C.I.5. Reactor Coolant and Connecting Systems

Chapter 5 of the final safety analysis report (FSAR) should provide information regarding the reactor coolant system and systems to which it connects. Special consideration should be given to the reactor coolant system and pressure-containing appendages out to and including isolation valving which is the "reactor coolant pressure boundary" (RCPB), as defined in Title 10, Section 50.2(v), of the Code of Federal Regulations [10 CFR 50.2(v)].

Evaluations, together with the necessary supporting material, should be included to show that the reactor coolant system is adequate to accomplish its intended objective and to maintain its integrity under conditions imposed by all foreseeable reactor behaviors, including both normal and accident conditions. The information should permit an independent determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete, and all necessary evaluations have been performed. Evaluations included in other chapters that have a bearing on the reactor coolant system should be referenced.

C.I.5.1 *Summary Description*

This section of the FSAR should provide a summary description of the reactor coolant system and its various components. This description should indicate the independent and interrelated performance and safety functions of each component, and should include a tabulation of important design and performance characteristics.

C.I.5.1.1 Schematic Flow Diagram

Provide a schematic flow diagram of the reactor coolant system denoting all major components, principal pressures, temperatures, flow rates, and coolant volume under normal steady-state full-power operating conditions.

C.I.5.1.2 Piping and Instrumentation Diagram

Provide a piping and instrumentation diagram of the reactor coolant system and connected systems delineating the following:

(1) extent of the systems located within the containment

(2) points of separation between the reactor coolant (heat transport) system and the secondary (heat utilization or removal) system

(3) extent of insolubility of any fluid system as provided by the use of isolation valves between the radioactive and non-radioactive sections of the system, isolation valves between the RCPB and connected systems, and passive barriers between the RCPB and other systems

C.I.5.1.3 Elevation Drawing

Provide an elevation drawing showing principal dimensions of the reactor coolant system in relation to the supporting or surrounding concrete structures from which a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout can be gained.

C.I.5.2 *Integrity of the Reactor Coolant Pressure Boundary*

This section of the FSAR should provide a discussion of the measures to be employed to ensure and maintain the integrity of the RCPB throughout the plant design lifetime.

C.I.5.2.1 **<u>Compliance with Codes and Code Cases</u>**

C.I.5.2.1.1 *Compliance with 10 CFR 50.55a*

Provide a table showing compliance with the NRC's regulations in 10 CFR 50.55a, "Codes and Standards." This table should identify pressure vessel components, piping, pumps, valves, and storage tanks. The applicable component code, code edition and addenda, and, when required, the component order date of each Class 1 component within the RCPB, as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, may be identified by reference to the table of structures, systems, and components in Section 3.2 of the FSAR or included in this section of the FSAR.

In the event there are cases in which conformance to the regulations of 10 CFR 50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, provide a complete description of the circumstances resulting in such cases and the basis for proposed alternative requirements. Describe how an equivalent and acceptable level of safety and quality will be provided by the proposed alternative requirements.

C.I.5.2.1.2 *Compliance with Applicable Code Cases*

Provide a list of ASME Code Case that will be applied to components within the RCPB. Each component for which a Code Case has been applied should be identified by Code Case number, revision, and title. Caution is advised in the use of Code Cases to ensure that the applicable revision of the Code Case is identified for each component application. Regulatory Guide 1.84,"Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (latest revision), lists those ASME Section III, Division 1, Code Cases that are generally acceptable to the NRC staff. Indicate the extent of conformance with the recommendations of Regulatory Guide 1.84. If Code Cases other than those listed are used, show that their use will result in as acceptable a level of quality and safety for the component as would be achieved by following the Code Cases that the NRC staff has endorsed in Regulatory Guide 1.84.

C.I.5.2.2 **<u>Overpressure Protection</u>**

Provide information, as set forth in the following subsections, to accommodate an evaluation of the systems that protect the RCPB and the secondary side of steam generators from over pressure. These systems include all pressure-relieving devices (safety and relief valves) for the following:

(1) reactor coolant system

(2) primary side of auxiliary or emergency systems connected to the reactor coolant system

(3) any blow down or heat dissipation systems connected to the discharge of these pressure-relieving devices

(4) secondary side of steam generators

C.I.5.2.2.1 ***Design Bases***

Provide the design bases on which the functional design of the over pressure protection was established. Address over pressure protection for the RCPB during reactor power operation and low-temperature operation. Describe compliance with General Design Criterion (GDC) 15, as defined in Appendix A to 10 CFR Part 50, as it relates to the RCPB design conditions not being exceeded during any condition of normal operation or anticipated operational occurrences, as well as GDC 31, as it relates to designing the RCPB with sufficient margin to ensure that it behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

C.I.5.2.2.2 ***Design Evaluation***

Provide an evaluation of the functional design of the over pressure protection system. This evaluation should include an analysis of the system's capability to perform its function, describe the analytical model used in the analysis, and discuss the bases for its validity. Also discuss and justify the assumptions used in the analysis, including the plant initial conditions and system parameters. List the systems and equipment assumed to operate and describe their performance characteristics. Provide studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics.

Describe the design of over pressure protection during low-temperature operations, including the capability to relieve pressure during all over pressure events during startup and shutdown conditions at low temperatures, particularly at water-solid conditions. Provide the analysis that demonstrates how over pressure protection is achieved, assuming any single active component failure. Identify all over pressure events and, as a subset, identify the events that can be prevented by preventive interlocks or locking out power. Describe how the over pressure protection system is enabled. Describe the alarms and indications associated with the system. Describe the power source for the system. Discuss whether any credit is taken for active components to mitigate an over pressure event and the additional analysis performed that consider inadvertent system initiation or actuation. If this system uses pressure relief from a low pressure system, discuss how the low-pressure interlocks will not interfere with the operation of this system.

C.I.5.2.2.3 ***Piping and Instrumentation Diagrams***

Provide piping and instrumentation diagrams for the over pressure protection system showing the number, type, and location of all components, including valves, piping, tanks, instrumentation, and controls. Identify the connections and interfaces with other systems.

C.I.5.2.2.4 ***Equipment and Component Description***

Describe the equipment and components of the over pressure protection system, including schematic drawings of the safety and relief valves and a discussion of how the valves operate. Identify the significant design parameters for each component, including the design, throat area, capacity, and set points of the valves and the diameter, length, and routing of piping. List the design parameters (e.g., pressure and temperature) for each component, and specify the number and type of operating cycles, as well as the environmental conditions (e.g., temperature and pressure) for which each component is designed.

C.I.5.2.2.5 ***Mounting of Pressure-Relief Devices***

Describe the design and installation details of the mounting of the pressure-relief devices within the RCPB and the secondary side of steam generators. Specify the design bases for the assumed loads (i.e., thrust, bending, and torsion) imposed on the valves, nozzles, and connected piping in the event that all valves discharge. Describe how these loads can be accommodated; include a listing of these loads and resulting stresses. Material contained in Section 3.9.3.3 of the FSAR may be cross-referenced.

C.I.5.2.2.6 ***Applicable Codes and Classification***

Identify the applicable industry codes and classifications applied to the system.

C.I.5.2.2.7 ***Material Specification***

Identify the material specifications for each component.

C.I.5.2.2.8 ***Process Instrumentation***

Identify all process instrumentation.

C.I.5.2.2.9 ***System Reliability***

Discuss system reliability and the consequences of equipment/component failures.

C.I.5.2.2.10 ***Testing and Inspection***

Identify the tests and inspections to be performed (1) prior to operation and during startup to demonstrate the functional performance and (2) as inservice surveillance to ensure continued reliability. Describe specific testing of the low-temperature over pressure protection system, particularly operability testing, exclusive of relief valves, prior to each shutdown.

C.I.5.2.3 **<u>Reactor Coolant Pressure Boundary Materials</u>**

C.I.5.2.3.1 ***Material Specifications***

Provide a list of specifications for the principal ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and weld materials, to be used in fabricating and assembling each component (e.g., vessels, piping, pumps, and valves) that is part of the RCPB, excluding the reactor pressure vessel. Identify the grade or type and final metallurgical condition of the material placed in service.

C.I.5.2.3.2 ***Compatibility with Reactor Coolant***

Provide the following information relative to compatibility of the materials of construction and the external insulation of the RCPB with the reactor coolant:

(1) Pressurized-water reactor (PWR) coolant chemistry (PWRs only). Describe the chemistry of the reactor coolant and the additives (such as inhibitors). Describe water chemistry, including maximum allowable content of chloride, fluoride, sulfate, and oxygen, as well as the permissible content of hydrogen and soluble poisons. Discuss methods to control water chemistry, including pH. Discuss the industry-recommended methodologies that will be used to monitor water chemistry, and provide appropriate references.

(2) Boiling-water reactor (BWR) coolant chemistry (BWRs only). Describe the chemistry of the reactor coolant and the methods for maintaining coolant chemistry. Provide sufficient information about allowable range and maximum allowable chloride, fluoride, and sulfate contents, maximum allowable conductivity, pH range, location of conductivity meters, performance monitoring, and other details of the coolant chemistry program to indicate whether coolant chemistry will be maintained at a level comparable to the recommendations in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors." Discuss the industry-recommended methodologies that will be used to monitor water chemistry, and provide appropriate references.

(3) Compatibility of construction materials with reactor coolant. Provide a list of the materials of construction exposed to reactor coolant and a description of material compatibility with the coolant, contaminants, and radiolytic products to which the materials may be exposed. If non-metallic are exposed to reactor coolant, include a description of the compatibility of these materials with the coolant.

(4) Compatibility of construction materials with external insulation and reactor coolant. Provide a list of the materials of construction of the RCPB and a description of their compatibility with external insulation and the environment, especially in the event of coolant leakage. Provide sufficient information about the selection, procurement, testing, storage, and installation of any nonmetallic thermal insulation for austenitic stainless steel to indicate whether the concentrations of chloride, fluoride, sodium, and silicate in thermal insulation will be within the ranges recommended in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Provide information on the leach able contaminants in insulation on non-austenitic piping.

C.I.5.2.3.3 ***Fabrication and Processing of Ferritic Materials***

Provide the following information relative to fabrication and processing of ferritic materials used for components of the RCPB:

(1) Fracture toughness. In regard to fracture toughness of the ferritic materials, including bolting materials for components (e.g., vessels, piping, pumps, and valves) of the RCPB, indicate how compliance with the test and acceptance requirements of Appendix G to 10 CFR Part 50 and with Section NB-2300 and Appendix G to Section III of the ASME Code is achieved. Submit the fracture toughness data in tabular form, including information regarding the calibration of instruments and equipment (FSAR).

(2) Control of welding. Provide the following information relative to control of welding of ferritic materials used for components of the RCPB:

(a) Sufficient information regarding the avoidance of cold cracking during welding of low-alloy steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Provide details on proposed minimum preheat temperature and maximum interpass temperature during procedure qualification and production welding. Provide information on the moisture control for low-hydrogen, covered-arc-welding electrodes.

(b) Sufficient information for electroslag welds in the low-alloy steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties." Provide details on the control of welding variables and the metallurgical tests required during procedure qualification and production welding.

(c) In regard to welding and weld repair during fabrication and assembly of ferritic steel components of the RCPB, provide sufficient details for welder qualification for weld areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

(d) Describe the controls to limit the occurrence of underclad cracking in low alloy steel components clad with stainless steel.

(5) Nondestructive examination. Provide sufficient information on nondestructive examination of ferritic steel tubular products (pipe, tubing, flanges, and fittings) for components of the RCPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) will be in conformance with the requirements of the ASME Code.

C.I.5.2.3.4 *Fabrication and Processing of Austenitic Stainless Steels*

Provide the following information relative to fabrication and processing of austenitic stainless steels for components of the RCPB:

(1) Avoidance of stress-corrosion cracking. Provide the following information relative to avoidance of stress-corrosion cracking of austenitic stainless steels for components of the RCPB during all stages of component manufacture and reactor construction:

(a) Sufficient details about the avoidance of sensitization during fabrication and assembly of austenitic stainless steel components of the RCPB to indicate whether the degree of freedom from sensitization will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Describe provisions in material selection and processing to minimize susceptibility to intergranular stress corrosion cracking (IGSCC) consistent with the recommendations in Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," and Revision 2 of NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.

(b) Sufficient details about the process controls to minimize exposure to contaminants capable of causing stress-corrosion cracking of austenitic stainless steel components of the RCPB to show whether process controls will provide, during all stages of component manufacture and reactor construction, a degree of surface cleanliness comparable to that obtainable by following the recommendations of Regulatory Guide 1.44 and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." Describe the controls for abrasive work on austenitic stainless steel surfaces. Identify any pickling used in processing austenitic stainless steel components and describe the restrictions placed on pickling of sensitized materials. Identify the upper yield strength limit of the austenitic stainless steel materials used.

(c) Cold-worked austenitic stainless steels. Provide assurance that cold-worked austenitic stainless steels will have a maximum 0.2% offset yield strength of 620 MPa (90,000 psi) to reduce the probability of stress corrosion cracking in RCPB applications. Identify augmented inservice inspection to ensure the structural integrity of the components during service. In general, cold-worked austenitic stainless steels should not be used for RCPB applications, but may be used when there is no proven alternative available. If such materials are used, describe the service experience and laboratory testing in the simulated environment to which the components will be exposed. Describe the controlled, measured, and documented fabrication process for cold-worked components.

(4) Control of welding. Provide the following information relative to the control of welding of austenitic stainless steels for components of the RCPB:

(a) Sufficient information on electroslag welds in austenitic stainless steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.34 and including the control of welding variables and the metallurgical tests required during procedure qualification and production welding.

(b) In regard to welding and weld repair during fabrication and assembly of austenitic stainless steel components of the RCPB, provide sufficient details about welder qualification for areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate the degree of weld integrity and quality comparable to that obtainable by following the recommendations of Regulatory Guide 1.71.

(3) Nondestructive examination. Provide sufficient information about the program for nondestructive examination of austenitic stainless steel tubular products (pipe, tubing, flanges, and fittings) for components of the RCPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) will be in conformance with the ASME Code.

C.I.5.2.3.5 ***Prevention of Primary Water Stress Corrosion Cracking (PWSCC) for Nickel-Base Alloys (PWRs only)***

Provide the following information relative to fabrication and processing of nickel-based alloys for components of the RCPB:

(4) Identify the nickel-based alloy components of the RCPB and discuss the prevention of primary water stress corrosion cracking (PWSCC). Identify the inservice inspections performed to demonstrate the nickel-based alloy materials are not susceptible to PWSCC, and describe test results that demonstrate the materials are not susceptible to PWSCC.

(5) For nickel-chromium-iron alloys used as RCPB materials, provide the technical basis for use either by identification (based upon demonstrated satisfactory use in similar applications) or presentation of information to support use of the material under the expected environmental conditions (e.g., exposure to reactor coolant). Operating experience has indicated that certain nickel-chromium-iron alloys (e.g., Alloy 600 and Alloy 182/82) are susceptible to PWSCC. Alloy 690 has improved resistance to stress corrosion cracking in comparison to Alloy 600 and Alloy 182/82 previously used in reactor plants.

C.I.5.2.3.6 ***Threaded Fasteners***

Provide a summary description of the program for ensuring the integrity of bolting and threaded fasteners and their adequacy. Reference FSAR Section 3.13, as appropriate.

C.I.5.2.4 **<u>Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary</u>**

C.I.5.2.4.1 ***Inservice Inspection and Testing Program***

Discuss the inservice inspection and testing program for the NRC Quality Group A components of the RCPB (ASME Boiler and Pressure Vessel Code, Section III, Code Class 1 components) that complies with the requirements of 10 CFR 50.55a. Provide sufficient detail to show that the inservice inspection program meets the requirements of Section XI of the ASME Code. Because the inservice inspection program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described with sufficient scope and level of detail to enable the staff to make a reasonable assurance finding regarding its acceptability. Provide descriptive information on the following:

(6) System boundary subject to inspection. Discuss components (other than steam generator tubes) and associated supports to include all pressure vessels, piping, pumps, valves, and bolting.

(7) Accessibility. Describe provisions for access to components and identify any remote access equipment needed to perform inspections.

(8) Examination categories and methods. Discuss the methods, techniques, and procedures used to meet Code requirements. For performing ultrasonic testing (UT) not covered by ASME Section XI, Appendix VIII, the applicant should address the issues/concerns identified in Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Pre-service and Inservice Examinations" to ensure that the UT methods, techniques, and procedures; used for Code examinations; are consistent with those recommended in the subject regulatory guide.

(9) Inspection intervals. Discuss program scheduling in compliance with the Code.

(10) Evaluation of examination results. Discuss provisions for evaluating examination results, including evaluation methods for detected flaws and repair procedures for components that reveal defects.

(11) System pressure tests. Provide descriptive information on system pressure tests and correlated technical specification requirements.

(12) Code exemptions. Identify any exemptions from Code requirements.

(13) Relief requests. Discuss any requests for relief from Code requirements that are impractical as a result of limitations of component design, geometry, or materials of construction.

(14) Code cases. Identify Code Cases that have been invoked.

It is acceptable to submit a general description of the programs as long as it "fully describes" the programs, as defined in SECY-05-0197, and references any applicable standards regarding the ISI and IST programs. When detail information for the ISI and IST programs, as required in this section, is not available at the time of submitting the application, the COL applicant should make a commitment in the application that such information will be provided at a later date (at lease one year prior to fuel load) for NRC review.

C.I.5.2.4.2 ***Pre-Service Inspection and Testing Program***

Describe the pre-service examination program that meets the requirements of Sub-article NB-5280 of Section III, Division I, of the ASME Code. Because the pre-service inspection program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described with sufficient scope and level of detail for the staff to make a reasonable assurance finding on its acceptability.

C.I.5.2.5 **Reactor Coolant Pressure Boundary Leakage Detection**

Describe the RCPB leakage detection systems in sufficient detail to demonstrate the extent to which the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," and Regulatory Guide 1.29, "Seismic Design Classification," have been followed.

Specifically, provide information that will permit comparison with the regulatory positions of the guides, giving a detailed description of the systems employed, their sensitivity and response time, and the reliance placed on their proper functioning. Also, identify the limiting leakage conditions that will be included in the technical specifications.

Identify the leakage detection systems that are designed to meet the sensitivity and response guidelines of Regulatory Guide 1.45 and that will be included in the Technical Specifications. Describe these systems and those that are used for alarm as an indirect indication of leakage, and provide the design criteria. Describe how signals from the various leakage detection systems are correlated to provide information to plant operators regarding leakage location and quantitative leakage flow rate.

Demonstrate that the system is capable of separately monitoring and collecting leakage from both identifiable and unidentifiable sources. Describe the floor drain system to demonstrate that leakage will flow to the sump or tank where it is collected. Identify all potential inter-system leakage paths and the instrumentation used in each path. Demonstrate adequate monitoring capability to ensure that the limits of inter-system leakage assumed in the accident analyses are not exceeded. For radioactivity monitoring leakage detection, describe the primary coolant radioactivity concentration assumption being used to analyze the sensitivity of the leak detection systems.

Describe the provisions to test and calibrate all leakage detection systems. Provide and justify the frequency of testing and calibration. Describe the periodic testing of the floor drainage system that will check for blockage and ensure operability.

C.I.5.3 *Reactor Vessels*

C.I.5.3.1 Reactor Vessel Materials

This section of the FSAR should contain pertinent data in sufficient detail to provide assurance that the materials (including weld materials), fabrication methods, and inspection techniques used for the reactor vessel and applicable attachments and appurtenances conform to all applicable regulations. The FSAR should also describe the specifications and criteria to be applied, and should demonstrate that the requirements have been met.

C.I.5.3.1.1 *Material Specifications*

List all materials in the reactor vessel, applicable attachments, and appurtenances, and provide the material specifications, making appropriate references to Chapter 5 of the FSAR. If any materials other than those listed in Appendix I to Section III of the ASME Code are used, provide the data called for under Appendix IV for approval of the new material. Information regarding material specifications provided in other documents or sections of the FSAR should be referenced in this section. Mechanical and physical properties of these materials should be addressed. The effects of radiation on these materials, where applicable, should be described.

C.I.5.3.1.2 *Special Processes Used for Manufacturing and Fabrication*

Describe the manufacture of the product forms and methods used to fabricate the vessel or any of its applicable attachments and appurtenances. Discuss any special or unusual processes used, and show that they will not compromise the integrity of the reactor vessel.

C.I.5.3.1.3 *Special Methods for Nondestructive Examination*

Describe in detail all special procedures for detecting surface and internal discontinuities, with emphasis on procedures that differ from those in Section III of the Code. Pay particular attention to calibration methods, instrumentation, method of application, sensitivity, reliability, reproducibility, and acceptance standards.

C.I.5.3.1.4 *Special Controls for Ferritic and Austenitic Stainless Steels*

Making appropriate references to Chapter 5 of the FSAR, describe controls on welding, composition, heat treatments, and similar processes covered by regulatory guides to verify that these recommendations or equivalent controls are employed. Include controls for abrasive work (e.g., grinding) on austenitic stainless steel. The following guidance should be addressed:

- Regulatory Guide 1.34, “Control of Electroslag Weld Properties”
- Regulatory Guide 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants”

- Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"
- Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel"
- Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel"
- Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility"
- Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials"
- Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"
- NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2

C.I.5.3.1.5 ***Fracture Toughness***

Describe the fracture testing and acceptance criteria specified for materials of the reactor vessel and appurtenances thereto. In particular, describe how the toughness requirements of Appendix G to 10 CFR Part 50 will be met. Specify the maximum nil-ductility reference temperature to which the ferritic materials of the reactor vessel will be fabricated. Identify ITAAC which will be completed to verify that these ferritic reactor vessel materials met these specifications.

C.I.5.3.1.6 ***Material Surveillance***

Describe the material surveillance program in sufficient detail to provide assurance that the program meets the requirements of Appendix H to 10 CFR Part 50. Describe the method for calculating neutron fluence for the reactor vessel beltline and the surveillance capsules. Because the material surveillance program is an operational program, as discussed in SECY-05-0197, the program and its implementation must be described with sufficient scope and level of detail to enable the NRC staff to make a reasonable assurance finding regarding on its acceptability. In particular, address the following topics:

(1) basis for selection of material in the program

(2) number and type of specimens in each capsule

(3) number of capsules and proposed withdrawal schedule comply with ASTM E-185, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials referenced to 10 CFR Part 50, Appendix H

(4) neutron flux and fluence calculations for vessel wall and surveillance specimens and conformance with the guidance of Regulatory Guide 1.190

(5) expected effects of radiation on vessel wall materials and basis for estimation

(6) location of capsules, method of attachment, and provisions to ensure that capsules will be retained in position throughout the vessel lifetime

C.I.5.3.1.7 ***Reactor Vessel Fasteners***

Describe the materials and design for the stud bolts, washers, nuts, and other fasteners for the reactor vessel closure. Include sufficient detail regarding materials property requirements, nondestructive evaluation procedures, lubricants or surface treatments, and protection provisions to show that the specifications of SRP 3.13, "Threaded Fasteners" and the recommendations of Regulatory Guide

1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” or equivalent measures, are followed. In the FSAR, describe the mechanical property and fracture toughness tests which will be performed to demonstrate that the materials from which these fasteners are fabricated conform to these recommendations or their equivalent. Identify any ITAAC which will be completed to verify that materials from which these fasteners are constructed met these specifications.

C.I.5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper Shelf Energy Data and Analyses

This section of the FSAR should describe the bases for setting operational limits on pressure and temperature for the RCPB during any condition of normal operation, including anticipated operational occurrences, and hydrostatic tests. In addition, this discussion should provide detailed assurance that Appendices G and H to 10 CFR Part 50 and 10 CFR Part 50.61 (PWRs only) will be complied with throughout the life of the plant.

C.I.5.3.2.1 *Limit Curves*

Describe how pressure-temperature limit curves for (1) pre-service system hydrostatic tests; (2) inservice leak and hydrostatic tests; (3) normal operation, including heat up and cool down; and (4) reactor core operation will be developed.

If procedures or criteria other than those recommended in the ASME Boiler and Pressure Vessel Code are used, show that equivalent safety margins are provided. Describe the bases used to determine these limits and provide typical curves with temperatures relative to the RT_{NDT} of the limiting material (as defined in paragraph NB-2331 of Section III of the ASME Code).

Based on material properties, including initial nil-ductility reference temperature, material chemical composition, etc., to which reactor vessel ferritic materials will be procured, demonstrate how pressure-temperature limits which meet the requirements of 10 CFR Part 50, Appendix G can be met for the licensed life of the facility. Describe the bases used for the prediction, and indicate the extent to which the recommendations of Regulatory Guide 1.99 are followed.

Describe procedures that will be used to update these limits during service. Address radiation effects and the extent to which the recommendations of Regulatory Guide 1.190 are followed.

C.I.5.3.2.2 *Operating Procedures*

Describe how operating procedures, which will ensure that the pressure-temperature limits in Section 5.3.2.1 of the FSAR will not be exceeded during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, will be developed. The FSAR should include a commitment that plant operating procedures will ensure that the pressure-temperature limits identified in Section 5.3.2.1 will not be exceeded during any foreseeable upset condition.

C.I.5.3.2.3 *Pressurized Thermal Shock (PWRs only)*

Discuss pressurized thermal shock (PTS). Using limiting material properties to which the ferritic reactor vessel materials will be procured, provide the calculational methods and assumptions and compare the projected values of reference temperature RT_{PTS} for reactor vessel beltline materials with the screening criteria in 10 CFR 50.61. If RT_{PTS} values are projected to exceed the PTS screening criterion before the expiration date of the operating license, provide safety analyses to support reactor operation.

C.I.5.3.2.4 *Upper Shelf Energy*

Specify minimum Charpy upper shelf energy values to which ferritic materials of the reactor vessel will be procured. Identify ITAAC which will be completed to verify that these ferritic reactor vessel materials meet or exceed these specifications. Provide projected Charpy upper shelf energy values at the expiration date of the operating license based on the methodology in Regulatory Guide 1.99 and demonstrate that beltline materials will satisfy the requirement of Appendix G (paragraph IV.A.1.a) to 10 CFR Part 50.

C.I.5.3.3 Reactor Vessel Integrity

This section of the FSAR should provide a summary of all information related to the integrity of the reactor vessel. Summarize the major considerations in achieving reactor vessel safety, and describe the factors contributing to the vessel's integrity. Also identify the reactor vessel designer and manufacturer, and describe their experience.

C.I.5.3.3.1 *Design*

Provide a brief description of the reactor vessel design, preferably with a schematic, including materials, construction features, fabrication methods, and inspections. Summarize applicable design codes and bases. Reference other sections of the FSAR as appropriate.

C.I.5.3.3.2 *Materials of Construction*

Identify the reactor vessel materials, including weld materials, and describe any special requirements to improve their properties or quality. Emphasize the reasons for selection, and provide assurance of suitability.

C.I.5.3.3.3 *Fabrication Methods*

Identify the reactor vessel fabrication methods, including forming, welding, cladding, and machining. Describe the service history of vessels constructed using these methods and the vessel supplier's experience with the procedures.

C.I.5.3.3.4 *Inspection Requirements*

Summarize the inspection test methods and requirements, paying particular attention to the level of initial integrity. Describe any methods that are in addition to the minimum requirements established in Section III of the ASME Code.

C.I.5.3.3.5 *Shipment and Installation*

Summarize the means used to protect the vessel so that its as-manufactured integrity will be maintained during shipment and site installation. Reference other FSAR sections as appropriate.

C.I.5.3.3.6 *Operating Conditions*

Summarize the operational limits that will be specified to ensure vessel safety. Provide a basis for concluding that vessel integrity will be maintained during the most severe postulated transients and PTS events at PWRs. Reference other FSAR sections as appropriate.

C.I.5.3.3.7 *Inservice Surveillance*

Summarize the inservice inspection and material surveillance programs, and explain their adequacy relative to the requirements of Appendix H to 10 CFR Part 50 and Section XI of the ASME Code. Reference Sections C.I.5.2.4 and C.I.5.3.1 as appropriate.

C.I.5.3.3.8 ***Threaded Fasteners***

Summarize the programs for ensuring the integrity of bolting and threaded fasteners and their adequacy. Reference Section C.I.3.13 as appropriate.

C.I.5.4 Component and Subsystem Design

This section of the FSAR should provide information regarding performance requirements and design features to ensure overall safety of the various components and subsystems within or allied with the reactor coolant system. Because these components and subsystems differ for various types and designs of reactors, the components and subsystems are not assigned specific subsection numbers. The FSAR should contain a separate subsection (numbered 5.4.1 through 5.4.x) for each principal component or subsystem. Each subsection should present the design bases, description, evaluation, and necessary tests and inspections for the component or subsystem, including radiological considerations from the viewpoint of how radiation affects operation and the viewpoint of how radiation levels affect the operators and capabilities of operation and maintenance. Appropriate details regarding the mechanical design should be described in Sections C.I.3.7, C.I.3.9, and C.I.5.2.

The following subsections identify components and subsystems that should be discussed, and the corresponding information that should be provided. As appropriate to the specific reactor type and design, certain subsections are "not applicable" and additional subsections are necessary to address other components and subsystems (e.g., core makeup tanks, automatic depressurization system valves, passive residual heat removal heat exchanger, isolation condenser system, gravity-driven cooling system, passive containment cooling system).

C.I.5.4.1 Reactor Coolant Pumps

Provide the reactor coolant pump design bases, description, evaluations, and tests and inspections. Discuss the provisions taken to preclude rotor over speeding of the reactor coolant pumps in the event of a design-basis loss-of-coolant accident (LOCA).

C.I.5.4.1.1 ***Pump Flywheel Integrity (PWRs only)***

Provide explicit information to demonstrate how the reactor coolant pump (RCP) flywheel design complies with the design criteria of GDC 1, "Quality Standards and Records," and GDC 4, "Environmental and Dynamic Effects Design Bases," as specified in Appendix A to 10 CFR Part 50. Discuss the extent to which the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," are followed with respect to the design, testing, analysis, Pre-service inspection, and inservice inspection of the RCP flywheels. Identify any particular exceptions or alternatives to the regulatory positions and criteria in Regulatory Guide 1.14 and, if applicable, discuss and justify how a particular exception or alternative will provide an acceptable level of quality and safety and will ensure continued compliance with GDC 4.

C.I.5.4.2 Steam Generators (PWRs only)

Provide estimates of design limits for radioactivity levels in the secondary side of the steam generators during normal operation, and discuss the bases for those estimates. Discuss the potential effects of tube ruptures.

Provide the steam generator design criteria to prevent unacceptable tube damage from flow-induced vibration and cavitation. Reference the information provided in Section C.I.3.9.3. Include the following specific information:

(1) design conditions and transients that will be specified in the design of the steam generator tubes and the operating condition category selected (e.g., upset, emergency, or faulted) that defines the allowable stress intensity limits to be used and the justification for this selection

(2) extent of tube-wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined above under the postulated condition of a design-basis pipe break in the RCPB or a break in the secondary piping during reactor operation

C.I.5.4.2.1 ***Steam Generator Materials***

Discuss the design of the steam generator, including, (1) the selection, processing, testing, and inspection (during fabrication/processing) of the materials used to fabricate the steam generator; (2) fracture toughness of the ferritic materials used in the steam generator; (3) design provisions for limiting the susceptibility of the steam generator to degradation and/or corrosion; (4) compatibility of materials with the primary (reactor) and secondary coolant; and (5) the provisions for accessing the secondary side of the steam generator for maintenance and cleaning.

Address the following considerations:

(1) Selection and Fabrication of Materials. Making appropriate references to Section C.I.5.2.3, provide information on the selection and fabrication of materials for components of the steam generators, including tubing, tube sheet, channel head casting or plate, tube sheet and channel head cladding, forged nozzles, shell pressure plates, access plates (manway and handhole), bolting, and threaded fasteners. List the Code Cases used in material selection. Provide technical justification for any Code Cases not listed in Regulatory Guide 1.84.

(2) Provide information on the fracture toughness properties of ferritic materials, making appropriate references to Section C.I.5.2.3. Provide sufficient information on materials for Class 1 components to show compliance with the requirements of Article NB-2300 and Appendix G to Section III of the ASME Code. For Class 2 materials, provide sufficient information to show compliance with the requirements of Article NC-2300 of Section III of the Code. Address welding qualification, fabrication, and inspection during manufacture and assembly in conformance with the requirements of Sections III and IX of the ASME Code.

(3) Steam Generator Design. Provide information on those aspects of design that may affect the performance of steam generator materials. Describe methods used to avoid extensive crevice areas where the tubes pass through the tube sheet and tubing supports. Describe the corrosion allowance for steam generator materials. Identify the method used to fasten tubes to the tube sheet and show that it meets the requirements of Sections III and IX of the ASME Code. Include the extent of tube expansion and the methods of expansion used. Describe the heat treatment of the steam generator tube material and the design of the support structures.

(4) Compatibility of Steam Generator Tubing with the Primary and Secondary Coolant. Provide information on the compatibility of steam generator tubing with both the primary and secondary coolant. Describe the methods used in monitoring and maintaining the chemistry of the primary and secondary coolant within the specified ranges.

(5) Accessing the Secondary Side of the Steam Generator. Describe the design provisions for removing surface deposits, sludge, loose parts (foreign objects), and excessive corrosion products from the secondary side of the steam generator.

C.I.5.4.2.2 ***Steam Generator Program***

Describe provisions in the design of the primary and secondary side of the steam generator that permit implementation of a steam generator tube integrity program. Describe the elements of the steam generator tube integrity program.

Address the following considerations:

(6) Steam Generator Design. Describe the design provisions for permitting access to both the primary and secondary side of the steam generator. Discuss the extent to which each tube is accessible for periodic inspection, testing, and repair (including plugging and stabilizing) using currently available methods and techniques (which are capable of finding the forms of degradation that may affect the tube throughout its service life). Discuss the extent to which secondary side internals that can affect tube integrity may be accessed. Describe design provisions for inspecting for and removing loose parts (foreign objects) from the steam generator. Describe design provisions for limiting the introduction of loose parts into the steam generator.

(7) Steam Generator Program. Describe the elements of the steam generator program and the extent to which they are consistent with the steam generator program requirements provided in Revision 3.1 of the Standard Technical Specifications. Discuss the method for determining tube repair criteria. Describe the scope and extent of the pre-service inspection of the steam generator tubes.

(8) Technical Specifications. Describe the steam generator tube inspection and reporting requirements to be adopted into the Technical Specifications (including the limiting conditions for operation, surveillance requirements, and primary-to-secondary leakage limits). Discuss the extent to which there are any potential conflicts (i.e., differences) between the Technical Specifications and Article IWB-2000 of Section XI of the ASME Code [refer to 10 CFR 50.55a(b)(2)(iii)].

C.I.5.4.3 **Reactor Coolant Piping**

Provide an overall description of the reactor coolant piping system with detailed information on the criteria, methods, and materials, and include appropriate references to Sections C.I.3 and C.I.5.2.3. Include the design, fabrication, and operation provisions to control those factors that contribute to stress corrosion cracking.

C.I.5.4.4 **[Reserved]**

C.I.5.4.5 **[Reserved]**

C.I.5.4.6 **Reactor Core Isolation Cooling System (BWRs only)**

The proposed pre-operational test program should be discussed. The discussion should verify test objectives, methods of testing, and test acceptance criteria. [Note: For the design of the Economic Simplified Boiling-Water Reactor (ESBWR), this subsection may address the isolation condenser system, in lieu of the reactor core isolation cooling system, and include analogous detailed information and reference to Section C.I.6.3, “Emergency Core Cooling System,” as appropriate.]

C.I.5.4.6.1 ***Design Bases***

Provide a summary of the reactor core isolation cooling (RCIC) system. Discuss the RCIC system design bases and criteria for both the steam side and pump side and include the following considerations:

(9) compliance with respect to GDCs 4, 5, 29, 33, 34, and 54

(10) reliability and operability requirements, including the bases for manual operations required to operate the system

(11) design for operation following a loss of offsite power event and compliance with 10 CFR 50.63 regarding station blackout events by conformance with Regulatory Guide 1.155, "Station Blackout"

(12) design bases for protecting the RCIC system from physical damage, including the bases for the RCIC system support structure and protection against incidents that could jointly fail the RCIC and high-pressure core spray (HPCS)

C.I.5.4.6.2 ***System Design***

(1) Piping and Instrumentation Diagrams. Provide a description of the RCIC system, including piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation. Provide a complete description of component interlocks, as well as a diagram showing temperatures, pressures, and flow rates for RCIC operation.

(2) Equipment and Component Descriptions. Describe each component of the system. Identify the significant design parameters for each component. State the design pressure and temperature of components for various portions of the system, and explain the bases for their selection.

(3) Applicable Codes and Classifications. Identify the applicable industry codes and classifications for the system design.

(4) System Reliability Considerations. Discuss provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates.

(5) Manual Actions. Discuss all manual actions required by an operator in order for the RCIC system to operate properly, assuming all components are operable. Identify any actions that are required to be taken from outside the control room. Repeat this discussion for the most limiting single failure in the combined RCIC and HPCS system.

C.I.5.4.6.3 ***Performance Evaluation***

Provide an evaluation of the ability of the RCIC system to perform its function. Describe the analytical methods used, and clearly state all assumptions.

C.I.5.4.7 **<u>Residual Heat Removal System</u>**

The proposed pre-operational test program should be discussed. The discussion should verify test objectives, methods of testing, and test acceptance criteria.

C.I.5.4.7.1 ***Design Bases***

Provide a summary description of the residual heat removal (RHR) system. Discuss the design bases, including the following considerations:

(6) Design bases with respect to GDCs 2, 4, 5, 19, and 34.

(7) Functional design bases, including the time required to reduce the reactor coolant system (RCS) temperature to approximately 100 EC (212 EF), and to a temperature that would permit refueling. Present the design-basis times for the case where the entire RHR system is operable, as well as the case with the most limiting single failure in the RHR system.

(8) Design bases for the isolation of the RHR system from the RCS. Discuss the isolation design bases, including any interlocks that are provided. Discuss the design bases regarding prevention of RHR pump damage in event of closure of the isolation valves.

(9) Design bases of the RHR system for prevention of interfacing systems loss-of-coolant accident (ISLOCA).

(10) Design bases for the pressure relief capacity of the RHR system. Discuss the design bases and considerations for limiting transients, equipment malfunctions, and possible operator errors during plant startup and cool down when the RHR system is not isolated from the RCS.

(11) Design bases for reliability and operability requirements. Describe the design bases regarding the manual actions required to operate the system with emphasis on any operations that cannot be performed from the control room in the event of a single failure. Describe protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, and redundancy of instrumentation. Describe protection against valve motor flooding and spurious single failures.

(12) Design bases established to protect the RHR system from physical damage. Discuss the design bases for the RHR system support structure and for protection against incidents and accidents that could render redundant components inoperative (e.g., fires, pipe whip, internally generated missiles, loss-of-coolant accident loads, seismic events).

(13) Design bases of the RHR system for shutdown and mid-loop operations.

(14) Design bases of the RHR system relief valves for low-temperature over pressure protection, if applicable.

(15) For passive core cooling system PWR designs with the passive residual heat removal heat exchanger (PRHRHX) and the active RHR system designated as a non-safety-related system for defense-in-depth functions, provide an evaluation in accordance with the process of “regulatory treatment of non-safety systems” to determine necessary regulatory oversight for the RHR system.

C.I.5.4.7.2 ***System Design***

(16) Schematic Piping and Instrumentation Diagrams. Provide a description of the RHR system, including piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation. Provide a description of component interlocks. Provide a mode diagram showing temperatures, pressures, and flow rates for each mode of RHR operation (for example, in a BWR, the RCIC condensing mode).

(17) Equipment and Component Descriptions. Describe each component of the system. Identify the significant design parameters for each component. State the design pressure and temperature of components for various portions of the system, and explain the bases for their selection. Provide pump characteristic curves and pump power requirements. Specify the available and required net positive suction head for the RHR pumps. Describe heat exchanger characteristics, including design flow rates, inlet and outlet temperatures for the cooling fluid and for the fluid being cooled, the overall heat transfer coefficient, and the heat transfer area. Identify each component of the RHR system that is also a portion of some other system [e.g., the emergency core cooling system (ECCS)].

(18) Control. State the RHR system relief valve capacity and settings, and state the method of collecting fluids discharged through the relief valve. Describe provisions with respect to the control circuits for motor-operated isolation valves in the RHR system, including consideration of inadvertent actuation. Include discussions of the controls and interlocks for these values (e.g., intent of IEEE Std. 279-1971), considerations for automatic valve closure (e.g., RCS pressure exceeds design pressure of residual heat removal system), valve position indications, and valve interlocks and alarms.

(19) Applicable Codes and Classifications. Identify the applicable industry codes and classifications for the system design.

(20) System Reliability Considerations. Discuss provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates (e.g., redundancy and separation of components and power sources).

(21) Manual Actions. Discuss all manual actions required to be taken by an operator in order for the RHR system to operate properly with all components assumed to be operable. Identify any actions that are required to be taken from outside the control room. Repeat this discussion for the most limiting single failure in the RHR system.

C.I.5.4.7.3 ***Performance Evaluation***

Provide an evaluation of the ability of the RHR system to reduce the temperature of reactor coolant at a rate consistent with the design basis (C.I.5.4.7.1).

Describe the analytical methods used and clearly state all assumptions. Provide curves showing the reactor coolant temperature as a function of time for the following cases:

(1) All RHR system components are operable.
(2) The most limiting single failure has occurred in the RHR system.

C.I.5.4.8 Reactor Water Cleanup System (BWRs only)

C.I.5.4.8.1 ***Design Bases***

Provide the design objectives and design criteria for the reactor water cleanup system (RWCS), in terms of (1) maintaining reactor water purity within the guidelines of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling-Water Reactors," (2) providing system isolation capabilities to maintain the integrity of the RCPB, and (3) precluding liquid poison removal when the poison is required for reactor shutdown. Describe how the requirements of 10 CFR Part 50 will be implemented, and indicate the extent to which the recommendations of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Guide 1.29, "Seismic Design Classification," will be followed.

C.I.5.4.8.2 ***System Description***

Describe each component and its capacity. Indicate the processing routes and the expected and design flow rates. Describe the instrumentation and controls to (1) isolate the system to maintain the RCPB, (2) isolate the system in the event the liquid poison system is needed for reactor shutdown, and (3) monitor, control, and annunciate abnormal conditions concerning the system temperature and differential pressure across filter/demineralizer units and resin strainers. Indicate the means to be used for "holding" filter/demineralizer beds intact if system flow is reduced or lost. Describe control features to prevent inadvertent opening of the filter/demineralizer backwash valves during normal operation. Describe the resin transfer system and indicate the provisions to ensure that transfers are complete and that crud traps in transfer lines are eliminated. For systems using other than filter/demineralizer units, provide appropriate information. Indicate the routing and termination points of system vents. Provide piping and instrumentation diagrams indicating system interconnections and seismic and quality group interfaces.

C.I.5.4.8.3 ***Performance Evaluation***

Provide the design bases for the system capacity, and discuss the system's capability to maintain acceptable reactor water purity for normal operation, including anticipated operational occurrences (e.g., reactor startup, shutdown refusing, condensate demineralizer breakthrough, equipment downtime). Indicate any reliance on other plant systems to meet the design objectives (e.g., liquid radwaste system). Present the design criteria for components and piping in terms of temperature, pressure, flow, or volume capacity. Provide the seismic design and quality group classifications for components and piping. Discuss the capability of the non-regenerative heat exchanger to reduce the process temperature to a level low enough to be compatible with the cleanup demineralizer resins in the event that there is no flow return to the reactor system.

C.I.5.4.9 [Reserved]

C.I.5.4.10 [Reserved]

C.I.5.4.11 <u>Pressurizer Relief Tank (PWRs only)</u>

C.I.5.4.11.1 *Design Bases*

Describe the design bases for the pressurizer relief tank system, including provisions for compliance with GDC 2 (meeting the guidelines of Regulatory Guide 1.29, "Seismic Design Criteria") and GDC 4. Describe the maximum step load and the consequent steam volume that the pressurizer relief tank must absorb, as well a the maximum heat input that the volume of water in the tank must absorb under any normal conditions or anticipated operational occurrences. Include information for (1) relief valve discharge to the tank, and (2) combined relief and safety valves discharge to the tank.

C.I.5.4.11.2 *System Description*

Provide a description of the system, including the tank, the piping connections from the tank to the loop seals of the pressurizer relief and safety valves, the relief tank spray system and associated piping, the nitrogen supply piping, and the piping from the tank to the cover gas analyzer and the reactor coolant drain tank. Provide a piping and instrumentation diagram and a drawing of the pressurizer relief tank.

C.I.5.4.11.3 *Performance Evaluation*

Demonstrate that the system, including the tank, is designed to handle the maximum heat load, and the tank design pressure and temperature are adequate. Present the results of a failure modes and effects analysis to demonstrate that the auxiliary systems serving the tank can meet the single-failure criterion without compromising safe plant shutdown. Identify the tank rupture disk and relief valve capacities, and show that their relief capacity is at least equal to the combined capacity of the pressurizer relief and safety valves.

C.I.5.4.11.4 *Instrumentation*

Discuss the instrumentation and controls for the pressurizer relief tank and associated piping.

C.I.5.4.12 <u>Reactor Coolant System High Point Vents</u>

C.I.5.4.12.1 *Design Bases*

Provide a summary of the reactor coolant system high point vents system, and discuss the design bases and criteria. Describe compliance with the provisions of 10 CFR 50.34(f)(2)(vi), 50.44, 50.46, 50.49, and 50.55a, and GDCs 1, 14, 17, 19, 30, 34, and 36.

C.I.5.4.12.2 *System Design*

Provide a description of the vent system, including its location, size, discharge capacity, functions, and discharge area(s). Provide piping and instrumentation diagrams showing all components, piping, and instrumentation and controls. Describe electrical power supplied from emergency buses. Describe operability from the control room and system instrumentation. Identify information available to the operator for initiating and terminating system operation.

C.I.5.4.12.3 ***Performance Evaluation***

Provide an evaluation of the vent system's capability to remove non-condensible gases from the primary coolant system with a minimal probability of inadvertent or spurious actuation. Evaluate vent system operation, including procedures that address (1) when venting is/is not needed, (2) method to determine the size of a non-condensible bubble, (3) initial conditions for venting, (4) requisite instrumentation, and (5) operator actions.

C.I.5.4.13 **[Reserved]**

C.I.5.4.14 **[Reserved]**