports are designed to sustain the loads imposed on the system under normal operating conditions. Consideration is also given to abnormal loading conditions such as seismic and pipe rupture. The two types of seismic loadings are Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE).

Normal as well as abnormal loads are considered singly and in combination as shown in Table 5-48. The stress levels for each of the possible combinations are as shown in Table 5-48.

The abnormal loads considered and evaluated for the vessel supports include the transient forcing functions resulting from the loop hydraulic forces, the vessel internals hydraulic loadings, and the asymmetric reactor cavity pressure forces as defined in Section 6.2.1.3.1.2. The above defined loads were combined as defined in Table 5-48. The design loads actually used are greater than the calculated maximum abnormal loads (7725^k vs. 2350^k, vertical direction; 3640^k vs. 3038^k, horizontal direction). All calculated design stress levels are below the allowables as defined in Table 5-48.

Equipment supports are designed in a way to allow virtually unrestrained lateral thermal movement of the loop during normal operating conditions.

5.5.14.2 Design Description

1. Steam Generator

   The steam generator support system consists of vertical steel columns, lower and upper lateral steel frames. See Figure 5-30 through Figure 5-32 for outline of the steam generator support system.

2. Pump

   The reactor coolant pump support system consists of vertical steel columns and a lateral steel frame. Figure 5-32 through Figure 5-34 show an outline of the support system of the reactor coolant pump.

3. Pressurizer

The pressurizer (Figure 5-35) is supported by:

a. A lower lateral frame (Figure 5-36) which is anchored to the crane wall and to vertical steel columns which in turn are anchored to the operating floor, and

b. An upper steel ring anchored to the crane wall and the pressurizer enclosure walls (Figure 5-37).

4. Reactor Vessel

Figure 5-38 shows an outline of a typical vessel support. The supports are individual rectangular steel box structures beneath the vessel nozzles and anchored to the primary shield wall. Loads in both the vertical and horizontal direction are transferred by the Westinghouse support shoe into the vessel support by direct bearing.

5.5.14.3 Structural Acceptance Criteria

The equipment supports are designed such that the stress levels in these supports are below the yield stress and for the supports to maintain their elastic behavior if subjected to any of the loading combinations subscribed in Table 5-48. In some localized areas the stress level is allowed to exceed the allowable yield stress but never exceed the ultimate stress.

5.5.14.4 Materials, Quality Control and Special Construction Techniques

The materials used for the design and construction of the equipment supports are:

1. Steel plates - ASME SA-516, Grade 70. (Rings: SG Upper Lateral Support Rings are SA-533, Type B, Class-2)
2. Steel Standard Shapes - ASTM A-36. (Shims: Top Shim Retainers are SA-516, GR 70 shims used between 4" splice plate are SA-240, Type-304 or 533)
3. Forged Pieces - ASME SA-540, Grade B23. Bolts - SA-193, Gr B7 and SA-540, Gr 22, CL-3 Nuts – SA-194, Gr 7 SA-520, Gr 22, CL-3
4. Concrete - Strength of 3000 psi or 5000 psi after 28-day test.
5. Reinforcing Steel - ASTM A-615, Grades 40 and/or 60.
6. Steel Pipes - USS T-1, Constructional Alloy Steel (ASTM 517 Grade F).

The design and fabrication of the equipment supports are in accordance with the AISC Specifications for the “Design, Fabrication, and Erection of Structural Steel for Buildings”, 1969 Edition, and applicable portions of the ASME Boiler and Pressure Vessel Code. Steam generator upper lateral restraint ring in accordance with ASME Boiler and Pressure Vessel Code Division-1, Section III, Subsection NF-Component supports 1986 thru 1988 Addenda, and ASME Code Division-1, Section III, App. "F", 1986 thru 1988 Addenda. Welder qualifications, Welding Procedures, and Inspection of Welded Joints are specified to be in accordance with Section IX of the ASME Code. The reinforced concrete used in the supports is designed according to the AC1-318, 1971 Edition.

5.5.14.5 Radiological Considerations

During normal operation, personnel radiological exposure associated with passive Component Supports for the Reactor Vessel, Reactor Coolant Pumps, Pressurizer, and Steam Generators is limited. Component Supports are normally not accessible during normal power operation due to radiological conditions. Maintenance or inspections of Component Supports would be performed during refueling outages, utilizing routine radiological controls.

5.5.14.6 Tests and Inspections

All tests and inspections are performed per ASME Section XI, Inservice Inspection Program. Reference section 18.2.16, Inservice Inspection Plan and section 5.2.8, Inservice Inspection Program.

5.5.15 Reactor Coolant Vent System

5.5.15.1 Design Basis

This system is designed to vent non-condensible gases from the RCS. The RCS vent system is safety grade, seismically qualified and meets the requirements of IEEE 279-1971. This system satisfies the single failure criteria.

5.5.15.2 System Description

Duke has installed a reactor vessel head high point vent that is remotely operable from the McGuire control room. A one-inch line has been added to the existing reactor vessel manual vent line with the connection located before the first isolation valve. The new vent line contains two parallel flow paths with redundant fail closed solenoid valves in each flow path. The valves have been designed to pass non-condensible gases, water, steam, and mixtures thereof. Under normal operation these valves are deenergized. Valve position is indicated in the control room. Train A emergency power serves both isolation valves in one flow path, and Train B emergency power serves both isolation valves in the parallel flow path. A flow limiting orifice has been installed in the common line downstream of the isolation valves.

The head vent system has been designed to single failure criteria. If any single failure prevents a venting operation through one flow path the second flow path is available for venting the RCS. The two isolation valves in each flow path provide a single failure method of isolating RCS venting.

Inadvertent actuation of RCS venting is limited by the use of fail closed solenoid isolation valves. In addition the use of an orifice in the common line downstream of the valves limits the flow to less than the makeup capacity of the RCS charging pumps.

Exhaust from the vent system is directed to the Pressurizer Relief Tank (PRT) and therefore will not impinge upon vital equipment. A path is available to allow venting hydrogen from the reactor vessel head to the Waste Gas System storage tanks, via the pressurizer relief tank, should such an option be judged desirable. The PRT is located in the lower containment which is ventilated and cooled by four air handling units. In addition, the hydrogen skimmer system has ducts in the lower containment high points to disperse any accumulated hydrogen.

5.5.15.3 Design Evaluation

Assuming that 100% hydrogen is being vented from the reactor vessel head, the vent system flow rate is 14 cfm. This allows the venting of the gas volume in the reactor head in approximately 1 hour.

The power-operated relief valves (PORV) are used to vent the RCS pressurizer. the PORV's are discussed in the McGuire FSAR Section 5.2.2. The RCS vent is located at the top of the reactor vessel head which is the high point of the reactor vessel and coolant loops. This system in conjunction with the PORV's provides a venting capability for the entire RCS with the exception of the U-tube steam generators.

A flow diagram of the McGuire RCS including the RCS vent system is provided in the McGuire FSAR Figure 5-1.

A postulated break of the reactor vessel head vent line upstream of the flow limiting orifice would result in a small LOCA of not greater than one inch diameter. Such a break is similar to those analyzed in

WCAP-9600 for hot leg breaks or pressurizer vapor space breaks. Since the break location in the head vent line would behave similarly to the hot leg break, this postulated break would result in no calculated core uncovery.

5.5.15.4 Radiological Considerations

The design of the Reactor Coolant Vent System does not pose any radiological consequences for routine operation. The Reactor Coolant Vent System is located inside the Reactor Building crane-wall and is normally not accessible during normal power operation due to radiological conditions. Any Reactor Coolant Vent System maintenance or inspections would be performed during refueling outages, utilizing routine radiological controls.

5.5.15.5 Tests and Inspections

The Reactor Coolant head vents are periodically tested per SLC 16.5.10 surveillance requirements.

5.5.16 References

1. Letter from William O. Parker, Jr. to Harold R. Denton (NRC) dated July 13, 1983. Subject: Loose Thermal Sleeves.
2. Letter from William O. Parker, Jr. to Harold R. Denton (NRC) dated January 20, 1982. Subject: Natural Circulation (Response to NRC Generic Letter 81-21).
3. Letter from Thomas M. Novak (NRC) to H. B. Tucker (Duke) dated May 17, 1981. Subject: Anticipatory Reactor Trip (NUREG 0737 Item II.K.3.10.
4. McGuire Nuclear Station, Responses to TMI Concerns, Item II.D.1.
5. McGuire Nuclear Station, Response to TMI Concerns, Item II.G.
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36. Letter from H.B. Tucker to Harold R. Denton (NRC) dated December 14, 1983 regarding justification for continued operation with the seven thermal sleeves removed from selected locations in the Reactor Coolant System.
37. Letter from Darl Hood (NRC) to H.B. Tucker dated December 30, 1986. Subject: Reactor Coolant System Thermal Sleeves – McGuire Nuclear Station, Units 1 and 2.
38. McGuire SER, Supplement 6, Section 3.9.2: Thermal Sleeves.
39. Letter from T.C. McMeekin (Duke) to U.S. NRC dated July 27, 1995, Closure of TACs 76433 and 76434 Regarding NUREG-0737, Item II.D.1.
40. Letter from T.C. McMeekin (Duke) to U.S. NRC dated January 30, 1992, titled "McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369 and 50-370, NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification."
41. Letter from Timothy A. Reed (NRC) to T.C. McMeekin (Duke) dated April 3, 1992, titled "NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification," for McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M72142/M72143)
42. Coslow, B.J., et al., "Pressurizer Surge Line Thermal Stratification Generic Detailed Analysis", WCAP-12639, June 1990.
43. Letter to the U.S. Nuclear Regulatory Commission from T.P. Harrall dated October 13, 2008, titled "Duke Energy Carolinas, LLC (Duke) Oconee Nuclear Station, Units 1, 2 & 3, Docket Nos. 50-269, 50-270, 50-287, McGuire Nuclear Station, Units 1 & 2, Docket Nos. 50-369, 50-370, Catawba Nuclear Station, Units 1 & 2, Docket Nos. 50-413, 50-414, Generic Letter 2008-01, 9-Month Response."
44. Letter from H.B. Tucker (Duke) to NRC dated December 20, 1990. Subject: Generic Letter 90-06.
45. Nuclear Regulatory Commission, Letter to All Licensees of Operating PWRs and Holders of Construction Permits for PWRs, from Frank J. Miraglia, July 9, 1987. "Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled (Generic Letter 87-12)."

46. "Generic Letter 87-12, Loss of RHR", transmitted by letter dated October 2, 1987 from Warren H. Owen, Duke Power Company, to U.S. Nuclear Regulatory Commission, Attention: Document Control Desk.
47. Nuclear Regulatory Commission, Letter to All Licensees of Operating PWRs and Holders of Construction Permits for PWRs, from Dennis M. Crutchfield, October 17, 1988, "Loss of Decay Heat Removal (Generic Letter 88-17) 10 CFR 50.54 (f)."
48. "Generic Letter 88-17, Loss of Decay Heat Removal", transmitted by letter dated January 3, 1989, H.B. Tucker, Duke to Document Control Desk, U.S. Nuclear Regulatory Commission.
49. "Generic Letter 88-17, Loss of Decay Heat Removal", transmitted by letter dated February 2, 1989, H.B. Tucker, Duke to Document Control Desk, U.S. Nuclear Regulatory Commission.
50. Nuclear Regulatgory Commission, Letter from T.A. Reed to M.S. Tuckman (DPC), October 16, 1991, re: NRC Generic Letter 88-17, Programmed Enhancements for Generic Letter 88-17, "Loss of Decay Heat Removal Systems" for McGuire Nuclear Stations, Units 1&2 (TAC Nos. 69752/69753).
51. NRC Generic Letter 87-12, dated July 9, 1987, Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled.
52. Letter from H.B. Tucker (Duke) to U.S. NRC dated October 30, 1989, Technical Specification Amendment to Residual Heat Removal Flow Requirements.
53. Babcock & Wilcox International Report No. BWI-222-7693-LR-01, Rev. 5, dated January 1996, "Replacement Steam Generator Topical Report"

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5.6 Instrumentation Application

Process control instrumentation is provided for the purpose of acquiring data on the pressurizer and on a per loop basis for the key process parameters of the Reactor Coolant System (including the reactor coolant pump motors) as well as for the Residual Heat Removal System. The pick-off points for the Reactor Coolant System are shown in the flow diagrams Figure 5-1); and for the Residual Heat Removal System, in flow diagram Figure 5-28. In addition to providing input signals for the protection and control systems, the instrumentation sensors furnish input signals for monitoring and/or alarming purposes for the following parameters:

1. Temperatures
2. Flows
3. Pressures
4. Water Levels

In general these input signals are used for the following purposes:

1. Provide input to the Reactor Protection System for reactor trips as follows:
   a. Overtemperature ΔT
   b. Overpower ΔT
   c. Low pressurizer pressure
   d. High pressurizer pressure
   e. High pressurizer water level
   f. Low primary coolant flow

      It is noted that the following parameter, which is also sensed to generate an input to the Reactor Protection System while not part of the Reactor Coolant System, is included here for purposes of completeness:
      Low low steam generator water level
2. Provide input to the Engineered Safety Features Actuation System as follows:
   a. Pressurizer low pressure

      It is noted that the following parameter, which is also sensed to generate an input to the Engineered Safety Features Actuation System, while not part of the Reactor Coolant System, is included here for purposes of completeness:
      Low steam line pressure
3. Furnish input signals to the non-safety related systems such as the control systems and surveillance circuits so that:
   a. Reactor coolant average temperature (T_{avg}) is maintained within prescribed limits.
   b. Pressurizer water level control, using 2nd highest T_{avg} to program the setpoint, maintains the coolant level in the pressurizer within prescribed limits.
   c. Pressurizer pressure is controlled within specified limits.
   d. Steam dump control, using 2nd highest T_{avg} control, accommodates a sudden loss of generator load.
   e. Information is furnished to the Control Room operator and at local stations for monitoring.

The following is a functional description of the system instrumentation. Unless otherwise stated, all indicators, recorders and alarm annunciators are located in the Control Room.

Temperature Measuring Instrumentation

1. Narrow Range Cold Leg and Hot Leg Temperatures

   The hot leg temperature measurement on each loop is accomplished with three fast response narrow range RTDs mounted in thermowells, spatially located at intervals of 120° around the hot leg. One fast response narrow range RTD is located in each cold leg at the discharge of the reactor coolant pump. Temperature streaming in the cold leg is not a concern due to the mixing action of the RCP, hence, only one cold leg RTD is required.

   This cold leg temperature measurement, together with the average T_{hot} obtained for the three hot leg temperatures, is used to calculate reactor coolant loop ΔT and $T_{avg.}$. A weighting technique may be applied to the three hot leg RTDs to reduce process noise or temperature variations resulting from flow shifts. A new penetration in each cold leg houses an additional well mounted narrow range RTD for use as an installed spare.

2. Wide Range Cold Leg and Hot Leg Temperatures

   Temperature detectors, located in the thermometer wells in the cold and hot leg piping of each loop, supply signals to wide-range temperature recorders. This information is used by the operator to control coolant temperature during startup and shutdown.

3. Pressurizer Temperature

   There are two temperature detectors in the pressurizer, one in the steam phase and one in the water phase. Both detectors supply signals to temperature indicators and alarms. The steam phase detector, located near the top of the vessel, is used during startup to determine water temperature when the pressurizer is completely filled with water. The water phase detector, located at an elevation near the center of the heaters, is used during cooldown when the steam phase detector response is slow due to poor heat transfer.

4. Surge Line Temperature

   This detector supplies a signal for a temperature indicator and a low-temperature alarm. Low temperature is an indication that the continuous spray rate is too small.

5. Safety and Relief Valve Discharge Temperatures

   Temperatures on the pressurizer safety and relief valve discharge lines are measured and indicated. (Note: strap-on RTD's are used.) An increase in a discharge line temperature is an indication of leakage through the associated valve.

6. Spray Line Temperatures

   Temperatures in the spray lines, one from each of two loops are measured and indicated. Alarms from these signals are actuated by low spray water temperature. Alarm conditions indicate insufficient flow in the spray lines.

7. Pressurizer Relief Tank Water Temperature

   The temperature of the water in the pressurizer relief tank is indicated, and an alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

8. Reactor Vessel Flange Leakoff Temperature

The temperature in the leakoff line from the reactor vessel flange 0-ring seal leakage monitor connections is indicated. An increase in temperature above ambient is an indication of 0-ring seal leakage. High temperature actuates an alarm.

9. Reactor Coolant Pump Motor Temperature Instrumentation

   a. Thrust Bearing Upper and Lower Shoes Temperature:

      Thermocouples are provided with one located in the shoe of the upper and one in the shoe of the lower thrust bearing. These thermocouples provide a signal for a high temperature alarm and indication. Monitoring of these detectors is provided by the computer.

   b. Stator Winding Temperature:

      The stator windings contain six (6) resistance-type detectors, two per phase, imbedded in the windings. A signal from one of these detectors is monitored by the computer which actuates a high temperature alarm.

   c. Upper and Lower Bearing Temperature:

      Thermocouples are located one in the upper and one in the lower radial bearings. Signals from these thermocouples actuate a high temperature alarm and indication.

<u>Flow Indication</u>

1. Reactor Coolant Loop Flow

   Flow in each reactor coolant loop is monitored by three differential pressure measurements at a piping elbow tap in each reactor coolant loop. These measurements on a two-out-of-three coincidence circuit provide a low flow signal to actuate a reactor trip.

<u>Pressure Indication</u>

1. Pressurizer Pressure

   Four pressurizer pressure transmitters provide signals for individual indicators on the main control board in addition to actuation of both a low pressure trip and a high pressure trip. One channel is recorded.

   Four transmitters provide low pressure signals for safety injection initiation and three transmitters provide safety injection signal unblock during startup. All four pressurizer pressure transmitter signals are transmitted to the Ovation PCS, where they are available for display and trending on any PCS workstation. The 2nd highest pressure signal is used to develop the demand signal for pressurizer pressure control.

   For normal operation, a small group of heaters is controlled by variable power to maintain the pressurizer operating pressure. If the pressure error signal falls towards the bottom of the variable heater control range all pressurizer heaters are turned on.

   The four pressurizer pressure transmitters are used to provide two separate signal sets to the Ovation PCS. Pressurizer Pressure Set 1 controls the variable heaters, ON/OFF heaters, PORV NC-32/36 interlock, PORV NC-34A actuation, and pressurizer spray valves. Pressurizer Pressure Set 2 controls PORV NC-32/36 actuation and PORV NC-34A interlock.

   The spray valves are proportionally controlled in a range above normal operating pressure with spray flow increasing as pressure rises. If the pressure rises significantly above the proportional range of the spray valves, PORV NC-34A is opened (interlocked with a separate transmitter to prevent spurious operation). A further increase in pressure will actuate NC-32B and NC-36B. Should pressure continue to increase, a high pressure reactor trip is initiated.

2. Reactor Coolant Loop Pressures

   Reactor coolant loop pressure is measured, indicated and transmitted as shown on the flow diagram (Figure 5-1).

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3. Reactor Coolant Pump Motor Pressure

   a. Oil Lift Switch

      One pressure switch is furnished at the discharge of both AC motor driven lift oil pumps that provide a low pressure alarm in the control room. Three pressure switches, arranged in a 2/3 coincidence network, are located on the oil lift supply header to prevent the start of the reactor coolant pump if there is inadequate pressure in the system. A local pressure gauge is also provided.

   b. Lower Oil Reservoir Liquid Level

      A level transmitter is provided in the motor lower radial bearing oil reservoir. The transmitter actuates a high and low level alarm on the main control board.

   c. Upper Oil Reservoir Liquid Level

      A level transmitter is provided in the motor upper radial bearing and thrust bearing oil reservoir. The transmitter actuates a high or low level alarm on the main control board.

1. Liquid Level Indication

   a. Pressurizer Level

      Three pressurizer liquid level transmitters provide signals for use in the Reactor Control and Protection Systems, and the Chemical and Volume Control System. Each transmitter instrument channel provides an independent high water level signal that is used to actuate an alarm and a reactor signal that is used to actuate an alarm and a reactor trip. The transmitters instrument channel also provides independent low water level signals that activate an alarm.

      Each transmitter instrument also provides a signal for a level indicator that is located on the main control board.

      In addition to the above pressurizer liquid level signals are transmitted to the Ovation PCS for specific functions as follows:

      1) Any or all of the three level transmitters may be selected by the operator for trend display on any PCS workstation located in the main control room. One pressurizer level signal provides input to a recorder in the main control room. This same recorder is used to display a pressurizer reference liquid level.

      2) The median signal select (MSS) value of the three transmitter signals is selected automatically to perform the following functions:

         a) The MSS pressurizer level actuates an alarm when the liquid level falls to a fixed level setpoint, trips the pressurizer heaters “off”, and closes the letdown line isolation valves.

         b) The MSS pressurizer level is sent to the liquid level controller for charging flow control and also initiation of a low flow (hi demand) alarm. The MSS signal is also compared to the reference level and actuates a high level alarm and turns on all pressurizer backup heaters if the actual level exceeds the reference level. If the actual level is lower than the reference level, a low alarm is actuated.

3) A fourth independent pressurizer level transmitter that is calibrated for low temperature conditions, provides water level indication during startup, shutdown and refueling operations.

2. Pressurizer Relief Tank Level

   The pressurizer relief tank level transmitter supplies a signal for an indicator and high and low level alarms.

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valve setpoints, and the protection system setpoint pressures are listed in Table 5-10.

Process control instrumentation for the Residual Heat Removal System is provided for the following purposes:

1. Furnish input signals for monitoring and/or alarming purposes for:
   a. Temperature indications
   b. Pressure indications
   c. Flow indications
2. Furnish input signals for control purposes of such processes as follows:
   a. Control valve in the residual heat removal pump and heat exchanger miniflow line so that it opens at flows below a preset limit and closes at flows above a preset limit.
   b. Residual heat removal inlet valves control circuitry. See Section 7.4.1.5 for the description of the interlocks.
   c. Residual heat removal pump circuitry for starting residual heat removal pumps on “S” signal.

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