C.I.10 Steam and Power Conversion System

Chapter 10 of the final safety analysis report (FSAR) should provide information concerning the plant steam and power conversion system. For purposes of this chapter, the steam and power conversion system includes the following:

- the steam system and turbine generator units of an indirect cycle reactor plant, as defined by the secondary coolant system
- the steam system and turbine generator units in a direct-cycle plant, as defined by the system extending beyond the reactor coolant system isolation valves

This section should describe the secondary plant (steam and power conversion system), emphasizing those aspects of the design and operation that affect or could potentially affect the reactor and its safety features or contribute toward the control of radioactivity. The information provided should show the capability of the system to function without compromising (directly or indirectly) the safety of the plant, under both normal operating and transient situations. In addition, beginning with Section C.I.10.2 and for the other sections that follow, include a discussion of how the system design meets the applicable regulatory requirements and is consistent with the applicable regulatory guidance.

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Where appropriate, this chapter should summarize the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems; Chapters 11 and 12 of the FSAR should present this information in more detail.

C.I.10.1 *Summary Description*

The applicant should provide a summary description indicating principal design features of the steam and power conversion system. In addition, the applicant should provide an overall system flow diagram and a summary table of the important design and performance characteristics, including a heat balance at rated power and at stretch power and indicate safety-related system design features.

C.I.10.2 *Turbine Generator*

C.I.10.2.1 Design Bases

This section should describe the turbine generator system (TGS) equipment design and design bases, including the performance requirements under normal, upset, emergency, and faulted conditions. It should also describe the intended mode of operation (base loaded or load following), functional limitations imposed by the design or operational characteristics of the reactor coolant system (e.g., the rate at which the electrical load may be increased or decreased with and without reactor control rod motion or steam bypass), and design codes to be applied.

The applicant should provide the seismic design criteria, the bases for the chosen criteria, and the seismic and quality group classifications for TGS components, equipment, and piping. The applicant may reference the seismic and quality group classifications provided in FSAR Section 3.2.

The applicant should also describe how the plant will meet the requirements of General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50), with respect to the protection of structures, systems, and components important to safety from dynamic effects such as turbine missiles.

C.I.10.2.2 Description

Describe the TGS, associated equipment (including moisture separation), use of extraction steam for feedwater heating, and control functions that could influence operation of the reactor coolant system. In addition, provide piping and instrumentation diagrams (P&IDs) and layout drawings that show the general arrangement of the TGS and associated equipment with respect to essential safety-related structures, systems, and components[1]. Include details related to construction materials of TGS components.

Describe the turbine generator control and overspeed system in detail, including redundancy and diversity of controls, type(s) of control utilized, overspeed setpoints, and valve actions required for each setpoint. Describe how this system will preclude an unsafe turbine overspeed and how the system will function in conjunction with support systems, subsystems, control systems, alarms, and trips for all abnormal conditions, including a single failure of any component or subsystem. Describe the inservice inspection and operability assurance program for valves essential to overspeed protection.

Describe the types, locations, valve closure times of the main steam stop, control, reheat stop, intercept, and extraction steam valve arrangements and of associated piping arrangements.

Describe any preoperational and startup tests.

Provide an evaluation of the TGS and related steam handling equipment, including a summary of the anticipated operating concentrations of radioactive contaminants in the system, radiation levels associated with the turbine components and resulting shielding requirements, and the extent of access control necessary based on radiation levels and shielding provided. Chapters 11 and 12 of the FSAR should provide details of the radiological evaluation, as appropriate.

If safety-related systems or portions of systems are located close to the TGS, describe the physical layout of the TGS with respect to precautions taken to protect against the effects of either high- and moderate-energy TGS piping failures or failure of the connections from the low-pressure turbine section of the main condenser.

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C.I.10.2.3 Turbine Rotor Integrity

Provide information to demonstrate the structural integrity of turbine rotors and the protection against damage to a safety-related component due to failure of a turbine rotor that produces a high-energy missile.

C.I.10.2.3.1 ***Materials Selection***

Describe the materials specifications, chemical analysis, fabrication history and techniques, coating processes, and nondestructive examinations during the fabrication process of the turbine rotor and rotor forgings, paying particular attention to items affecting metallurgical stability. List the materials properties of the rotor, including yield strength and the stress-rupture properties of the high-pressure rotor material. Describe the methods of obtaining these properties, including the procedures to minimize flaws. If plant-specific data and information are unavailable at the time of the COL application, representative or bounding data and information may be submitted for staff review as part of the COL application. The COL applicant should submit the plant-specific data and information to the staff at a pre-determined time agreed

[1] See Regulatory Guide 1.117, "Tornado Design Classification," Section C, for guidance on what safety-related SSCs are considered to be essential.

upon by the both parties. The applicant may need to work with the NRC staff during the COL application review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

C.I.10.2.3.2 ***Fracture Toughness***

Describe the criteria used to ensure protection against brittle failure of turbine rotors. Provide a detailed discussion of the materials' fracture toughness, ductile-brittle transition temperatures (fracture appearance transition temperature or nil-ductility transition temperature), and minimum operating temperatures. Describe the fracture toughness and Charpy V-notch test programs. If a fracture mechanics approach is used, describe the analytical method and the key assumptions made, including all supporting references. If plant-specific data and information are unavailable at the time of the COL application, representative or bounding data and information may be submitted for staff review as part of the COL application. The COL applicant should submit the additional information to the staff at a pre-determined time agreed upon by the both parties. The applicant may need to work with the NRC staff during the COL application review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

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C.I.10.2.3.3 ***Preservice Inspection***

Describe the preservice inspection procedures and acceptance criteria to demonstrate the integrity of the rotors. If the plant-specific information is unavailable at the time of the COL application, the representative information may be submitted for staff review as part of the COL application. The COL applicant should submit the plant-specific information to the staff at a pre-determined time agreed upon by both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

C.I.10.2.3.4 ***Turbine Rotor Design***

Describe how the turbine rotor assembly is designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. Provide the following design information for low-pressure rotors:

- design overspeed conditions, turbine trip speed, and normal operating speed
- allowable stresses, including the tangential stress due to centrifugal loads, interference fit, and thermal gradients at the bore region at normal speed and design overspeed
- maximum tangential and radial stresses and their location in the rotor
- temperature distributions in the rotor
- diagrams of the rotor and how blades or buckets are attached to the rotor

C.I.10.2.3.5 ***Inservice Inspection***

Describe the inservice inspection program (including both the baseline and inservice phases) for the turbine assembly and the inspections and tests of the main steam stop and control valves and the reheat stop and intercept valves. Describe the types of inspections and inspection techniques, areas to be inspected, frequencies of inspection, and acceptance criteria.

If the plant-specific information are unavailable at the time of the COL application, the representative information may be submitted for staff review as part of the COL application. The COL applicant should submit the plant-specific information to the staff at a pre-determined time agreed upon by both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

C.I.10.3 *Main Steam Supply System*

The main steam supply system (MSSS) consists of the components, piping, and equipment that function to transport steam from the nuclear steam supply system to the power conversion system and various safety-related and nonsafety-related auxiliaries. For the boiling-water reactor (BWR) direct cycle plant, the MSSS extends from the outermost containment isolation valves up to and including the turbine stop valves and includes connected piping of 6.4-centimeters (2.5-inches) nominal diameter and larger, up to and including the first valve that is either normally closed or is capable of automatic closures during all modes of reactor operation. For the pressurized-water reactor (PWR) indirect cycle plant, the MSSS extends from the connections to the secondary sides of the steam generators up to and including the turbine stop valves and includes the containment isolation valves, safety and relief valves, connected piping of 6.4-centimeters (2.5-inches) nominal diameter and larger, up to and including the first valves that are either normally closed or capable of automatic closure during all modes of operation, and the steamline to the auxiliary feedwater pump turbine.

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C.I.10.3.1 Design Bases

Describe the MSSS design and design bases, including performance requirements, environmental design bases, inservice inspection requirements, and design codes to be applied. Discuss the system's capability to dump steam to the atmosphere, if required. Include a description of steamlines to and from any feedwater turbines, if applicable.

Describe the design features incorporated to permit appropriate functional testing of system components important to safety. Describe the design features incorporated to ensure that the system will maintain its essential functions, as required, in the event of adverse environmental phenomena, certain pipe breaks, or loss of offsite power. Describe the design features incorporated to ensure that essential portions of the MSSS will function following design-basis accidents, assuming a concurrent single active component failure.

Describe design features and procedures implemented to minimize the potential for water hammer and relief valve discharge loads.

Provide the seismic design criteria, the bases for selection of the chosen criteria, and the seismic and quality group classifications for MSSS components, equipment, and piping. The applicant may reference the seismic and quality group classifications provided in FSAR Section 3.2.

In accordance with SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, for new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which main condenser holdup and plateout of fission products are credited in the analysis of design-basis accident radiological consequences, describe the seismic analysis performed to ensure that the main steam drain lines are capable of maintaining structural integrity during and after a safe-shutdown earthquake.

Describe how the plant will meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4; GDC 5, "Sharing of Structures, Systems, and Components"; and GDC 34, "Residual Heat Removal," of Appendix A to 10 CFR Part 50. In addition, indicate compliance with the regulations in 10 CFR 50.63, "Loss of All Alternating Current Power," and conformance with the guidance of Regulatory Guide 1.155, "Station Blackout," as they relate to the capability of the MSSS to cope with and recover from a station blackout of a specified duration. Also demonstrate conformance with guidance related to the design of the MSSS provided in Regulatory Guide 1.29, "Seismic Design Classification"; Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles"; and Regulatory Guide 1.117, "Tornado Design Classification. If the applicant does not follow this guidance, it should describe the specific alternative methods used.

C.I.10.3.2 Description

Describe the MSSS and main steamline piping. Provide P&IDs showing system components, including interconnected piping. On the P&IDs, indicate the physical division between the safety-related and nonessential portions of the system.

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C.I.10.3.3 Evaluation

Evaluate the design of the main steam system piping, including an analysis of the system's ability to withstand limiting environmental and accident conditions and provisions for permitting the performance of inservice inspections. The applicant may also reference the analysis of postulated high-energy line failure provided in FSAR Section 3.6.

C.I.10.3.4 Inspection and Testing Requirements

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Describe the inspection and testing requirements of the main steam system piping. Describe the proposed requirements for preoperational and inservice inspection of main steam piping, and inservice testing of steamline isolation valves. Reference other sections of the FSAR, as appropriate. If the plant-specific information are unavailable at the time of the COL application, the representative information may be submitted for staff review as part of the COL application. The COL applicant should submit the plant-specific information to the staff at a pre-determined time agreed upon by both parties. The applicant may need to work with the NRC staff during the review to agree on an appropriate method (e.g., ITAAC, license condition, FSAR update) to ensure that the as-built plant is consistent with the design reviewed during the licensing process.

C.I.10.3.5 Water Chemistry (PWR only)

Discuss the effect of the water chemistry chosen on the radioactive iodine partition coefficients in the steam generator and air ejector. Provide detailed information on the secondary-side water chemistry, including methods of treatment for corrosion control and proposed specification limits. Discuss methods for monitoring and controlling water chemistry.

C.I.10.3.6 Steam and Feedwater System Materials

In this section, provide the information indicated below on the materials used for American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code), Section III, Class 2 and 3 components, as defined in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." (Discuss Class 1 component materials in Chapter 5 of the FSAR.)

Describe how the plant will meet the regulatory requirements of 10 CFR 50.55a, "Codes and Standards"; GDC 1, "Quality Standards and Records," and GDC 35, "Emergency Core Cooling," of Appendix A to 10 CFR Part 50; and Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Indicate the extent of conformance with the guidance of Regulatory Guides 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"; Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility"; and Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." If the applicant does not follow this guidance, it should describe and justify the specific alternative methods used.

C.I.10.3.6.1 ***Fracture Toughness***

Indicate the degree of compliance with the test methods and acceptance criteria of the ASME Code Section III in Articles NC-2300 (Class 2) and ND-2300 (Class 3) for fracture toughness for ferritic materials used in Class 2 and 3 components. If the applicant does not follow this code, it should describe and justify the specific alternative methods used for NRC review and approval. For non-ASME Code components, provide expected plant-specific material property data such as chemistry, yield strength, fracture toughness data (KIC), Charpy V-notch energy, nil-ductility temperature, and fracture appearance transition temperature.

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C.I.10.3.6.2 ***Materials Selection and Fabrication***

Provide information on the materials selection and fabrication methods used for Class 2 and 3 components, including the following:

(1) Specify whether the materials used for the piping and components of the feedwater and main steam systems are consistent with Appendix I to Section III, Parts A, B, and C of Section II of the ASME Code or Regulatory Guide 1.84. For any material not included in the above, provide the data requested under Appendix IV to Section III of the ASME Code for approval of new materials. Justify the use of any such materials.

(2) For austenitic stainless steel components, indicate the extent of conformance with the guidance in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping" (Revision 2), issued January 1988, or Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as applicable, and describe and justify any specific alternative methods used. Provide a detailed discussion of the mitigation implemented in the design, materials selection, fabrication, and operation to reduce the susceptibility of components made of stainless steel and nickel-based materials to intergranular stress-corrosion cracking.

(3) Describe the cleaning and handling procedures for all Class 2 and 3 components. Indicate the extent of conformance with the guidance of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

(4) Indicate whether the preheat temperatures used for welding low-alloy steel are in accordance with Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." For carbon or low-alloy steel components, describe the controls placed on the welding procedures. For carbon steel materials, indicate whether the preheat temperatures are in accordance with Section III, Appendix D, Article D-1000, of the ASME Code.

(5) Describe the qualification procedures for welds in areas of limited accessibility. For all applicable components, indicate the extent of conformance with the guidance of Regulatory Guide 1.71 (i.e., assurance of the integrity of welds in locations of restricted direct physical and visual accessibility).

(6) Indicate that the nondestructive examination procedures and acceptance criteria used for the examination of tubular products conform to the provisions of the ASME Code, Section III, Paragraphs NC/ND-2550 through 2570.

(7) Develop a plant-specific preservice inspection and inservice inspection program that will include examinations of ASME Code and non-ASME Code components. These programs should reference the edition and addenda of the ASME Code Section XI used for selecting components subject to examination. Describe the components that are exempted from examination by the applicable code, and provide drawings or other descriptive information used for the examination. The applicant is responsible for ensuring the accessibility and inspectability of the subject components.

(8) When cast austenitic stainless steel materials are used, discuss the measures to be taken to ensure that these materials can be adequately inspected by volumetric methods as required in the inservice inspection program.

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For all of the above, if the applicant does not follow the recommended guidance, it should justify any deviations from the guidance and describe the specific alternatives used.

C.I.10.3.6.3 ***Flow-Accelerated Corrosion (Previously Referred to as Erosion/Corrosion)***

Describe the design features implemented to mitigate flow-accelerated corrosion (FAC), including the following:

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- utilization of FAC-resistant materials
- specification of an adequate corrosion allowance that accounts for the design life of the plant and that meets Section III of the ASME Code or ANSI/ASME B.31.1, "Power Piping," for non-ASME Code components
- implementation of piping design and layout considerations to minimize the FAC effects from fluid velocity, geometry effects such as bend locations, and flash points

Indicate the degree to which the plant design has implemented the recommendations of Electric Power Research Institute (EPRI) NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," issued April 1999.

Develop an erosion/corrosion (EC)/FAC monitoring program for carbon steel portions of the steam and power conversion system that contains water or wet steam.

(The terms flow-accelerated corrosion (FAC) and erosion/corrosion (EC) have often been used interchangeably because early cases of FAC (high-energy carbon steel piping failures) were initially attributed to EC. The NRC issued GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," issued May 1989, and the associated NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," issued April 1989, to address those piping failures, which are now recognized as FAC. FAC and EC are two distinct thinning mechanisms related to flow. FAC results from mass transfer and corrosion effects; EC results from mechanical and corrosion effects. Since FAC and EC are both related to flow effects, some licensees manage FAC as a subset of a comprehensive EC program. Computer programs

designed for FAC management (e.g., CHECWORKS) are unlikely to accurately model corrosion rates for other forms of flow-related thinning such as EC. The subject of this review area is FAC.)

C.I.10.4 *Other Features of Steam and Power Conversion System*

In this section, discuss each of the principal design features and subsystems of the steam and power conversion system. As these systems vary in number, type, and nomenclature for various plant designs, this regulatory guide does not assign specific subsection numbers to these systems. Thus, provide separate subsections (numbered C.I.10.4.1 through C.I.10.4.x) for each system, as appropriate. Provide the following information in each of these subsections:

(1) design bases (including design codes to be applied)

(2) system description

(3) system layout drawings, process flow diagrams, and P&IDs

(4) safety evaluation

(5) performance requirements for startup and normal operation

(6) inspections and periodic testing requirements, including preoperational and startup tests (reference Chapter 14 of the FSAR, as appropriate)

(7) instrumentation applications for each subsystem or feature

(8) seismic design criteria, the bases governing chosen criteria, and the seismic and quality group classifications for main system components, equipment, and piping (reference the seismic and quality group classifications provided in FSAR Section 3.2)

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The following paragraphs provide examples of subsystems and features that should be discussed, as appropriate to the individual plant, and identify specific information that the section should provide in addition to the items identified above.

C.I.10.4.1 <u>Main Condensers</u>

Describe the main condenser system, including the following elements:

- materials of construction
- methods used to reduce the probability of erosion/corrosion of tubes and components
- anticipated inventory of radioactive contaminants during power operation and shutdown
- design provisions to detect loss of condenser vacuum and to effect isolation of the steam source
- anticipated air leakage limits
- instrumentation and control features
- control functions that could influence operation of the primary reactor coolant or secondary systems
- potential for hydrogen buildup
- provisions for dealing with flooding from a complete failure of the main condenser and for protection of safety-related equipment from flooding as a result of failure of the condenser
- methods used to detect, control, and facilitate correction of the leakage of cooling water into the condensate

- methods used to detect radioactive leakage into or out of the system
- methods used to preclude accidental releases of radioactive materials to the environment in amounts in excess of established limits (Appendix B to 10 CFR Part 20, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage")

Describe the inventory of radioactive contaminants in the main condenser during power operation and during shutdown. Chapter 11 of the FSAR should provide details of the radiological evaluation. Describe the procedure to repair condensate leaks, the permissible cooling water inleakage, and the length of time the condenser may operate with inleakage without affecting the condensate/feedwater quality for safe reactor operation.

In accordance with SECY 93-087, for new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which main condenser holdup and plateout of fission products are credited in the analysis of design-basis accident radiological consequences, describe the seismic analysis performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining structural integrity during and after a safe-shutdown earthquake.

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Describe how the plant will meet the regulatory requirements of GDC 60, "Control of Releases of Radioactive Materials to the Environment," of Appendix A to 10 CFR Part 50, as they relate to minimizing excessive releases of radioactivity to the environment, maintaining acceptable condensate quality, and preventing flooding of areas housing safety-related equipment. Demonstrate conformance with the guidance of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," and Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." If not following this guidance, the applicant should describe and justify the specific alternative methods used.

C.I.10.4.2 Main Condenser Evacuation System

Describe the main condenser evacuation system design, design objectives, capacity, method of operation, and factors that influence gaseous radioactive material handling (e.g., system interfaces and potential bypass routes). Also describe anticipated release rates of radioactive materials, evaluation of the capability to limit or control loss of radioactivity to the environment, and control functions that could influence operation of the reactor coolant system. Specifically describe any design features that preclude the possibility of an explosion if the potential for explosive mixtures exists, as well as those design features incorporated to detect explosive gas mixtures and monitor radioactive materials in gaseous effluents from the main condenser evacuation system. FSAR Chapter 11should contain details of the radiological evaluation.

Describe how the plant will meet the regulatory requirements of GDC 60 and GDC 64, "Monitoring Radioactivity Releases," of Appendix A to 10 CFR Part 50, as they relate to controlling and monitoring releases of radioactive materials to the environment. Demonstrate compliance with 10 CFR 50.55a requirements for water- and steam-containing components. Indicate the extent of conformance with the guidance of Regulatory Guide 1.26. Also discuss "Standards for Steam Surface Condensers," issued by the Heat Exchanger Institute in 1970, as it relates to main condenser evacuation system components that may contain radioactive materials. If not following this guidance, the applicant should describe and justify the specific alternative methods used.

C.I.10.4.3 Turbine Gland Sealing System

Describe the turbine gland sealing system design, design objectives, method of operation, and factors that influence gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). Also include in the description identification of the source of noncontaminated steam, potential radioactivity leakage to the environment in the event of a malfunction, and the means to be used to monitor system performance. Describe design provisions used to control and monitor the release of radioactive materials from the seal condenser vent. Chapter 15 of the FSAR should evaluate the estimate of potential radioactivity leakage to the environment in the event of a malfunction of the turbine gland sealing system. Chapter 11 of the FSAR should provide details of the radiological evaluation.

Describe how the plant will meet the regulatory requirements of GDC 60 and 64 of Appendix A to 10 CFR Part 50, as they relate to controlling and monitoring releases of radioactive materials to the environment. Demonstrate conformance with the guidance of Regulatory Guide 1.26. If not following this guidance, the applicant should describe and justify the specific alternative methods used.

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C.I.10.4.4 Turbine Bypass System

Describe the turbine bypass system design, including the system capability to meet design criteria and environmental criteria. The evaluation of the turbine bypass system should include a failure analysis to determine the effect of equipment malfunctions on the reactor coolant system.

Describe how the plant will meet the regulatory requirements of GDC 4 and 34 of Appendix A to 10 CFR Part 50, as they relate to the integrity of safety-related components and residual heat removal capability. Indicate the extent of conformance with the guidance in Regulatory Guide 1.68 and Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." If not following this guidance, the applicant should describe and justify the specific alternative methods used.

For new BWR plants that do not incorporate a main steam isolation valve leakage control system and for which turbine bypass system holdup and plateout of fission products are credited in the analysis of design-basis accident radiological consequences, demonstrate conformance with the seismic analysis described in SECY 93-087.

C.I.10.4.5 Circulating Water System

Describe the circulating water system, including dependence on the system for cooling during shutdown, anticipated operational occurrences, accident conditions (e.g., loss of offsite power), capability to detect leaks and to secure the system quickly and effectively, effects of adverse environmental occurrences, and potential interaction of cooling towers, if any, with the plant structure. Discuss the methods used to control the circulating water chemistry, corrosion, and organic fouling, and their compatibility with system components and piping materials. Also discuss the potential for flooding of safety-related equipment due to the failure of a system component such as an expansion joint and include the interfaces of the circulating water system with other systems. Describe the design provisions implemented to prevent or detect and control this flooding and to annunciate abnormal and unsafe operating conditions. Reference Sections 2.4.11.5 and 2.4.11.6 of the FSAR, as appropriate.

Describe how the plant will meet the regulatory requirements of GDC 4 of Appendix A to 10 CFR Part 50, as they relate to design provisions implemented to accommodate the effects of discharging

water that may result from a failure of a component or piping of the system. Provide P&IDs and elevation drawings to support the design description.

C.I.10.4.6 Condensate Cleanup System

Describe the condensate cleanup system, including the fraction of condensate flow to be treated, purity requirements, and the basis for those requirements. The evaluation of the condensate cleanup system should include an analysis of demineralizer capacity and anticipated impurity levels, an analysis of the contribution of impurity levels from the secondary system to reactor coolant system activity levels, and an analysis of performance monitoring. Describe design features implemented to ensure that, in the event of condenser tube leaks, concentrations of chloride and other contaminants can be limited to allowable values until the condensate and feedwater systems are isolated. Demonstrate the compatibility of the materials of construction with service conditions and reactor water chemistry.

C.I.10.4.7 Condensate and Feedwater Systems

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Describe the condensate and feedwater systems, including the capability to supply adequate feedwater to the nuclear steam supply system, criteria for isolation from the steam generator or reactor coolant system, supply of condensate available for emergency purposes, and environmental design requirements. Describe the design considerations incorporated to minimize erosion/corrosion, referencing applicable guidance in GL 89-08 and EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping and Inspection Program and Guidelines," issued April 1985, as appropriate. Include an analysis of component failure and of the effects of equipment malfunction on the reactor coolant system and an analysis of detection and isolation provisions to preclude release of radioactivity to the environment in the event of a pipe leak or break and/or degradation of the integrity of safety-related equipment.

For PWRs, provide the following information with reference to fluid flow instabilities (e.g., water hammer, for steam generators using top feed):

(1) A description of normal operating transients that could cause the water level in the steam generator to drop below the sparger or cause the nozzles to uncover and allow steam to enter the sparger and feedwater piping.

(2) A summary of the criteria for routing or isometric drawings showing the routing of the feedwater piping system from the steam generators to the restraint that is closest, on the upstream side, to the feedwater isolation valve that is outside containment.

(3) A description of the piping system analyses, including any forcing functions, or the result of test programs performed to verify that uncovering of feedwater lines could not occur or that such uncovering would not result in unacceptable damage to the system. (Demonstrate conformance with guidance for water hammer prevention and mitigation, as found in NUREG-0927.)

For BWRs, provide a description of the feedwater nozzle design, inspection, and testing procedures, and system operating procedures incorporated to minimize nozzle cracking at low feedwater flow with reference to fluid flow instabilities.

Demonstrate conformance with the guidance in NUREG-0619 and GLs 80-95 and 81-11.

Demonstrate compliance with the requirements of GDC 5; GDC 44, "Cooling Water"; GDC 45, "Inspection of Cooling Water System"; and GDC 46, "Testing of Cooling Water System," of Appendix A to 10 CFR Part 50.

Demonstrate compliance with the requirements of GDC 2 and 4 of Appendix A to 10 CFR Part 50 and conformance with the associated guidance in Regulatory Guide 1.29 and BTP ASB 10-2, "Design Guidelines for Avoiding Water Hammer in Steam Generators," respectively.

If not following any of the above guidance, the applicant should describe and justify the specific alternative methods used.

C.I.10.4.8 <u>Steam Generator Blowdown System (PWR)</u>

C.I.10.4.8.1 *Design Bases*

Provide the design bases for the steam generator blowdown system (SGBS) in terms of its ability to maintain optimum secondary-side water chemistry in recirculating steam generators of PWRs during normal operation, including anticipated operational occurrences (e.g., main condenser inleakage, primary-to-secondary leakage). The design bases should include consideration of expected and design flows for all modes of operation (i.e., process and process bypass), process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis), and process instrumentation and controls for maintaining operations within established parameter ranges.

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C.I.10.4.8.2 *System Description and Operation*

Describe the SGBS and its components. Provide equipment general arrangement drawings, referencing pertinent information in FSAR Section 11.2, as appropriate. Discuss the operating procedures and the processing to be provided for all anticipated modes of operation, including system or process bypass, significant primary-to-secondary leakage, main condenser inleakage, and process sampling capabilities.

Discuss the specific instrumentation and controls provided to protect temperature-sensitive elements (e.g., demineralizer resins or reverse osmosis membranes) and to control flashing, liquid levels, and process flow through system components. Describe the radioactive waste treatment and process and effluent radiological monitoring aspects of the SGBS in Sections 11.2 through 11.5 of the FSAR.

C.I.10.4.8.3 *Safety Evaluation*

Discuss the interfaces between the SGBS and other plant systems. Identify and evaluate unusual design conditions that could lead to safety problems. Provide a failure mode and effects analysis of any interactions that may incapacitate safety-related equipment. Provide coolant chemistry specifications to demonstrate compatibility with primary-to-secondary system pressure boundary material. Include a description of the bases for the selected chemistry limits as well as a description of the secondary coolant chemistry program for steam generator blowdown samples.

C.I.10.4.9 <u>Auxiliary Feedwater System (PWR)</u>

C.I.10.4.9.1 *Design Bases*

Describe the design bases for the auxiliary feedwater system in terms of the safety-related functional performance requirements of the system, including the required pumping capacities of the pumps, diversity of power supplied to the system pumps and system control valves, capabilities of the pumps (i.e, head, flow) with respect to supply requirements of the steam generator, and the auxiliary feedwater supply capacity requirements for makeup during maximum hot standby conditions and for cold

shutdown of the facility following a reactor trip or accident condition. Describe the system's ability to withstand adverse environmental occurrences and the effects of pipe breaks, and the system's ability to perform its safety-related function in the event of a single malfunction, a failure of a component, the loss of a cooling source, a failure coincident with pipe breaks, environmental occurrences, and the loss of offsite power and/or the standby alternating current (ac) power system. Note that this section is only applicable to auxiliary feedwater systems that perform a safety function and it is not applicable to those plant designs where this is not the case.

Describe the design features implemented to ensure the following:

(1) System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from the effects of internally and externally generated missiles.

(2) The system has protection against the effects of pipe whip and jet impingement that may result from high- or moderate-energy piping breaks or cracks.

(3) Failure of nonessential equipment or components does not affect essential system functions.

(4) The system is capable of withstanding a single active failure.

(5) The system possesses diversity in motive power sources such that either of the assigned power sources (e.g., a system with an ac subsystem and a redundant steam direct current (dc) subsystem) can meet the system performance requirements.

(6) The system design precludes the occurrence of fluid flow instabilities (e.g., water hammer) in system inlet piping during normal plant operation or during upset or accident conditions.

(7) Suitable protection during abnormally high water levels (adequate flood protection considering the probable maximum flood) ensures functional capability.

(8) The system has the capability to detect, collect, and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.

(9) Provisions exist for operational testing.

(10) Instrumentation and control features verify that the system is operating in the correct mode.

(11) The system has the capability to automatically initiate auxiliary feedwater flow upon receipt of a system actuation signal.

(12) The system can manually initiate protective action by the auxiliary feedwater system, in accordance with the guidance of Regulatory Guide 1.62, "Manual Initiation of Protective Actions."

(13) The system design provides the capability to automatically terminate auxiliary feedwater flow to a depressurized steam generator and to automatically provide feedwater to the intact steam generator. Alternatively, if the applicant shows that the intact steam generator will receive the minimum required flow without isolation of the depressurized steam generator and that containment design pressure is not exceeded, then the system may rely on operator action to isolate the depressurized steam generator.

(14) The system has sufficient auxiliary feedwater flow capacity to allow the achievement of a cold shutdown (i.e., the system meets the minimum flow requirements for decay heat removal).

(15) Technical specifications ensure the continued system reliability during plant operation (i.e., the specifications state the limiting conditions for operation and surveillance testing requirements that are consistent with the Standard Technical Specifications).

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(16) The system design meets the generic short- and long-term recommendations identified in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," issued January 1980, and NUREG-0635, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants," issued January 1980 (all PWRs).

(17) A system reliability analysis has been performed, pursuant to Three Mile Island (TMI) Action Plan Item II.E.1.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980, and 10 CFR 50.34(f)(1)(ii);

(18) The system design meets the provisions of TMI Action Plan Item II.E.1.2 of NUREG-0737 regarding the automatic and manual initiation of the system and 10 CFR 50.62(c)(1) regarding the automatic initiation of the system on conditions indicative of an anticipated transient without scram.

(19) The system has the capability to permit operation at hot shutdown for at least 4 hours followed by cooldown to the residual heat removal cut-in temperature from the control room using only safety-grade equipment and assuming the worst-case single active failure, in accordance with Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System."

(20) The diversity and performance of the system with regard to the decay heat removal capability and capacity for station blackout events is in accordance with 10 CFR 50.63 requirements.

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C.I.10.4.9.2 ***System Description***

Describe the auxiliary feedwater system, including the location of components in the station complex. The description and associated system drawings should also include subsystems, system interconnections, cross-connections and interactions, components utilized, piping connection points, instrumentation and controls utilized, and system operations (i.e., system function during normal operations and the minimum functional conditions of the system in the event of pipe breaks, loss of main feedwater system, or loss of offsite power). Also state the maximum length of time the plant could do without normal feedwater and the minimum auxiliary feedwater flow rate required after this time period (i.e., pumps started and control valves open) for these conditions.

C.I.10.4.9.3 ***Safety Evaluation***

An evaluation of the capability of the auxiliary feedwater system should include (either in this section or by reference) a description of the features protecting the system and auxiliary supporting systems from postulated failures of high- and moderate-energy systems and the means by which the system is capable of withstanding the effects of site-related natural phenomena. Provide failure mode and effects analyses ensuring that the system meets minimum safety requirements, assuming a postulated pipe failure concurrent with a single active component failure in any system required to ensure performance of the auxiliary feedwater system. Perform an analysis for all modes of operation to demonstrate the capability of the system to preclude hydraulic instabilities (e.g., water hammer).

Perform an analysis to demonstrate the capability of the system to perform its safety function when subjected to a combination of environmental occurrences, environmental conditions, pipe breaks, and loss of power during normal and accident conditions. In addition, perform an analysis to demonstrate the capability of the system to perform its safety function utilizing diverse power sources, so as to ensure system operability without reliance on ac power.

Demonstrate compliance with the requirements of GDC 2, 4, 5, 19 ("Control Room"), 34, 44, 45, and 46 of Appendix A to 10 CFR Part 50. Demonstrate conformance with the associated guidance in Regulatory Guides 1.29 and 1.62, and Branch Technical Positions RSB 5-1 and ASB 10-1, "Design

Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

Demonstrate compliance with 10 CFR 50.63, as related to the design provisions for withstanding and recovering from a station blackout, as well as conformance with the applicable guidance in Regulatory Guide 1.155.

If not following any of the above guidance, the applicant should describe and justify the specific alternative methods used.

C.I.10.5 ***References***

10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

10 CFR 50.55a, "Codes and Standards."

10 CFR 50.63, "Loss of All Alternating Current Power."

Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

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Regulatory Guide 1.29, "Seismic Design Classification."

Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

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 1.62, "Manual Initiation of Protective Actions."

Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants."

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants."

Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

Regulatory Guide 1.117, "Tornado Design Classification."

Regulatory Guide 1.155, "Station Blackout."

Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

Branch Technical Position ASB 10-2, "Design Guidelines for Avoiding Water Hammer in Steam Generators."

Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System."

NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Rev. 2, January 1988.

NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking."

NUREG-0635, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants."

NUREG-0737, "Clarification of TMI Action Plan Requirements."

NUREG-0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants."

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NUREG-1344, “Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants”

Generic Letter 80-95, “Final Edition of NUREG-0619, ‘BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking.’”

Generic Letter 81-11, “BWR Feedwater Nozzle Cracking,” February 29, 1981.

Generic Letter 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,” January 25, 1988.

Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning,” May 1989.

SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” April 2, 1993.

ANSI Standard N45.2.1-1973, “Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants.”

ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers.

ANSI/ASME B.31.1, “Power Piping.”

EPRI NP-3944, “Erosion/Corrosion in Nuclear Plant Steam Piping and Inspection Program and Guidelines,” April 1985.

EPRI NSAC-2021-R2, “Recommendations for an Effective Flow-Accelerated Corrosion Program,” April 1999.

“Standards for Steam Surface Condensers,” 6th Edition, Heat Exchanger Institute, 1970.

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