10. STEAM AND POWER CONVERSION SYSTEM

Chapter 10 of this safety evaluation (SE) describes the review by the staff of the U.S. Nuclear Regulatory Commission (NRC), hereinafter referred to as the staff, of Chapter 10, "Steam and Power Conversion System," of Mitsubishi Heavy Industries, Ltd., hereinafter referred to as the applicant, design control document (DCD) for the design certification (DC) of the United States - Advanced Pressurized Water Reactor (US-APWR). The steam and power conversion system removes heat from the reactor coolant system via four steam generators (SGs) and converts it to electrical power in the turbine generator. The main condenser condenses turbine exhaust steam, removes air and other non-condensables from the condensate and transfers waste heat to the circulating water system (CWS). The main condenser hotwell collects the condensate, which is removed by the condensate system. In the condensate system, the deaerator additionally deaerates the condensate, and supplies deaerated water to the regenerative feedwater cycle. The regenerative turbine cycle heats the feedwater, and the main feedwater system returns it to the SGs.

10.1 Summary Description

DCD Tier 2, Section 10.1, "Summary Description," provides a general description of the steam and power conversion system and provides summaries of the protective features incorporated in the design of the system. DCD Tier 2, Table 10.1-1, "Significant Design Features and Performance Characteristics for Major Steam and Power Conversion System Components," provides the design and performance data for the major system components. DCD Tier 2, Figure 10.1-1, "Overall System Flow Diagram," depicts the conceptual overall system flow diagram. Figure 10.1-2, "Heat Balance Diagram Rated Condition," depicts the heat balance of the system at rated power, and Figure 10.1-3, "Heat Balance Diagram VWO Condition," depicts the heat balance at stretch power.

Detailed descriptions of the main elements of the steam and power conversion system are provided in subsequent Sections 10.2 through 10.4.11 of the DCD. The staff's review is performed separately for each of these sections, and therefore the staff's evaluation of the US-APWR turbine generator (TG) is provided below in Section 10.2 of this SE.

10.2 Turbine Generator

10.2.1 Introduction

The TG is a non-safety related system that converts the energy of the steam produced in the SGs into mechanical shaft power and then into electrical energy. The flow of steam is directed from the SGs to the turbine through the main steam system, turbine stop valves, and turbine control valves. After expanding through a series of turbines that drive the main generator, exhaust steam is transported to the main condenser.

10.2.2 Summary of Application

Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1 Section 2.7.1.1. Subsection 2.7.1.1, "Turbine Generator," contains two subsections: Subsection 2.7.1.1.1, "Design Description," which describes the system purpose and functions, location and functional arrangement, key design features, seismic and ASME code classifications, system operation, alarms, logic, interlocks, Class 1E electrical power sources and divisions, interface requirements, and performance specifications, and Subsection 2.7.1.1.2, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)," for the TG system.

Tier 2: The applicant has provided a Tier 2 system description of the TG in Section 10.2 of the US-APWR DCD, summarized here, in part, as follows:

> The TG layout is shown in DCD Tier 2, Figure 10.2-1, "Turbine Generator Outline Drawing." DCD Tier 2, Section 10.2, "Turbine Generator," provides the TG design details for the US-APWR in the following sections of the DCD.
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> - Section 10.2.1, "Design Bases" of the DCD provides the safety and nonsafety power generation design bases.
>
> - Section 10.2.2, "General Description," of the DCD provides the TG system description, the component description, and the turbine control and over speed protection systems.
>
>   The steam turbine portion of the TG of the US-APWR standard design is an 1800-rpm, tandem compound, six-exhaust flow, reheat unit. It consists of one double-flow, high-pressure (HP) turbine and three double-flow, low-pressure (LP) turbines. Two external moisture separator reheaters (MSRs) with two stages of reheating are located on each side of the TG centerline. The generator is a direct-driven, three-phase, 60-Hertz (Hz), four-pole synchronous generator with water-cooled stator and hydrogen-cooled rotor. The turbine rotors, valves and control/protection systems are designed to minimize the probability of turbine missile generation to less than 1.0E-5 per year. Orientation of the TG unit is such that a high-energy missile would be directed at an approximately 90-degree angle away from safety-related structures, systems, and components (SSCs). Turbine rotor integrity is provided by the integrated combination of rotor design, fracture toughness requirements, tests, and inspections. The TG unit and associated piping, valves, and controls are located completely within the TB.
>
>   Other related system components include a digital electro-hydraulic control system with supervisory instrumentation, overspeed protection system, and hydraulic and pneumatic systems. The US-APWR TG overspeed protection system is designed to have a mechanical overspeed trip device and an independent and diverse electrical emergency overspeed trip device.
>
> - Section 10.2.3, "Turbine Rotor Integrity," describes the turbine rotor design, material selection, fracture toughness, and rotor fatigue analysis, preservice inspection, and inservice inspection.
>
> - In Section 10.2.4, "Evaluation," the applicant provided its evaluation of the TG system.
>
> - In Section 10.2.5, "Combined License Information," the applicant provides COL items for the COL applicants.

Inspection, Test, Analysis, and Acceptance Criteria (ITAAC): The ITAAC for the TG are provided in Tier 1, Table 2.7.1.1-1, "Turbine Generator Inspections, Tests, Analyses, and

Acceptance Criteria," of the DCD.

Technical Specification (TS): There are no TS requirements identified for the US-APWR TG.

10.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 10.2, "Turbine Generator," Revision 3, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant (LWR Edition)," the Standard Review Plan (SRP), and are summarized below. Review interfaces with other SRP sections can also be found in Section 10.2.I, "Areas of Review," of NUREG-0800.

1. General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," of Appendix A, "General Design Criteria," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix A) as it relates to the TG for the protection of SSCs which are important to the safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generation of turbine missiles.

2. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC regulations.

10.2.4 Technical Evaluation

The staff reviewed the information in the DCD regarding the TG system in accordance with the review procedures in SRP Section 10.2, "Turbine Generator," Revision 2. Conformance with the acceptance criteria of SRP, Section 10.2 formed the basis for the evaluation of the TG system with respect to the applicable regulations. In order for the design to be acceptable, it must comply with the requirements of GDC 4, "Environmental and Dynamic Effects Design Bases," as they relate to the protection of SSCs which are important to the safety from the effects of turbine missiles.

To satisfy GDC 4, and as discussed in Section 3.5.1.3 of this SE, the main turbine should have a low probability of rotor failure to minimize the likelihood that turbine missiles will affect SSCs, which are important to safety. As turbine speed increases above its design limit of 120 percent of rated speed, the probability of rotor failure increases to the point where rotor failure ultimately occurs at its destructive overspeed limit. Therefore, the evaluation in Section 3.5.1.3 relies on the TG control and protection systems to ensure that turbine overspeed conditions that exceed 120 percent of rated speed, are very unlikely.

This SE subsection, Subsection 10.2.4, discusses the staff's evaluation of the TG control and protection systems to determine their adequacy in this regard. This SE subsection (under 10.2.4.6 below) also addresses DCD Section 10.2.3, "Turbine Rotor Integrity," The staff also reviewed the turbine arrangement and orientation to determine whether the assumption made in the turbine missile analysis in Section 3.5.1.3, in terms of the turbine orientation being favorable or unfavorable, is valid. Further, the staff's evaluation was to confirm that steam released from a rupture of the connection joints between the LP turbine elements and condenser will not adversely affect SSCs that are important to safety.

During its review of the Tier 1 and Tier 2 sections of Revisions 0, 1, and 2 of the DCD, the staff found that these sections lack sufficient details of the TG design and testing. The staff needed additional information, primarily to address redundancy and diversity, and common-mode failure considerations of the TG system and subsystems, such as: TG control systems, hydraulic and pneumatic systems, and associated components. The staff's review also considered the vulnerability of SSCs important to safety to turbine missiles. Since information is lacking in the application, and in order to complete its review, the staff issued Requests for Additional Information (RAIs) and follow-up RAIs, to the applicant. The applicant provided additional information in response to the staff's RAIs to address these considerations and also provided DCD markups to include corresponding changes in the next revision of the DCD. The staff's evaluation of these responses and DCD markups is reflected appropriately in the following subsections of this SE.

10.2.4.1 Design Considerations

10.2.4.1.1 Turbine Arrangement and Orientation

In SRP Section 10.2, it is noted that the TG could be a potential source for high energy missiles. In this regard, as noted in Items 1.A and 1.B of SRP Section 10.2.I, "Areas of Review," turbine orientation is an important consideration in the staff's evaluation of Tier 2, Sections 3.5.1.3 and 10.2. Therefore, the staff focused its review on the location and general arrangement of the TG system and associated components with respect to the safety related SSCs.

DCD Tier 2, Section 10.2.2.1, "General Description," provides details of the US-APWR TG system. Accordingly, the TG system consists of one double-flow HP turbine and three double-flow LP turbines, two sets of external MSRs, main generator, an exciter, associated controls and auxiliary subsystems. The HP turbine and three LP turbines are connected in tandem, i.e., their shafts are connected, and they operate at their nominal/rated speed of 1800 rpm (188.5 rad/s), turning the four-pole synchronous generator to produce 60 Hz alternating current (ac) power. The TG unit and associated piping, valves and controls are located in the turbine building (TB). No safety-related SSCs have been identified as being located in the TB; thus a failure in the TG package does not directly affect any SSCs that are important to safety. Also, the layout of the TG unit and associated equipment in relation to the safety-related SSCs are shown in Tier 2, Figure 1.2-1, "Typical US-APWR site Arrangement Plan." The orientation of the TG is such that any high-energy missiles generated, will be directed at a 90-degree angle away from the safety-related SSCs. Based on the above descriptions in the DCD and a review of the Tier 2, Figure 1.2-1, the staff finds that there are no safety-related SSCs that could be adversely impacted by a turbine missile. Therefore, the staff finds the TG orientation acceptable, since it meets the SRP guidance with respect to the TG arrangement.

Additionally, Section 10.2.2.1 states that the probability of a destructive overspeed condition and missile generation is less than 1×10^{-5} per year in accordance with SRP Section 3.5.1.3, assuming proper inspections and tests are performed as recommended by the TG supplier. The staff reviewed Tier 2, Section 3.5.1.3, "Turbine Missiles," where the applicant described the TG orientation and stated that the potential for low trajectory turbine missiles to impact safety-related SSCs within the same unit is minimized since these SSCs are located outside the high-velocity, low-trajectory missile strike zone. The staff's evaluation of turbine missile probability is discussed in Section 3.5.1.3 of this SE.

Further, as indicated earlier in Section 10.2.2 of this SE, the TG system is non-safety related, and is located in the TB, which contains no safety-related SSCs. Therefore, in the event of a failure of a high-, or moderate-energy line break, or ruptures of the LP turbine exhaust hood connection joints, safety-related SSCs will not be affected.

10.2.4.1.2 Turbine Speed Control and Overspeed Protection

The US-APWR main turbine control and overspeed trip functions are described in Section 10.2.2.3, "Control Function," of the DCD. Accordingly, the US-APWR turbine control function consists of a turbine control system (TCS), turbine protection system (TPS), and the turbine supervisory instrument system. These systems have individual cabinets located in the non-Class 1E instrumentation and control (I&C) room in the auxiliary building. The main turbine is protected from overspeed conditions by the TCS, TPS, and the mechanical overspeed trip function.

The speed control mode of the TCS protects the turbine from overspeed by maintaining the desired speed. The overspeed protection mode of the TCS operates if normal speed control should fail or upon a loss of load.

The TPS includes an emergency back-up electrical overspeed trip (EOST) function that is independent from the TCS and provides the capability to trip the turbine in the event that the rotating speed exceeds the overspeed protection trip set-point.

In addition to the EOST, the US-APWR main turbine is equipped with a primary mechanical overspeed trip (MOST) feature that is independent and diverse from the EOST. The MOST consists of only mechanical and hydraulic devices.

Based on the guidance provided in SRP Section 10.2, the staff considered the following in its evaluation of the TG system: (1) the capability of the TG control and overspeed protection systems to detect the turbine overspeed conditions and actuate appropriate valves to preclude such condition, and (2) implement redundancy/independency/diversity, testability, and reliability in the overspeed protection I&C features. To verify conformance of the US-APWR main turbine control and overspeed protection design to the GDC 4 criteria, the staff reviewed the DCD and evaluated the functional requirements of the TG components, control features, and associated air/hydraulic systems using the SRP Section 10.2 guidance. The staff evaluation of these design features is provided below:

- **Turbine Control System**

  As described in, DCD Section, 10.2.2.3.1, "Turbine Control System," during normal plant operation, the speed control function of the TCS provides speed control, acceleration, and overspeed protection. The TCS has two modes of operation to protect the turbine against overspeed. The first mode is the normal speed control, which maintains the turbine desired speed; whereas the second is an overspeed protection control mode that operates if the normal speed control fails or upon a loss of load. Additionally, the TCS performs a power/load unbalance (PLU) function. For US-APWR, the PLU function is initiated when the difference between the turbine power and generator load exceeds 30 percent.

Based on the review of DCD, Section 10.2.2.3.1, the staff determined that redundancy is built into the TCS for speed and overspeed controls to protect the turbine during normal plant operation. The TCS employs three electric speed sensors that are independent from the TPS. The TCS consists of two redundant microprocessors and redundant power supplies. One processor is in control mode while the other is in standby mode. If the microprocessor in the control mode fails, the standby microprocessor takes over automatically. The turbine is tripped automatically in the event that both of the microprocessors fail to perform their function or both of the redundant power supplies fail. Redundancy is built into the OPC in the TCS. Thus, failure of a single component in the TCS will not disable the turbine speed control and overspeed protection capabilities. Loss of hydraulic pressure in the TCS causes the turbine to trip. Therefore, damage to the overspeed protection components results in a turbine trip. The staff further identified that the speed controller cuts off steam to the turbine at approximately 103 percent of rated speed by closing turbine control and intercept valves, which conforms to Item 2.A.i of SRP, Section 10.2, Subsection III, “Review Procedures.” Based on the above findings, the staff finds that the TCS design of the US-APWR turbine for the normal speed/overspeed control is acceptable.

- **Turbine Protection and MOST Systems**

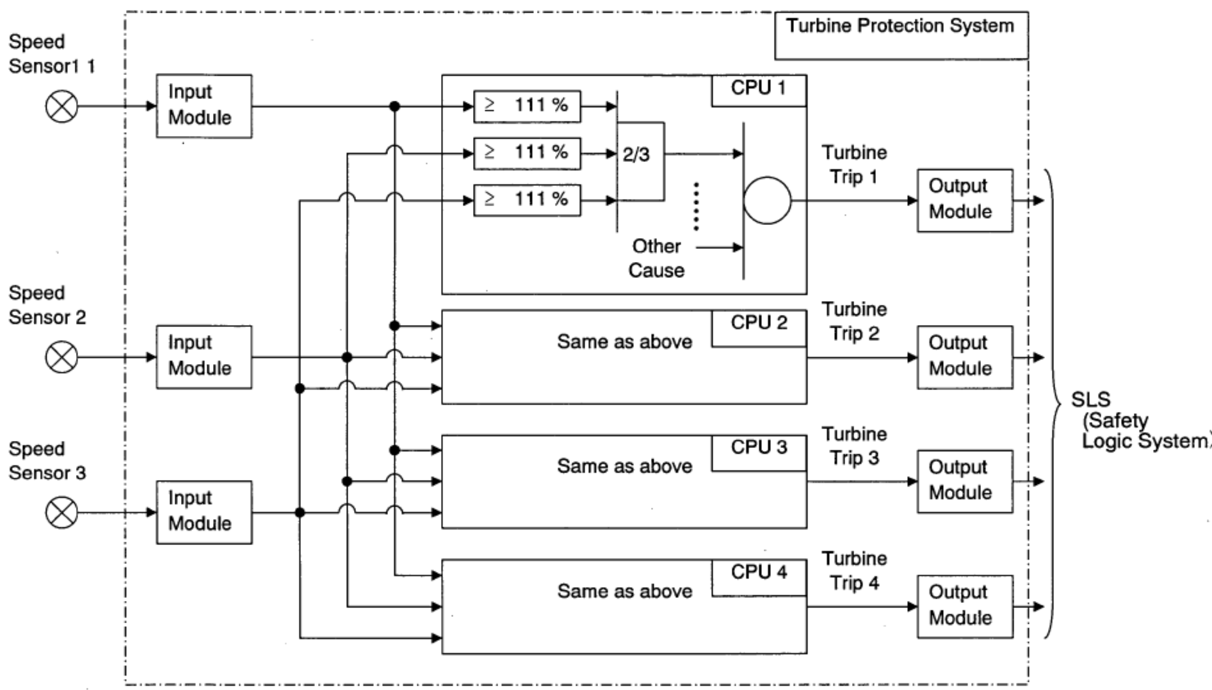
Upon review of the information provided in Section 10.2.2.3 of the DCD and its Subsection 10.2.2.3.2, “Turbine Protection System,” the staff found that additional information was needed to establish the redundancy and diversity, and single failure considerations of the TG overspeed protection system and the MOST. Therefore, in **RAI 237-2141, Questions 10.2-2, 10.2-3, and 10.2-4 and RAI 598-4754, Questions 10.2-5, and 10.2-6**, the staff requested additional information for the overspeed protection system, and for the associated pneumatic and hydraulic fluid subsystems. Also, the staff requested schematics and logic diagrams for these systems to show how they function.

The applicant provided its responses to the above referenced RAI questions in letters dated March 25, 2009, and June 7, 2011, which included a DCD markup reflecting these responses. Additionally, as part of the response to Question 10.02-5, the applicant included Figure 1, “Overspeed Trip Device” and Figure 10.2-3, “Simplified schematic of turbine control and protection system.” Also, Figure 2-1, “Schematic and Logic Diagram for Emergency Back-up Electrical Overspeed System,” was included in response to Question 10.2-2. These figures were included to depict the redundant and diverse features and functional performance of the overspeed control systems. Based on its review of the responses and DCD markup, the staff identified the following information:

The purpose of the TPS is to detect undesirable operating conditions of the turbine, take appropriate trip actions, and provide information to the operator about the detected conditions. The TPS is independent of the TCS, including the speed sensors, as shown in Figure 10.2-1.

The emergency back-up electrical overspeed turbine trip at approximately 111 percent of the rated turbine speed is performed by the TPS and is designated as the EOST. The TPS closes the main stop, control, intercept, reheat stop, and extraction non-return valves resulting in turbine trip.

The TPS trips the turbine by opening four turbine trip solenoid valves to drain the emergency trip header hydraulic fluid. The TPS is independent from the TCS, including the speed sensors, as show in Figure 10.2-1. The TPS cabinet consists of quadruple-redundant microprocessors and redundant power supplies. Each microprocessor corresponds to each trip solenoid valve. Each TPS output module provides an independent turbine trip command (normally closed contact) to its corresponding trip solenoid valve via the Power Interface module of the Safety Logic System (SLS). This hardwired interface between the TPS and the safety-related SLS is evaluated in Section 7.3 of this SE. Failure of a TPS microprocessor or its redundant power supply initiates a turbine trip command (open contact), which results in de-energizing of the corresponding turbine trip solenoid valve, thus allowing the valve to open under internal spring force and dump steam valve hydraulic pressure. The TPS is designed as a fail-safe system. The emergency trip header pressure is established when the turbine trip solenoid valves are energized and closed. The valves are arranged in two channels as shown in Figure 10.2-2. At least one solenoid valve in both channels must open (de-energize) to trip the turbine. This arrangement allows testing of each channel without tripping the turbine, and prevents turbine trip due to failure of a single component.



<u>Figure 10.2-1; Electrical Overspeed Trip (EOST)</u>

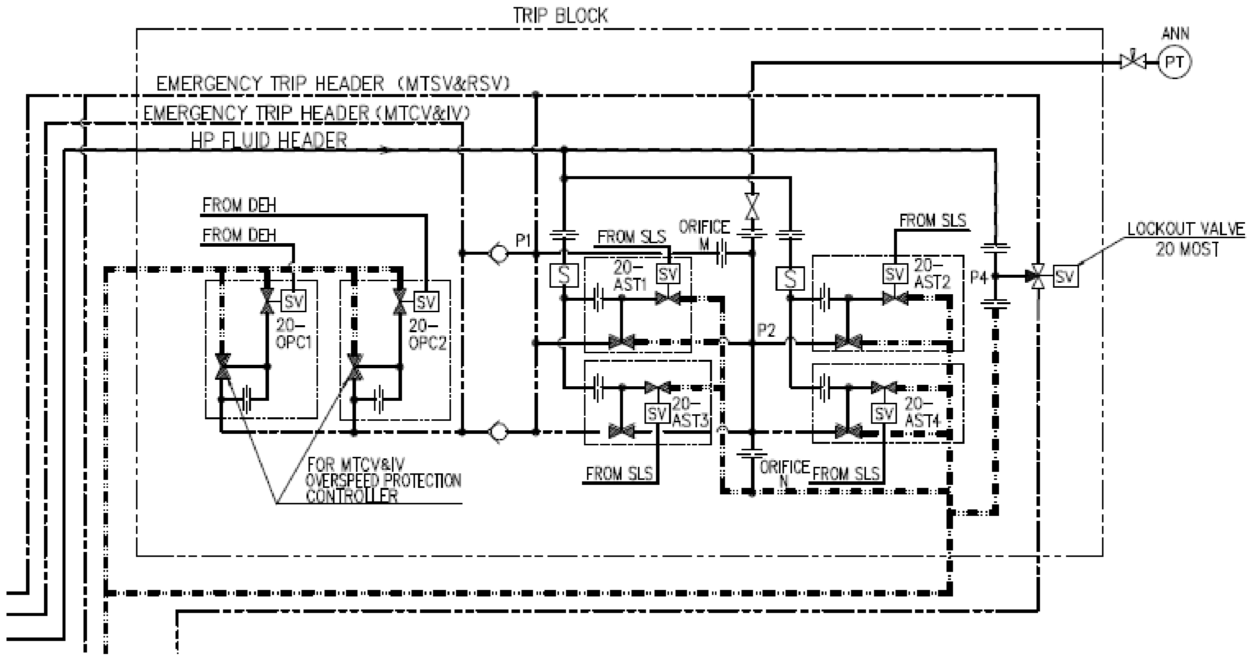


Figure 10.2-2; Turbine Trip Block

In the US-APWR, the primary turbine overspeed trip function is performed by the MOST, which consists of only mechanical and hydraulic devices and is independent of the TPS, which includes EOST. The MOST device releases the fluid pressure of the emergency trip header of the main turbine stop valve (MTSV) and reheat stop valve (RSV) (i.e., MTSV and RSV trip header) and consequently the fluid pressure of the emergency trip header of the main turbine control valve (MTCV) and intercept valve (IV) (i.e., MTCV and IV trip header) through redundant check valves in the event that the turbine speed reaches 110 percent of rated speed. Similar to the trip action initiated by the EOST, the resultant drop in fluid pressure to the hydraulic actuators of all the turbine steam supply valves initiated by the MOST allows the steam valves to shut quickly under spring force and cut off steam to the turbine.

The turbine trip manual switch is located on the operator console in the main control room (MCR). The signal of the turbine trip switch is sent to the SLS through the TPS de-energizing the turbine trip solenoid valves resulting in turbine trip. A back-up turbine trip manual switch, which is part of the diverse actuation system (DAS), is also provided in the MCR. This manual turbine trip feature is independent from TPS and SLS, and can perform its function in the presence of a software common cause failure (CCF).

The mechanical overspeed and manual trip header can be tripped manually via a trip lever handle mounted on the turbine governor pedestal.

Based on its review of the responses, the staff determined that the US-APWR TG emergency overspeed protection systems conform to the GDC 4 requirements, because they are redundant in that the emergency electrical trip system is separate functionally and physically. The MOST is designed to actuate before the EOST to initiate a turbine trip. Also they are diverse, since one is mechanical and the other is an electrical device, which meets the guidance in Item 2.A of the SRP Section 10.2.III as it relates to diversity. Further, the overspeed trip set-points for MOST and EOST are 110 percent and 111 percent of turbine rated speed, respectively, which are within the bounds of the SRP guidance. The guidance in Items 2C and 2D of SRP Section 10.2.III describes that the mechanical overspeed trip device actuate all the steam admission valves to close at approximately 111 percent of turbine rated speed, and the redundant backup electrical trip device actuate these valves at 112 percent of rated speed. Further, operators can manually trip the turbine remotely from the control room and locally at the turbine. There is a turbine trip manual switch, which is part of the DAS and independent of the TPS. Also, operators can manually trip the turbine locally by means of the trip lever on the governor. The evaluation of the DAS is provided in Chapter 7 of this SE. Therefore, the staff determined that the US-APWR TG turbine emergency trip protection systems meet the requirements of GDC 4 and SRP guidance as discussed above, and therefore finds the overspeed protection systems acceptable, as they meet the applicable NRC regulations.

Additionally, with respect to the two electrical overspeed control systems, i.e., TCS and EOST, the guidance in Item 2.D of SRP Section 10.2.III states that control signals from the two electrical systems are isolated from, and independent, of each other. Based on the applicant's responses and DCD markups, the staff identified the following:

- The TCS has a function of speed control (including overspeed control) and load control, whereas, the EOST has a function to trip the turbine under abnormal and emergency conditions, such as load rejection.

- Both systems have dedicated triple-redundant speed sensors and are independent of each other, with separate processors and input/output modules.

- For each system, control signals are processed in quadruple-redundant microprocessors, and these trip controllers are separate from each other.

- Cabinets for the EOST are independent from those of the TCS. These systems have individual cabinets located in the non-class 1E I&C room in the auxiliary building.

Power sources, as both TCS and EOST systems are installed in separate cabinets, has its own redundant power supplies. Further, loss of electric power will result in all the turbine steam supply valves closing, i.e., a fail-safe condition. Based on the above considerations, the staff determined that the control signals from the normal TCS control system and the EOST trip system are isolated from, and independent of, each other, and therefore are consistent with the review guidance specified in Item 2.D. of SRP 10.2.III. Accordingly, **RAI 237-2141, Questions 10.2-2, through 10.2-4 and RAI 598-4754, Questions 10.2-5 and 10.2-6**, are resolved and closed. The proper incorporation in the next revision of the DCD will be confirmed by the staff. It will be tracked as **Confirmatory Item 10.2-1**.

10.2.4.1.3. Turbine Overspeed Control & Air/Hydraulic Systems – Failure Modes

- **Single-Failure Considerations**

  Item 2A of SRP Section 10.2.III, as it relates to the adequacy of the control and overspeed protection system, specifies that a single failure of any component or subsystem will preclude an unsafe turbine overspeed. In this regard, the staff issued

RAI 237-2141, Questions 10.2-3 and RAI 598-4754, Question 10.2-6(14) to address any single failure vulnerabilities that exist in the TCS and the two emergency trip systems.

In its responses to these RAI questions, dated March 25, 2009 and June 7, 2011, the applicant described that the emergency overspeed trips (MOST and EOST) consist of a mechanical and an electrical trip device, which are redundant and diverse to each other. The two electrical overspeed control systems (i.e., TCS and EOST) have separate speed sensors, redundant processors, and their control signals are isolated from, and independent of, each other. The four solenoid-operated turbine trip valves are arranged in two parallel flow paths, such that the failure of any one to open on demand will not prevent a turbine trip. Further in its response to **RAI 237-2141, Question 10.2-3**, the applicant stated that the TPS and the EOST have redundant power supply sources, which are backed by station batteries and alternate alternating current power source. Furthermore, the spring-assisted non-return valves in those extraction steam lines that have sufficient steam to cause turbine overspeed are provided with redundant valves. Therefore failure of a single extraction non-return valve to close will not cause the turbine speed to exceed the design rated speed. Therefore, based on the above discussions and the applicant responses to staff's RAIs, the staff determined that there would be no single failure vulnerability in the design of the US-APWR turbine control and overspeed trip systems. Thus, the staff determined that the design is acceptable since it meets the above referenced SRP guidance as it relates to the single failure criteria.

- **Common-Mode and Common-Cause Failures**

  Common mode and CCF vulnerabilities could prevent the turbine overspeed trip systems from functioning properly. Therefore, to address these issues identified in NUREG-1275, Volume 11, "Operating Experience Feedback Report - Turbine-Generator Overspeed Protection Systems," in **RAI 237-2141, Question 10.2-3 and RAI 598-4754. Question 10.2-5**, the staff requested additional information pertinent to the design and operational considerations of the above systems. Also, the staff requested the applicant to address the fluid flow paths, shared components, failure modes and CCF vulnerabilities. Specifically, the staff requested that the applicant provide a schematic to show how the turbine overspeed protection and associated fluid control systems function, and how the fluid flow-paths are shared. Based on the applicant's responses, dated March 25, 2009, and June 7, 2011, and the DCD markup, the staff identified the following information:

  > As evaluated earlier in this SE, the turbine overspeed protection systems (i.e., MOST and EOST), are designed with built-in redundancy and diversity features. These design features allow on-line testing of their trip functions while the unit is in operation at rated speed and detect if there are any defects or functional deficiencies to trip the turbine under abnormal conditions. Also, the EOST has a self-diagnostic function to ensure reliable operation of the system. Furthermore, as the applicant described in 10.2.2.3.2.1, "Trip Block," and depicted in Table 10.2-5, "Inspection and Test Requirement for Overspeed Trip Device ," of the DCD markup, the MOST, EOST, and associated components are to be tested once a month during TG operation.
  >
  > In the hydraulic portion of the turbine control and overspeed systems, multiple headers and drain lines and flow paths are used to drain the hydraulic fluid.

Periodic testing and inservice inspections of both air and hydraulic systems will be performed, which can identify problems and eliminate CCF in those systems. Hydraulic fluid systems will be sampled and tested every three months to confirm that all the control parameters of the fluid are within the limits of manufacturer's specifications.

Further, to avoid plugging in the hydraulic lines, the applicant considered measures such as: using 100-percent triaryl phosphate in the hydraulic fluid so that sludge deposition is unlikely to block the pipes, air-cooled oil heat exchangers to be used to minimize water intrusion into the control fluid system, and installing appropriate filters in the hydraulic fluid supply and return lines. Furthermore, the applicant will be conducting periodic sampling and testing of these fluid systems.

Additionally, all turbine steam supply valves are designed to be kept open when control oil pressure or emergency trip header pressure is held at design pressure. Also, the non-return check valves in the extraction line to the No. 5 feedwater heater are designed to close by loss of air supply system. All solenoid-operated, air or hydraulic control valves for steam valves are designed to fail open if deenergized upon loss of electric power to them to effect shutting of the steam valves who's hydraulic or air actuators they control. Therefore, all the turbine valves are closed due to loss of pressure in the air and/or hydraulic fluid lines if they are broken or due to loss of electric power.

Based on the above discussions, the staff determined that the applicant adequately addressed the countermeasures against the common-mode and CCF vulnerabilities, as they pertain to the design and operation of the turbine overspeed control systems. The hydraulic fluid systems are designed with multiple headers, adequate flow paths and pipe sizes to drain the fluid from the overspeed trip valves and steam admission valves to the central hydraulic fluid reservoir. These provisions ensure proper functioning of the MOST and EOST of the overspeed protection systems. Also, periodic testing and monitoring of the pressure in the common drain lines will reduce the probability of blockage and plugging of drain lines with corrosion products. Thus, the staff determined that the applicant provided adequate design provisions to the air/hydraulic systems and its flow paths to support the turbine overspeed protection functions. Therefore, the staff finds the applicant adequately addressed the issues identified in NUREG-1275, and further finds the design acceptable as it relates to the common-mode and CCFs of the main turbine overspeed protection systems and its associated systems.

10.2.4.1.4 Turbine Steam Admission and Extraction Steam Non-Return Isolation Valves

The primary mechanical and the emergency backup EOST systems actuate to close the turbine stop, control, intermediate stop, intercept, and spring-assisted extraction steam non-return isolation valves to prevent the turbine from exceeding its design overspeed limit of 120 percent of rated speed. The turbine stop and intermediate stop valves are redundant from their respective control and intercept valves. The valve arrangements are typical of designs previously approved by the staff and are shown in various figures in DCD Tier 2, Sections 10.3 and 10.4.7. Regarding the extraction non-return valves, in response to staff's **RAI 598-4754 Question 10.2-6 (1)**, the applicant added a new Section 10.2.2.8, "Extraction Non-Return Valves," to the DCD, which describes the details, such as the location, number, and type of these non-return valves. Also, the applicant added a new Figure 10.2-2, "Arrangement of

Extraction Non-Return Valves," to the DCD, which depicts the arrangement of these valves in the TG and MSR extraction lines to the feedwater heaters.

Further, Table 10.2-4, "Turbine-Generator Valve Closure Times," provides valve closing time for the turbine stop and control, reheater stop and intercept valves, and extraction non-return valves. Also, in its DCD markup of Section 10.2.2.3.2, the applicant stated that the valve closing times of the above steam admission valves are equal to or less than those values listed in Table 10.2-4. This is to ensure that at the over-speed protection control OPC (i.e., normal overspeed) activation, the turbine speed does not reach the primary overspeed set-point of 110 percent of rated speed, and also does not hit the design overspeed (i.e. 120 percent of rated speed) at turbine trip. Furthermore, according to the applicant, these closing times will be confirmed to be equal to or less than the specified time during pre-operational test prior to fuel loading and start-up test in the field. The applicant has established allowable closure time limits for the turbine stop and control valves to satisfy reactor performance and transient analysis considerations. As discussed in DCD Tier 2, Section 10.2.2.2.8, the spring-assisted extraction steam non-return isolation valves are designed to be fail-safe and do not have any CCF possibility

Based on the above discussions, the staff determined that the applicant has adequately addressed the considerations referred to in SRP Section III.3, and the design ensures that no single valve failure can disable or otherwise compromise the overspeed control function of the TG system. Therefore, the staff finds the design to be acceptable in this regard.

10.2.4.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

DCD, Tier 1, Section 2.7.1.1, "Turbine Generator (T/G)," provides DC information for its TG system and associated ITAAC. In accordance with 10 CFR 52.47(b)(1) requirements as identified earlier in the regulatory criteria of this SE, the applicant identified these ITAAC in Tier 1, Table 2.7.1.1-1, "Turbine Generator Inspections, Tests, Analysis, and Acceptance Criteria." Also, Section 14.3.7 of this SE evaluates DCD Tier 1 information for balance-of-plant SSCs, and the evaluation of DCD Tier 1 information in this section is an extension of the evaluation provided in SE Section 14.3.7. This evaluation pertains to plant systems aspects of the DCD Tier 1 information for the main turbine.

The staff reviewed the Tier 1 information provided in Revision 0 through Revision 2 of the DCD to confirm that it included appropriate Tier 1 requirements for the TG system. Based on its review of these sections, the staff determined that additional information was needed, as it relates to testing of the as-built TG, specifically the turbine control and emergency overspeed trip systems and associated controls. Therefore, the staff issued **RAI 237-2141, Question 10.2-4**, and followup **RAI 599-4756, Revision 2, Questions 14.3.7-51 and 14.3.7-52**, requesting the applicant to provide additional information with respect to the turbine missile probability, arrangement of the main turbine and associated valves, testing of the mechanical and electrical emergency overspeed trip devices, testing of manual and remote turbine trip functions, controls and alarms in the MCR, and a turbine trip in response to a reactor trip.

In letters dated March 25, 2009, and July 2010, the applicant provided its responses to the staff's RAI questions. Further, the applicant provided a markup of DCD, Tier 1, Section 2.7.1.1, with revisions to the location and functional arrangement, key design features, and alarms, displays, and controls, reflecting these responses. Also, the applicant stated that a further description of the turbine trip upon reactor trip is described in DCD Tier 2, Subsection 7.3.1.11, and depicted in Figure 7.3-4. More importantly, a markup of the proposed revisions and additions to ITAAC Table 2.7.1.1-1 was also provided reflecting these responses as related to new tests and inspections of the as-built turbine. The staff reviewed and evaluated the applicant's responses in light of the NRC requirements and relevant guidance. However the response to Questions 14.03.07-51 and 14.03.07-52 are not acceptable, since the key design features and the turbine missile probability acceptance criteria have been deleted from the ITAAC in Table 2.7.1.1-1 of the DCD, Tier 1. Therefore, in **RAI 5910, Question 14.3.7**, the staff requested that the applicant revise the ITAAC in DCD, Tier 1, to include the acceptance criteria for both the TG arrangement and for the turbine missile. **RAI 5910, Question 14.3.7** is being tracked as **Open Item 10.2-1**

Based on its review of the RAI responses; additional information in the above DCD markups, and proposed new tests and inspections, the staff concludes that the proposed ITAAC for the main turbine of the US-APWR DC provides reasonable assurance that the turbine will trip on receiving overspeed trip signals, and all the associated valves will close as designed and therefore meet the requirements of 10 CFR 52.47(b)(1). Therefore, the staff finds the ITAAC for the TG control system acceptable, as it relates to the design commitment, testing and inspection of the turbine overspeed trip devices and the steam admission valves, and associated acceptance criteria in the proposed ITAAC, pending a resolution to the open item identified above.

10.2.4.3 Inspection and Testing

The staff reviewed the DCD Tier 2, Section 10.2.2.3.5, "Inspection and Testing Requirements," where the DCD states that the major system components are readily accessible for inspection and testing during plant operation. Turbine trip circuitry is tested prior to startup, and control valves are tested with the load reduced to that capable of being carried with one control valve closed for testing.

However, in **RAI 598-4754, Questions 10.2-6(1), 10.2-6(5), and 10.2-6(18)**, the staff requested additional information regarding the tests (and test frequencies) that will be performed on the turbine to ensure that the turbine control and overspeed protection systems, including the remote manual trip are maintained and function properly.

In its response dated June 7, 2011, the applicant proposed to add DCD Tier 2, Table 10.2-5, "Inspection and Testing Requirements for Overspeed Trip Device," and provided a markup of the table and revisions to pertinent Tier 2 subsections. The test items in the table included: trip block, mechanical and overspeed trips, turbine valves, turbine control fluid sampling and testing, and valve inspection. Also included are confirmation criteria of the test, and frequency of the tests and inspections. Furthermore, the valve closure times are identified in DCD Tier 2, Table 10.2-4, "Turbine-Generator Valve Closure Times."

Based on its review of DCD Tier 2, Section 10.2.2.3.5, and the proposed addition of Tier 2 Table 10.2-5, the staff determined that there is reasonable assurance that the tests identified in the DCD are adequate to meet the industry practice and the guidance provided in Item 1.C of the SRP acceptance criteria in Section 10.2.II, as related to periodic testing. However, a further evaluation of these tests (and test frequencies) of the turbine valves and their effects on turbine missile generation is provided in the staff's evaluation of DCD Tier 2, Section 10.2.3.5, "Inservice Inspection," in this SE.

10.2.4.4 Initial Test Program

Section 14.2 of this SE contains the evaluation of the initial plant test program for the US-APWR DC. The staff evaluation in this SE section is an extension of the evaluation provided in Section 14.2. The following initial test program specifications in the DCD pertain to the TG design considerations evaluated in Section 10.2.2.1 of this SE:

- 14.2.12.1.26 Extraction Steam Preoperational Test
- 14.2.12.1.27 Main Turbine System Valves Preoperational Test

The above tests in the DCD describe the initial test program to verify the functional performance of the extraction non-return valves and the TG steam inlet valves, including actuation (closing) from turbine trip signals. After its review of the above sections in Revision 2 of the DCD, the staff found the information to be insufficient to confirm that it adequately addressed the TG design and performance considerations. Therefore, in **RAI 600-4755, Question 14.2-122**, the staff requested the applicant to provide additional information in this regard. In Revision 2 of its response to the **RAI 600-4755**, the applicant provided the information that the staff requested. The applicant also provided a mark-up of the DCD reflecting the responses.

The staff reviewed the specifications provided in the revised test programs referred to above to confirm that they adequately addressed the TG design and performance considerations. Upon its review, the staff confirmed that the test program verifies proper performance and integrated operation of the main turbine and associated controls and closure of steam admission and extraction non-return valves on turbine trip signals. Therefore, the staff considers the initial test program to be acceptable with respect to the above referenced sections.

10.2.4.5 Inservice Inspection, Testing, and Maintenance of Valves Essential for the Turbine Overspeed Protection

Since turbine overspeed protection relies upon the capability of the main steam stop and control valves, MSR intercept stop and intercept valves, and extraction non-return valves, the staff verified that Tier 2, Section 10.2.3.5, “Inservice Inspection,” included these valves as part of this program. The staff’s evaluation of this area is provided under the evaluation of DCD Section 10.2.3, “Turbine Rotor Integrity,” in SE Subsection 10.2.4.6 below.

10.2.4.6 Turbine Rotor Integrity (Evaluation of DCD Tier 2, Section 10.2.3, “Turbine Rotor Integrity”)

Turbine rotors have large masses and rotate at high speeds during normal operation, and therefore, failure of a turbine rotor may result in the generation of high-energy missiles that may affect safety related equipment and components. Therefore, the staff has reviewed the turbine rotor using the guidelines in SRP Section 10.2.3, “Turbine Rotor Integrity,” to ensure that the turbine rotor materials have acceptable fracture toughness and mechanical properties to maintain the integrity of the turbine rotor and that the turbine rotor has a low probability of failure.

<u>Material Specifications:</u>

DCD, Revision 1, Tier 2, Section 10.2.3.1, “Materials Selection,” specifies that the turbine rotors are made from ladle-refined, vacuum-deoxidized, Ni-Cr-Mo-V alloy steel that meets the chemical properties of American Society for Testing and Materials Standard A470 (ASTM A470), Classes 5, 6, and 7. This alloy steel has the lowest practical concentrations of residual elements due to the oxidizing electric furnace melting process, which de-phosphorizes the alloy steel. Oxygen, sulfur and hydrogen are removed by ladle furnace refining, and then further degassed when the alloy steel is poured into a mold under a vacuum, which minimizes chemical segregation.

DCD, Revision 1, Tier 2, Section 10.2.3.1, specifies that the turbine rotor material complies with the chemical properties of ASTM A470, (Grade C) Classes 5, 6, and 7, but the specification for the rotor has lower limitations for phosphorous, sulfur, aluminum, antimony, tin, argon and copper than ASTM A470. In addition, DCD, Revision 1, Tier 2, Section 10.2.3.2, "Fracture Toughness," states that the impact energy and transition temperature requirements are more rigorous than those given in ASTM A470 Class 6 or 7. ASTM A470 Grade C includes Class 5, 6, and 7 materials. The above does not provide sufficient information concerning the material used for the LP and HP turbine rotors in accordance with SRP Section 10.2.3 to assess the acceptability of the material with regard to turbine rotor integrity.

In its response to **RAI 199-2073, Question 10.2.3-1**, dated March 10, 2010, the applicant stated that the turbine rotor material complies with ASTM A470, except that for the low-pressure turbine rotor, the chemical composition ranges for trace elements, such as phosphorous, sulfur, aluminum, and antimony, are more restrictive than ASTM A470. The applicant stated that these additional requirements regarding chemical composition are provided in its purchasing specifications. In addition, the applicant stated that the chemical composition ranges for tin, argon, and copper are not specified in ASTM A470. However, the applicant stated that it specifies chemical composition ranges for tin, argon, and copper in procurement specifications to limit the amount of these trace elements. The staff finds the applicant's response acceptable and subsequently confirmed that DCD, Revision 2, Tier 2, Section 10.2.3.1, clarified the chemical composition range as stated in the applicant's RAI response. The staff finds that the additional restrictions regarding tramp elements for the ASTM A470 materials provides a cleaner material and a more homogeneous material that will provide more resistance to degradation mechanisms, such as stress corrosion cracking.

However, in DCD Revision 2, the applicant revised Section 10.2.3.1 to delete the reference to Grade C (Classes 5, 6, and 7). Therefore the DCD no longer specifies the type of material (grade or classification) from ASTM A470. Since there are different grades and classifications in ASTM A470 that have different chemical compositions and mechanical properties, the staff cannot assess the acceptability of the material, nor determine whether the material is bounded by the turbine missile analysis. Therefore, in **RAI 574-4633, Question 10.2.3-8**, the staff requested that the applicant revise the DCD to include grade and classification of ASTM A470 material or include a reference to the specific material ordering requirements in the DCD that are bounded by the turbine missile analysis. **RAI 574-4633, Question 10.2.3-8**, is being tracked as **Open Item 10.2.3-1**.

Regarding mechanical properties, the applicant's response to **RAI 199-2073, Question 10.2.3-1**, dated March 10, 2009, stated that the turbine rotor material purchasing specifications call for mechanical properties that meet or exceed those specified in ASTM A470, Class 6. For example, yield strength and toughness are slightly higher than those specified in ASTM A470, Class 6. However, DCD, Revision 1, Tier 2, Section 10.2.3.2, allows the use of either Class 5, 6, or 7 materials from ASTM A470. The staff notes that the mechanical properties of Class 5 and 7 materials differ from Class 6 material. These purchasing specification mechanical properties that exceed the requirements of ASTM A470, Class 6, were used in the bounding turbine missile analysis (Mitsubishi Technical Report MUAP-07028, "Probability of Missile Generation from Low Pressure Turbines," dated December 31, 2007). Therefore, Class 5 and 7 materials are not bounded by the turbine missile analysis. In addition, in DCD Revision 2, Tier 2, Section 10.2.3.1, was modified by deleting all classifications of ASTM A470 (Class 5, 6, and 7) material. Since all classification information for the material was removed from DCD Section 10.2.3.1, the staff cannot assess whether the allowed ASTM A470 materials are bounded by the turbine missile analysis. Therefore, in **RAI 574-4633, Question 10.2.3-8**, the staff also requested that the applicant provide the material specification, including the type of material (grade and classification) or a reference to the specific material ordering requirements that are bounded by the turbine missile analysis. This is part of **Open Item 10.2.3-1**.

DCD, Revision 1, Tier 2, Section 10.2.3.1, specifies that Charpy V-notch tests and tensile tests in accordance with ASTM A370 and/or the equivalent are required from the forging supplier. This does not provide specifics about the tests in order for the staff to review it against the guidelines of SRP Section 10.2.3, which has specific guidelines for the acceptability of the material selection and fracture toughness of the rotor.

In its response to **RAI 199-2073, Question 10.2.3-2**, dated March 10, 2009, the applicant stated that a heat analysis and product analysis will be performed to verify the chemical composition of the rotor material. In addition, tensile tests and a Charpy impact test in accordance with ASTM A370 will be performed to verify the mechanical properties of the rotor material. In addition, DCD Section 10.2.3.1 will be revised to delete "equivalent standards" for ASTM A370 as provided in the applicant's proposed DCD markup. The staff finds that the mechanical testing is acceptable since it will be accomplished in accordance with ASTM A370, which meets the guidelines of SRP Section 10.2.3. The staff confirmed subsequently that in Revision 2 of the DCD, Section 10.2.3.1 was revised as indicated in the DCD markup. However, the staff notes that the acceptance criteria provided in the response to **RAI 199-2073, Question 10.2.3-2**, for the 50 percent fracture appearance transition temperature (FATT) and Charpy V-notch energy do not meet the acceptance criteria of -18 °C (0 °F) and 8.3 kg-m (60 ft lbs), respectively, which are provided in SRP Sections 10.2.3 (Paragraphs II.1b and II.1c). Therefore, the applicant is requested to provide a discussion on why the material properties for the 50 percent FATT and Charpy V-notch energy provided in the response **to RAI 199-2073, Question 10.2.3-2**, is sufficient to ensure that the turbine rotor has adequate fracture toughness during startup and normal operating temperatures. **RAI 574-4633, Question 10.2.3-9**, which is associated with the above request, is being tracked as **Open item 10.2.3-2**.

Regarding the locations of the test specimens and the material properties of the internal regions for a non-bored rotor, the applicant stated in its response to **RAI 199-2073, Question 10.2.3-2**, that the tensile and Charpy testing will be performed on five specimens from the outer periphery of the turbine rotor. For a bored rotor, additional tensile and Charpy testing will be performed from three specimens on the interior bore periphery of the turbine rotor. The staff finds the number of specimens acceptable since it meets the guidance provided in SRP 10.2.3. However, the staff identifies **Open Item 10.2.3-3**, because Revision 2 of the DCD does not include the number of specimens to be tested as provided in the response to RAI 199-2073, Question 10.2.3-2. RAI 574-4633, Question 10.2.3-10, which is associated with the above request, is being tracked as **Open Item 10.2.3-3**.

Based on the above, the turbine rotor can be either a bored or non-bored rotor design. The applicant also stated that the homogeneity and quality of the material at the center core of a non-bored rotor is ensured through the steel making process. In addition, in its response to RAI 199-2073, Question 10.2.3-2, the applicant provided some material test result comparisons between the rotor outer periphery and the rotor center core so that the mechanical properties at the rotor center core can be evaluated using the material at the outer periphery of the turbine rotor. Therefore the applicant will not perform chemical composition and mechanical testing of the core for non-bored rotors. The NRC staff notes that the comparative material test results provided show that the material at the center core of the turbine rotor has material properties (lower reduction of area, lower impact energy and higher 50 percent FATT temperature) that are less conservative than at the outer periphery, which is due to the different solidification rates of this large component. Therefore, the material properties cannot be accurately and consistently determined using only test specimens from the outer periphery of the turbine rotor. Since the material properties do vary from the outer periphery to the internal center core, and the internal core of each non-bored as-built turbine rotor cannot be verified, the applicant should delete the non-bored turbine rotor from its design, or provide specific destructive testing and non-destructive testing taking into consideration that the internal material properties and the presence of internal defects of the as-built rotor cannot be confirmed. In addition, appropriate operating experience should be provided which justifies that the integrity of the rotor can be maintained. RAI 574-4633, Question 10.2.3-11, which is associated with the above request, is being tracked as Open item 10.2.3-4

For the reasons described above, except for the issues identified in Open Item 10.2.3-1 through Open Item 10.2.3-4, the staff finds that the material specifications and the associated processing procedures will provide a suitable material for the turbine rotor that will maintain its toughness to resist brittle fracture.

Fracture Toughness:

DCD, Revision 1, Tier 2 Section 10.2.3.2, states that turbine rotors are made from material that has adequate material strength and toughness, while providing high reliability and efficiency during operation. The impact energy and transition temperature requirements for the turbine rotor exceed those specified in ASTM A470, Class 6 and 7. The staff notes that neither the applicant's response to RAI 199-2073, Question 10.2.3-2, nor Section 10.2.3.2 of DCD, Revision 2, provides the method of calculating the fracture toughness value for the turbine rotor material. SRP Section 10.2.3 (Paragraph II.2) lists four acceptable methods for obtaining the fracture toughness properties. Therefore, the staff requested that the test method that will be used for the turbine rotor design be included in the DCD. This is part of Open Item 10.2.3-3 above.

The fracture toughness of the turbine rotor will be 200 ksi√in minimum. Mitsubishi Technical Report MUAP-07028-P, provides the brittle fracture and fatigue analysis for the turbine rotor design. This report used conservative fracture toughness, which is lower than 200 ksi√in to account for unexpected lower fracture toughness values during fabrication. The evaluation of the above report is discussed in Section 3.5.1.3 of this SE. The staff concluded that using the conservative fracture toughness in the analysis provides an acceptable approach for determining the integrity of the rotor. However, it should be verified that the material property assumptions in the turbine missile probability analysis, Mitsubishi Technical Report MUAP-07028, are bounding for the actual fabricated turbine rotor. Therefore, the staff requested in RAI 199-2073, Question 10.2.3-3, that the ITAAC (Commitment 2) in Table 2.7.1.1-1 of DCD, Revision 1, Tier 1, Section 2.7.1.1, specify acceptance criteria that determine that the as-built turbine material properties and in-service inspection and testing plans meet the requirements of the turbine missile probability analysis (Mitsubishi Technical Reports MUAP-07028, "Probability of Missile Generation from Low Pressure Turbines," and MUAP-07029, "Probabilistic Evaluation of Turbine Valve Test Frequency"). In response to the staff's RAI 199-2073, Question 10.2.3-3, dated March 10, 2009, the applicant will revise Section 2.7.1.1 of DCD, Revision 1, Tier 1 to state the following:

> "Inspections and tests of the as-built LP rotors shall be conducted to verify that the as-built data and calculated toughness curves satisfy the material properties assumptions in the turbine rotor analysis, which determines the turbine maintenance program and inspection interval to meet the requirements of the turbine missile probability analysis (Mitsubishi Technical Reports MUAP-07028, and MUAP-07029)."

Based on the applicant's response, the acceptance criteria for the ITAAC (Commitment 2) in Table 2.7.1.1-1 of DCD, Revision 1, Tier 1, Section 2.7.1.1, provides sufficient information to confirm that the as-built turbine rotor material properties will be bounded by the turbine missile probability analysis (Mitsubishi Technical Reports MUAP-07028, and MUAP-07029). In Revision 2 of the DCD, Table 2.7.1.1-1 and Section 2.7.1.1.1 were revised to provide appropriate acceptance criteria for the ITAAC (renumbered 2.a). This is acceptable to the staff as discussed above.

<u>Turbine Rotor Design:</u>

DCD, Revision 2, Tier 2 Section 10.2.3.4, "Turbine Rotor Design," states that the turbine assembly is designed to withstand normal operating conditions, and anticipated transients resulting in a turbine trip without loss of structural integrity. The staff concluded that the design over-speed of the turbine (120) percent was found to be at least 5 percent above the highest anticipated speed resulting from a loss of load, since a mechanical over-speed trip device closes the main turbine stop and reheat valves at 110 percent of the rated speed, and an independent electrical over-speed trip system will close the valves at 111 percent of the rated speed. In addition, the turbine rotor is design so that the combined stresses at the design over-speed of the turbine (120 percent) will not exceed 0.75 of the minimum specified yield strength of the material. This was found to be consistent with the guidance provided in SRP, Section 10.2.3 (Paragraph II.4), and therefore is acceptable.

Furthermore, the fully integral low-pressure turbine rotor is fabricated from a single ingot of alloy steel, which has a lower failure rate than the shrunk-on discs. Since the fully integral rotor does not have keyways, there are no stress risers to concentrate the stresses locally. Since there are no stress risers from keyways, the stresses are lower than in a rotor with shrunk-on discs, and therefore, the potential for crack initiation and growth is reduced when using this fully integral turbine rotor.

DCD, Revision 2, Tier 2, Section 10.2.3.4, specifies that the non-bored design of the HP and LP turbine rotor provides the necessary design margin by virtue of its inherently lower centerline stress. Also the use of solid rotor forgings was verified by an evaluation of the material removed from center-bored rotors for fossil power plants, and this evaluation demonstrated that the material at the center of the rotors satisfied the rotor material specification requirements. The staff noted that DCD, Revision 2, Tier 2, including Section 10.2.3.3, implies that there is an option for the rotor to be bored or solid (non-bored). Typically each as-built rotor has destructive testing performed at various locations (outside periphery and internal bore region) to ensure homogeneity and acceptable material properties and it is also ultrasonically inspected (outside periphery and internal bore region) to ensure the integrity of the turbine rotor.

In its response to RAI 199-2073, Question 10.2.3-5, dated March 10, 2009, the applicant stated that due to its drum shape, the ultrasonic inspection of the turbine rotor will be performed prior to gashing (final outside periphery machining) so that 100 percent of the ultrasonic inspection can be performed on the turbine rotor. However, it also states that as ultrasonic testing technology advances, potential defects at the center core region will be detected. Therefore, this implies that currently, ultrasonic inspection is not capable of ensuring the integrity of non-bored turbine rotors at the center region. The integrity of non-bored turbine rotors cannot be verified, since the non-destructive examination is not capable of detecting defects at the center core region and destructive testing cannot be performed on non-bored rotors (as discussed above) to confirm the material properties. Therefore, the non-bored rotor design should be deleted from the DCD, or specific non-destructive testing should be provided, which takes into consideration that the presence of any internal defects of the as-built rotor cannot be confirmed. Also, appropriate operating experience that justifies that the integrity of the solid turbine rotor can be maintained should be provided. This is considered part of Open Item 10.2.3-4.

Pre-Service Inspection

The rotors are rough machined prior to heat treatment and a 100 percent volumetric (ultrasonic) examination is performed. A surface magnetic particle and visual inspection is performed on the finished-machined rotor. The acceptance criteria for the magnetic particle test is that all finished-machined surfaces of the turbine rotor will be free of all flaws in the bore or other highly stressed areas.

The staff finds that the acceptance criteria for the above examinations are more restrictive than those specified for Class 1 components in the ASME Code, Sections III and V. This includes the criteria that subsurface ultrasonic indications will be removed or evaluated to verify that they do not grow to a size that compromises the integrity of the turbine rotor during its service life. The staff confirmed that in DCD, Revision 2, the applicant modified Section 10.2.3.3 as stated in its response to RAI199-2073, Question 10.2.3-4, dated March 10, 2009, to correct a typographical error. Therefore, the staff finds the non-destructive examination of the turbine rotors to be acceptable since they will be inspected in accordance with the ASME Code.

The fully bladed turbine rotor assembly is also spin tested at 120 percent overspeed (design overspeed of 2160 revolutions per minute (rpm)) prior to service. The guidelines in SRP Section 10.2.3, "Turbine Rotor Integrity," specify that a spin test is to be performed at 5 percent above the highest anticipated speed. For the US-APWR turbine design, the highest anticipated speed (1998 rpm) is 111 percent of normal speed (1800 rpm). Since 5 percent above 1998 rpm is 2098 rpm, the spin test will be performed at the design overspeed of 2160 rpm, which exceeds the 2098 rpm required by SRP Section 10.2.3. Therefore, the staff finds that a spin test of the turbine rotor assembly performed prior to service at the design overspeed of 2160 rpm is acceptable, and ensures the turbine rotor assembly will maintain its structural integrity during an overspeed event. This meets the guidelines specified in SRP Section 10.2.3. In addition, this spin test is part of the testing specified in ITAAC Commitment 2a of Table 2.7.1.1-1 in DCD, Revision 2, Tier 1, Section 2.7.1.1.

Since the proposed pre-service inspection and acceptance criteria exceed the requirements of the ASME Code, Section III and V and are consistent with the guidelines of SRP Section 10.2.3, the staff finds them acceptable.

The initial turbine rotor condition provides a baseline for future inservice inspections to ensure that no flaws will propagate resulting in the fracture of the turbine rotor and generation of potential missiles.

Inservice Inspection

DCD, Revision 2, Tier 2, Section 10.2.3.5, "Inservice Inspection," states that the turbine rotor inservice inspection program consists of visual, surface and volumetric examination to inspect the turbine rotor assembly. The test program includes the disassembly of the turbine and inspection of all parts, including parts normally considered inaccessible, such as couplings, coupling bolts, turbine rotors and LP turbine blades. This program is based on the bounding turbine missile analysis, Mitsubishi Technical Report MUAP-07028, "Probability of Missile Generation from Low Pressure Turbines," and operating experience of similar units. Each rotor is visually and magnetic particle tested on all accessible areas. The specific inservice inspection for the HP and LP turbine rotor assemblies is to be performed at intervals equal to or less than 10 years. The staff finds the inspection interval acceptable since it is bounded by the turbine missile probability analysis.

The inspection program also includes a 100 percent ultrasonic inspection of the side entry blade grooves, which are of a Christmas tree design. However, in its response to RAI 199-2073, Question 10.2.3-7, dated March 10, 2009, the applicant stated that the side entry blade grooves will not be ultrasonically inspected. Instead, a magnetic particle inspection will be performed every 10 years for both the HP and LP turbine rotors. The staff notes that typically the inspections performed on turbine rotor components include visual, magnetic particle and volumetric (i.e., ultrasonic) as outlined in SRP, Section 10.2.3 (Paragraph II.5), SRP, Section 3.5.1.3 (Paragraphs II.4 and II.5), and the ASME Code. However, Section 10.2.3.5 of Revision 2 of the DCD still specifies an ultrasonic inspection of the side-entry blade grooves in the inspection plan. Therefore, since a volumetric examination is included in the inspection plan, the staff finds the inspection plan for a bored turbine rotor specified in Section 10.2.3.5 of Revision 2 of the DCD acceptable since it meets the guidelines of SRP, Section 10.2.3. However, the integrity of non-bored turbine rotors cannot be verified, since the ultrasonic inspections are not capable of detecting defects at the center core region on non-bored rotors based on the applicant's response to RAI 199-2073, Question 10.2.3-5, (as discussed above). This is part of Open Item 10.2.3-4.

Since the bounding turbine missile analysis, Mitsubishi Technical Reports MUAP-07028, "Probability of Missile Generation from Low Pressure Turbine," which demonstrates that the probability of generating turbine missiles is less than 1 x 10-4 for an inspection interval longer than 10 years, along with the operating experience of similar units, the staff finds the inservice inspection program and interval to be acceptable, and consistent with the guidelines in SRP, Section 10.2.3.

The maintenance and inspection program also includes turbine valve inspection and testing of the overspeed protection system, which is based on Mitsubishi Technical Report MUAP-07029, titled "insert title." The test program includes the dismantling and inspection of at least one main steam stop valve, one main steam control valve, one reheat stop valve and one intercept valve every four years during scheduled refueling outages. The inspections consist of a visual and surface examination of all valve internals. The turbine valve testing is performed at quarterly intervals. Mitsubishi Technical Report MUAP-07029 demonstrates that the probability of generating turbine missiles with quarterly valve testing is less than 1 x 10-5. This is less than the criteria in Regulatory Guide (RG) 1.115, "insert title," of 1 x 10-4 for a favorable oriented TG as discussed in Section 3.5.1.3 of this SE. The bounding analysis reports, Mitsubishi Technical Reports MUAP-07028 and MUAP-07029 are discussed in Section 3.5.1.3 of this SE. Therefore, the staff finds the test program and test interval acceptable, since it meets the criteria of RG

1.115 and is consistent with the guidelines of SRP Section 10.2.3.

Therefore, the staff finds the inservice inspection of the turbine rotor and inservice testing of the overspeed protection system, except for the issue identified in Open Item 10.2.3-4, acceptable, since it meets the guidelines of SRP, Section 10.2.3, to ensure that the turbine rotor integrity is maintained to preclude the generation of missiles, as required by GDC 4 of 10 CFR Part 50, Appendix A.

Evaluation of Combined License (COL) Information Items

DCD, Revision 1, Tier 2, Section 10.2.5, and Table 1.8-2-1 DCD list COL Information Item COL 10.2(1) for inservice inspection of the TG as follows:

> *The Combined License Applicant is to develop turbine maintenance and inspection procedure and then to implement prior to fuel load. Plant startup procedures including warm-up time will be completed therein.*

The staff finds that this COL information item does not specify that it will be consistent with the inservice inspection plan outlined in DCD, Revision 1, Tier 2, DCD Section 10.2.4, and that it meets the requirements of the bounding turbine missile probability analysis.
However, per RAI 199-2073, Question 10.2.3-3, the staff requested this information to be a part of the ITAAC (Commitment 2) from Table 2.7.1.1-1 of the DCD, Revision 1, Tier 1, Section 2.7.1.1. The ITAAC is used to verify that the COL information item is performed by the COL holder, since the material properties and the inspections will not be verified until the turbine rotor is fabricated. In response to the staff's RAI 199-2073, Question 10.2.3-3, the revised Section 2.7.1.1 of the DCD provides sufficient information to confirm that the as-built turbine rotor material properties will be bounded by the turbine missile probability analysis (Mitsubishi Technical Reports MUAP-07028, and MUAP-07029), which determines the inspection and testing intervals. The staff confirmed that Revision 2 of the DCD, revised Section 2.7.1.1.1 and Table 2.7.1.1-1 as specified in its RAI response. Therefore, the COL Information Item 10.2(1) is acceptable, in conjunction with the ITAAC (renumbered Commitment 2a) from Table 2.7.1.1-1 of the DCD, Revision 2, Tier 1, Section 2.7.1.1, since it confirms that the COL applicant is required to implement a turbine maintenance and inspection program prior to the fuel load. The ITAAC is then used to verify that the pre-service inspection and tests are performed and that the inspection program (including inspection interval) based on the as-built turbine rotor material properties, is still bounded by the turbine missile probability analyses, Mitsubishi Technical Reports MUAP-07028 and MUAP-07029.

10.2.5 Combined License Information Items

COL information numbers and descriptions from DCD Tier 2, Table 1.8-2, and from DCD Tier 2, Section 10.2.5, "Combined License Information":

Table 10.2-1
US-APWR Combined License Information Items for Section 10.2

Item No.	Description	Section

Table 10.2-1
US-APWR Combined License Information Items for Section 10.2

Item No.	Description	Section
10.2(1)	*Inservice Inspection. The Combined License Applicant is to develop turbine maintenance, inspection and test procedure prior to fuel load. Plant startup procedure including warm-up time will be completed therein.*	10.2.3.5

10.2.6 Conclusions

On the basis of its review, the staff concludes that the information provided in the Tier 1, Section 2.7.1.1 and Tier 2, Section 10.2 of the DCD, related to the TG system, in particular turbine control and overspeed protection, with the exception of Open Item 10.2-1 and Confirmatory Item 10.2-1, it is sufficient to perform the design basis functions and meet the regulatory criteria discussed identified earlier in this section of this SE. The applicant has met these requirements based on the following considerations:

1. The design of the TG will control the speed of the turbine under all operating conditions and will ensure that the turbine speed will not exceed 120 percent of the rated speed following a load rejection while operating at full power. The turbine overspeed protection system consists of a primary mechanical and backup electrical emergency overspeed trip device with built-in redundancy and diversity features.

2. The turbine is provided with fail-safe components and subsystems, and will not have adverse affects on safety-related SSCs in the plant under abnormal operating conditions of the TG system and its subsystems and components.

The staff also concludes that the applicant has established ITAAC to perform proper testing and inspections of the turbine control systems to ensure turbine trips on receiving overspeed trip signals. Further, the staff concludes that the TG is designed to allow periodic tests, which are to be performed while the plant is in operation.

With regard to turbine rotor integrity, the staff further concludes that with the exception of Open Items 10.2.3-1 through 10.2.3-4, the integrity of the turbine rotor will be acceptable and will meet the requirements of GDC 4 of Appendix A to 10 CFR Part 50, since the turbine rotor assemblies are conservatively designed and will use suitable materials with acceptable fracture toughness that will be inspected before and during service. This provides reasonable assurance that the probability of generating a turbine missile from a turbine rotor failure is low during normal operation, including transients up to design overspeed.

In conclusion, with the exception of the open and confirmatory items cited above, the staff considered the information in the DCD concerning the TG to be sufficient. The design conforms to the requirements specified by GDC 4 and 10 CFR 52.47(b)(1) and meets the guidance provided in SRP, Section 10.2. Therefore, the staff concludes that the design of the TG is acceptable, pending the resolution of the open items and confirmatory items.

10.3 Main Steam Supply System

10.3.1 Introduction

The main steam supply system (MSSS) transports steam produced in the SGs to the HP turbine of the main turbine-generator and to the moisture separator reheater from system warm-up to valves wide open turbine conditions. The MSSS also supplies steam to the Feedwater Heater No. 5 to remove corrosive gases entrained in the incoming feedwater, turbine gland seal system (TGSS), the emergency feedwater (EFW) pump turbine(s), and as backup to the auxiliary steam system.

10.3.2 Summary of Application

Tier 1: The Tier 1 information associated with this section is found in Tier 1 Section 2.7.1.2 of the US APWR DCD.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.3 of the US APWR DCD, summarized in part, as follows:

The MSSS is a steam transport system consisting of piping and valves and associated instrumentation. Main steam supply piping and components are located within the containment, in the main steam/feedwater piping area in the reactor building and the TB. The system includes the following major components:

- Main steam piping from the SG outlet stem nozzles to the main turbine stop valves.
- Main steam isolation valve (MSIV) and main steam bypass isolation valve (MSBIV) in each steam line.
- Main steam check valve (MSCV) in each main steam line.
- Main steam safety valves (MSSVs), main steam relief valve (MSRV) and main steam depressurization valve (MSDV) in each main steam line.
- Main steam relief valve block valve (MSRVBV) in each main steam line.
- Main steam branch line from each main steam line to emergency feedwater pump turbine.
- Turbine bypass valves (TBVs), described in US APWR DCD Section 10.4.4, “Turbine Bypass System.”

ITAAC: The ITAAC associated with Tier 2 Section 10.3 are given in Tier 1 Section 2.7.1.2.2 of the US APWR DCD.

TS: The TS associated with Tier 2 Section 10.3, are given in Tier 2, Chapter 16 of the US-APWR DCD as follows:

- Section 3.7.1, “Main Steam Safety Valves (MSSVs)”
- Section 3.7.2, “Main Steam Isolation Valves (MSIVs)”

- Section 3.7.4, “Main Steam Depressurization Valves (MSDVs)”

10.3.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are provided in Section 10.3, “Main Steam Supply System,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Revision 4, and are summarized below. Review interfaces with other SRP sections are also provided in SRP Section 10.3, Item I, “Review Interfaces.”

1. GDC 2, “Design Bases for Protection Against Natural Phenomena,” as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

2. GDC 4, “Environmental and Dynamic Effects Design Bases,” with respect to safety-related portions of the system to withstand the effects of external missiles, internal missiles, pipe whip and jet impingement forces associated with pipe breaks.

3. GDC 5, “Sharing of Structures, Systems, and Components,” as it relates to the capability of shared systems and components important to the safety to perform the required safety functions.

4. GDC 34, “Residual Heat Removal,” as it relates to the system function of transferring residual and sensible heat from the reactor system in indirect-cycle plants.

5. 10 CFR 50.55a, which requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

6. 10 CFR 50.63, “Loss of all alternating current power,” as it relates to the ability of a plant to withstand for a specified duration and then recover from a station blackout (SBO).

7. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC regulations.

8. GDC 1, “Quality Standards and Records”, which requires that SSC’s important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the important of the safety functions to be performed..

9. GDC 35, “Emergency Core Cooling,” requires that for ferritic pressure-retaining components of a critical nature, the containment is assured, in part, by requiring minimum fracture toughness performance of the materials from which they are fabricated.

10. 10 CFR Part 50, Appendix B, “Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants,” which establishes quality assurance requirements for the design, construction, and operation of those SSCs of nuclear plants

that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

10.3.4 Technical Evaluation

10.3.4.1 System Design and Conformance to GDCs

The NRC staff reviewed the MSSS design, described in DCD Tier 1 and Tier 2 sections in accordance with SRP, Section 10.3. The acceptability of the system is based on meeting the requirements of the GDCs and the SRP acceptance guidance as described, above, in the regulatory basis section of this SE.

In DCD Tier 2 Section 10.3.2.1, "General Description," and Section 10.3.2.2, "Main Steam System - Detailed Description," the DCD identifies the major piping and valves and describes the design details of the MSSS, which include pipe sizing, velocities, and pressure drops. The MSSS of the US-APWR DCD consists of four SGs, a main TG (including MSRs), and associated piping, valves, and instrumentation. Each of the four steam lines connects to one of the SG outlet nozzle and terminates in the TB at each of four turbine stop valves. The system consists of safety-related, as well as, nonsafety-related portions. The safety-related portions of the MSSS include piping and valves between each SG outlet nozzle and its respective MSIV. The remainder of the system and equipment, including the main turbine, are nonsafety-related. Under accident conditions, the MSSS isolates the SGs and the safety-related portion of the system from the nonsafety-related portions. The DCD provides the schematics of the MSSS in DCD, Tier 2, Figures 10.3-1 through 10.3-4, "Main Steam Supply System Piping and Instrumentation Diagram."

In Section 10.3.2.3, "Component Description," the DCD provides design parameters for the major piping and valves. The major components of the MSSS include, but are not limited to, a MSIV and a MSBIV in each of the main steam lines. These valves isolate the secondary side of the SGs to prevent the uncontrolled blowdown of more than one SG and isolate the non safety-related portions of the system. Each main steam line also contains one MSCV, six MSSVs, one MSRV (credit for the MSRV is not taken for safe shutdown), one MSDV and a MSRVBV. A branch line from each main steam line to the EFW pump turbine is also part of the MSSS. Also, the design data for the MSSS and the associated piping and valves is provided in the following Tier 2 DCD tables: Table 10.3.2 -1, "Main Steam Supply System Design Data," Table 10.3.2-2, "Main Steam System Valves," Table 10.3.2-3, "Main Steam and Feedwater Piping Design Data," and Table 10.3.2-4, "Main Steam Branch Piping Design Data (2.5-Inch and Larger)." Furthermore, the DCD identifies the industry codes and standards for the design of the MSSS piping and valves.

In DCD Tier 2 Section 10.3.1.1, "Safety Design Bases," the US-APWR DCD stipulates the regulatory criteria and the applicable industry codes and standards that are used in the design of the MSSS. Also, the DCD identifies the quality groups and seismic classifications for the MSSS piping and component design for the safety-related and non-safety-related portions of the system. Further, in Section 10.3.1.2, "Non Safety Power Design Bases," the DCD described the MSSS design bases with respect to functional capabilities.

Regarding conformance to the codes and standards for MSSS piping and components, Item 3, Section III, "Review Procedure," of the SRP, Section 10.3 describes that the essential portions of the MSSS should be designed to Quality B and/or Seismic Category I requirements. In DCD Section 10.3.1.1, the DCD describes that the main steam lines (i.e., piping, valves, and other components) from the SGs up to and including the fixed restraint downstream of the MSIVs, and also the branch lines from the main steam piping up to and including the first isolation valve, are designed and constructed according to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, "Rules for Construction of Nuclear Facility Components," Class 2 (equivalent to Quality Group B), Seismic Category I. The remaining components (i.e., piping and valves) of the MSSS up to the turbine stop valve and second stage reheaters are located outside the reactor building, and they are nonsafety-related and non-seismic. These components will be designed in accordance with ASME B31.1 "Power Piping Code." Additionally, DCD Tier 2 Section 3.2, "Classification of Structures, Systems, and Components," and Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," provide the quality group and seismic design classification details of components and equipment of the MSSS. The staff reviewed these ASME codes and classifications of the US-APWR MSSS components and verified that they conform to the codes and standards as described in the SRP guidelines.

Conformance to GDC 2, "Design Bases for Protection Against Natural Phenomena," as related to protection against natural phenomena, requires that SSCs which are important to safety are designed to withstand the effects of postulated local natural phenomena, such as earthquakes, tornadoes, and floods without loss of the capability to perform safety functions. DCD Tier 2 Section 10.3.1.1, "Safety Design Basis," states that the safety-related portions of the MSSS are protected from the effects of wind and tornado and flood protection, and the system is seismically designed. Also, DCD Tier 2 Section 10.3.3, "Safety Evaluation," states that the safety-related portions of the MSSS are located in the containment and the main steam/feedwater piping area of the reactor building, which are designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles and other natural phenomena. Thus, the US-APWR assures that the SSCs, which are important to safety of the MSSS, can withstand the effects of natural phenomena, hence ensuring the capability of the system to perform its safety functions. The staff reviewed DCD Tier 2 Figures 10.3-1 through 10.3-4 and confirmed the locations of the safety-related portions of the MSSS as stated in the DCD. Based on the above discussion, the staff concludes that the MSSS design conforms to the requirements of the GDC 2 criteria, and therefore, the staff finds the MSSS design acceptable as related to withstanding the effects of natural phenomena.

Conformance to GDC 4, "Environmental and Dynamic Effects Design Bases," as related to the environmental and dynamic effects, requires that the safety-related portions of the MSSS are designed to withstand the effects of externally and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks. In addition, the system design should adequately consider water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed and should assure that operating and maintenance procedures include adequate precautions to prevent these effects. The system design should also include protection against water entrainment. DCD, Tier 2, Section 10.3.1.1 identifies these GDC 4 requirements and states that the proposed design of the MSSS includes suitable protection so that the dynamic effects, including internally generated missiles, pipe whip and discharging fluids due to equipment malfunctions and external events, do not pose a threat to the systems integrity. Additionally, DCD, Sections 3.5, "Missile Protection," and 3.6, "Protection Against Dynamic Effects Associated with Postulated Rupture of Piping," describe the protection of the safety-related portions from missiles and dynamic effects associated with the postulated rupture of piping. Further, the environmental design is described in DCD, Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment" of the DCD. However, during its review of DCD, Section 10.3.2.4, "System Operation," the NRC staff noted that the applicant did not address the issue of water (steam) hammer, relief valve discharge loads, and water entrainment effects as described in GDC 4 (“SRP Acceptance Criteria,” Item II of SRP, Section 10.3). Therefore, the staff requested the applicant in US-APWR RAI 329-1860, Question 10.3-1, dated April 8, 2009, to provide additional information to address these effects.

Further, Item 1 in Section IV, “Evaluation Findings” of SRP, Section 10.3 describes that the applicant will review operating and maintenance procedures to alert plant personnel to the potential for, and means to minimize, water (steam) hammer occurrences, and this commitment is to be stated in the applicant’s safety analysis report. However, Section 10.3 of the DCD does not address operating and maintenance procedures that include any precautions to avoid the water (steam) hammer or water entrainment effects. Also, the DCD does not address any COL information items for the COL applicants to develop and implement these procedures. Therefore, the staff requested that the applicant, in US-APWR RAI 329-1860, Question 10.3-1 dated April 8, 2009, to provide additional information to the staff, and also to update the DCD to address these effects. Further, the staff requested the applicant to provide a COL information item to ensure procedures are established to preclude water (steam) hammer and water entrainment effects. The staff also requested the applicant to provide information pertaining to any analyses performed, if any, in this regard.

In its response to RAI 329-1860, Question 10.3-1, dated May 26, 2009, the applicant stated that the MSSS components and piping supports consider steam hammer forces resulting from rapid closure of the turbine stop valves, as well as fluid forces resulting from safety and relief valve operations. (The turbine stop valves have the most rapid closure time in the system). The applicant further stated that DCD, Section 3.12.5.3.5, “Fluid Transient Loads,” describes (steam) hammer and relief valve discharge loads. Regarding water (steam) hammer, relief valve discharge loads, and water entrainment effects, the applicant committed to revise the DCD to add Section 10.3.2.4.3, “Water (Steam) Hammer Prevention,” as described below. Additionally, the DCD revision will include an additional COL information item to develop and implement operating and maintenance procedures to avoid water (steam) hammer, relief valve discharge loads, and water entrainment effects. The following subsection will be added to the DCD Tier 2 Section 10.3.2.4:

10.3.2.4.3 Water (Steam) Hammer Prevention.

The MSS design considers water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed. Refer to DCD Subsection 3.12.5.3.5, “Fluid Transient Loads,” for a description of steam hammer caused by rapid valve closure and relief valve discharge loads in the piping analysis.

MSIV operation and drain pot operation, considers steam line water entrainment effects. Before opening the MSIV during plant start-up, main steam piping downstream of the MSIV is warmed gradually by opening the MSBIV; hence, water slug formation in the condensate is prevented. The automatic power operated valve attached parallel to the steam trap opens when the drain pot high level switch activates, and the high level alarm is annunciated to the MCR to give warning to the operator.

The COL Applicant is to provide operating and maintenance procedures including adequate precautions to prevent water (steam) hammer, relief valve discharge loads and water entrainment effects in accordance with NUREG-0927, “Evaluation of Water Hammer Occurrence in Nuclear Power Plants.” The procedures should address:

- Prevention of rapid valve motion.

- Introduction of voids into water-filled lines and components.
- Proper filling and venting of water-filled lines and components.
- Introduction of steam or heated water that can flash into water-filled lines and components.
- Introduction of water into steam-filled lines or components.
- Proper warm-up of steam-filled lines.
- Proper drainage of steam-filled lines.
- The effects of valve alignments on line conditions.

Additionally, the following COL Information Item 10.3(3) will be added to DCD, Section 10.3.7 as shown below:

> COL 10.3(3) Operating and maintenance procedures for water hammer prevention: The Combined License Applicant will develop a milestone schedule for implementation of the procedure.

The staff reviewed the applicant's response to RAI 329-1860, Question 10.3-1, as described above and determined that the proposed revisions to the DCD meet the requirements of GDC 4 as identified in the staff's RAI. Therefore the applicant's response is acceptable and the staff's concern described in the US-APWR RAI 329-1860, Question 10.3-1 is resolved. Later, the staff verified that the applicant incorporated Section10.3.2.4.3 and the COL Information Item 10.3(3) in Revision 2 of the US-APWR DCD as committed above.

Conformance to GDC 5, "Sharing of Structures, Systems, and Components," as related to sharing of SSCs, requires that SSCs which are important to safety shall not be shared by nuclear power units unless it can be shown that such sharing will not significantly impair the ability to perform the intended safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. In DCD, Section 10.3.1.1, the US-APWR DCD states that no equipment of the MSSS is shared between units to preclude consequential effects of malfunctioning components within the system. Therefore, the staff finds that the requirements of GDC 5, as related to the capabilities of shared systems, are not applicable to the MSSS of the US-APWR DCD.

MSSS compliance with GDC 34, "Residual Heat Removal," as it relates to residual heat removal (RHR), is described in SRP, Section 10.3. In Item 4 of the "SRP Acceptance Criteria," the guidance states that acceptance of the system, as related to GDC 34, is based on meeting: (a) the positions in Branch Technical Position 5-4, as it relates to the design requirements for the RHR, and (b) Issue Number 1 of NUREG-0138, "Staff Discussion to Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director NRR to NRR staff," as it relates to credit being taken for all valves downstream of the MSIVs to limit blowdown of a second SG, if a steam line were to break upstream of the MSIV.

With respect to Item (a) above, the functional requirement for the safety-related portion of the MSSS is to transfer residual and sensible heat from the reactor coolant system (RCS) in pressurized water reactor (PWR) plants (Item 3, Section IV, “Evaluation Findings,” of SRP, Section 10.3). The MSSS of the US-APWR performs the function of cooling the RCS by venting SG steam to the atmosphere by the MSDVs. In DCD, Section 10.3.2.4.2, “Emergency Operation,” the DCD describes that the MSDVs are used to remove the reactor decay heat and primary systems (i.e., RCS) sensible heat in order to cool down the RCS to the conditions at which the RHR can perform the remaining cooldown function. In the event that one MSDV is not available, the US-APWR DCD further states that the remaining MSDVs are sufficient to cool down the plant. The valves are designed such that they are capable of cooling at a rate of 10^{o}C (50^{o}F) per hour and are controlled from the MCR. DCD Table 10.3.2-2, “Main Steam System Valves,” provides the design details of these valves. Furthermore, DCD Section 10.3.2.3.3, Item B, “Main Steam Depressurization Valve,” describes that the MSDVs are designed to provide controlled removal of reactor decay heat, in conjunction with the emergency feedwater system (EFWS), during safe shutdown after a plant transient, accident condition or emergency condition when the turbine bypass system is not available. The staff reviewed the information provided in the DCD and finds that the APWR design is in compliance with GDC 34 requirements as related to supporting the RHR function and cooling the RCS by depressurizing the SGs, under transient and emergency conditions. However, it is not clear how the motor operated MSDVs will function during an accident coincident with a loss of off-site power (LOOP) and what design features are used in the coping analysis for a SBO event in accordance with 10 CFR 50.63, “Loss of all Alternating Current Power.” Therefore, in US-APWR RAI 329-1860, Question 10.3-2, dated April 8, 2009, the staff requested the applicant, to provide additional information in this regard. Also, the staff requested the applicant to explain the capabilities of the MSDVs cooling the RCS, in case one MSDV is not available.

In its response to RAI 329-1860, Question 10.3-2, dated May 26, 2009, the applicant stated that the MSDVs are three-phase ac motor operated valves, and the valves are activated by converting direct current (dc) from Class 1 E batteries into ac with inverters. Therefore, the valves can be operated during LOOP or SBO condition. The applicant indicated that during a LOOP, battery charging is provided from the Class 1E gas turbine generator (GTG) and from the alternate ac (AAC) GTG during an SBO. However, the applicant stated that MSDVs are not required during a SBO condition, since the plant is designed to maintain a hot-standby condition for more than eight hours during a SBO event, which is further discussed later in this SE. Regarding the MSDVs capability to cool the RCS when an MSDV is not available, the applicant stated that two of the four MSDVs have adequate steam release capacity for RCS cooling. Therefore, failure of one MSDV does not affect the RCS cooling, and therefore failure of one valve is acceptable. Since the applicant is not relying on MSDVs for a SBO condition and the four MSDVs ensure redundancy in case of one failing, the staff’s concern described in US-APWR RAI 329-1860, Question 10.3-2, is resolved.

Further, regarding conformance with GDC 34, as it relates to Issue 1 of NUREG-0138, DCD Section 10.3.2.3.4, “Main Steam Isolation Valves and Main Steam Check Valves” and Section 10.3.2.4.2, “Emergency Operation,” describe that in the event of a main steam line break (MSLB), the MSIVs will limit uncontrolled steam release from one SG. In the event of a MSLB upstream of MSIVs, assuming a single failure of this valve, the broken side is isolated by the MSIVs on the main steam piping of the intact SGs, or the MSCV of the broken line which will prevent the steam blowdown through more than one SG. Check valves located in the main steam supply lines to prevent potential backflow during a steam line break. The DCD further specifies that if the break is on the downstream side of a MSIV, assuming a single failure, the MSIVs of the intact SGs will prevent blowdown from the remaining intact SGs. The MSIVs and the respective MSBIVs are designed to fully close automatically within five seconds. The design parameters of these MSIVs are provided in DCD, Table 10.3.2-2 of the DCD.

Based on the above discussions, the staff finds that the DCD adequately addresses the MSSS compliance with GDC 34 requirements.

With respect to conformance to 10 CFR 50.63, as it relates to a SBO event, DCD Tier 2, Section 10.3.1.1, “Safety Design Bases,” states that the US-APWR is provided with an Alternate Alternating Current (AAC) power source to cope with a blackout event. The regulatory requirement is that each light-water-cooled nuclear plant must be able to withstand, for a specified duration, and recover from a SBO event. The factors that are considered for the SBO duration include, but are not limited to, redundancy and reliability of the onsite emergency ac power sources. Also, DCD Tier 2, Section 8.4, “Station Blackout,” describes the regulatory requirements, recovery from a SBO, an analysis, and other SBO pertinent details. However, the DCD does not provide, nor identify, any details regarding which components of the MSSS are required to be functional and what the emergency power sources are during a SBO event. DCD, Section 10.3.3, “Safety Evaluation,” identifies that redundant power supplies are provided to operate MSIVs for containment isolation. However, the DCD does not address its functionality and emergency power source during a SBO. Additionally, DCD Tier 2, Section 8.4.3, “Combined License Information,” states that no additional information is required to be provided by a COL applicant as related to a SBO. In order to complete a review of this area for compliance with the 10 CFR 50.63 requirements for a SBO event, the staff requested the applicant in US-APWR RAI 329-1860, Question 10.3-3, dated April 8, 2009, to provide design and operating details for the MSSS and its components as related to the SBO. Also, the staff requested the applicant to provide supporting justification that the ac power source to the MSSS components is adequate to withstand and recover from a SBO event.

In its response to RAI 329-1860, Question 10.3-3, dated May 26, 2009, the applicant stated that during SBO events, the plant can be kept in hot-standby condition for more than eight hours. This is accomplished by using 105 percent rated steam flow capacity main steam valves for decay heat removal in conjunction with feedwater supply from the EFWS. The applicant further stated that MSDVs are not required in hot-standby condition. Further it is stated that the AAC GTG can power the MSDV via the Class 1 E power system until power is restored in accordance with DCD Section 8.4.1.4, “Recovery from SBO.” However, this does not address the staff’s RAI with respect to which components of the MSSS are required to be functional and what their emergency power sources are during a SBO event. Additionally, the applicant did not address the MSIV functionality for containment isolation and emergency power source during a SBO. Therefore, the staff requested in a follow up RAI (US-APWR Supplemental RAI 431-3274 Question 10.3-3-1) further clarification and/or additional information to provide a complete response to US-APWR RAI 329-1860, Question 10.3-3, that there is sufficient margin for AAC GTG to supply power to critical MSSS components.

In its letter (UAP-HF-09436) dated August 28, 2009, the applicant provided its response to staff’s Supplemental RAI 431-3274, Question 10.3-3-1 and stated that the critical components that are required for a SBO event are MSSVs, MSIVs, and MSBIVs. The applicant further noted as follows: During a SBO, the MSSVs are opened when the reactor pressure reaches its set-point pressure to remove the decay heat from the RCS in conjunction with the EFW system. These MSSVs are spring loaded valves, and therefore do not need any power to be opened. Also, during a SBO event, the MSIVs and MSBIVs are closed to keep inventory of the SGs. Each of these valves performs its function in conjunction with an associated solenoid valve. The MSIVs are gate valves, which use the valve’s inside pressure for self-closure. During power operation, the solenoid valves are energized from Class 1E dc bus (to keep itself in closed position). In a SBO event, to close the MSIV, the solenoid valve is de-energized to deliver MSIV's inside-pressure to the upper piston chamber of the MSIV resulting in its closure. Therefore, the dc power is not required for the MSIVs to close during a SBO. In case of the MSBIVs, these are fail-closed air-operated valves and have solenoid valves to release air from its diaphragm. During power operation, the solenoid valves are de-energized to release air from the (MSBIVs) diaphragm, and the MSBIVs are kept closed. In this case, also the dc power is not required for the MSBIV to close during a SBO. Therefore, the applicant stated that no power source is required for the MSSS during a SBO. The staff finds the applicant's response acceptable, since it addresses the functionality of the valves, their power sources, and containment isolation during a SBO event. Further, these valves are closed in fail-safe positions with no power, and therefore the staff's concerns in the Supplemental RAI 431-3274, Question 10.3-3-1 are resolved.

10.3.4.2 Inspection and Testing Requirements

MSSS components are inspected and tested as part of the initial test program as discussed in DCD Section 14.2, "Initial Plant Test Program." The staff's evaluation of the US-APWR initial test program is addressed in the Final Safety Evaluation Report (FSER), Section 14.2. In DCD Section 10.3.4, "Inspections and Tests," the applicant states that the MSSS is designed to allow system testing for both normal and emergency operating modes which includes applicable protection system components. A description of inservice testing of ASME Code, Section III, Class 2 and 3 components is discussed in DCD Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." The staff's evaluation of the US-APWR inservice testing program is addressed in FSER Section 3.9.6. The applicant also stated that safety-related components of the MSSS are designed and located to permit preservice and inservice inspection. A description of inservice inspection of ASME Code, Section III, Class 2 and 3 components are discussed in DCD Section 6.6, "Inservice Inspection and Testing of Class 2 and 3 Components." The staff's evaluation of the US-APWR inservice inspection program is addressed in FSER Section 6.6.

10.3.4.3 Secondary-Side Water Chemistry

Secondary-side water chemistry is discussed in FSER, Section 10.4.6, "Condensate Polishing System," in conjunction with the condensate polishing system.

10.3.4.4 Steam and Feedwater System Materials

As set forth below, the staff reviewed and evaluated the information in DCD Tier 2, Section 10.3.6, "Steam and Feedwater Materials," to ensure that the materials and fabrication of ASME Code Class 2 and MSSS and condensate and feedwater system (CFS) components comply with the guidelines detailed in SRP, Section 10.3.6. The staff's findings are discussed below:

10.3.4.4.1 Material Selection and Fabrication of Class 2 and 3 Components

To meet the requirements of GDC 1, "Quality Standards and Records" 10 CFR 50.55a and 10 CFR Part 50, Appendix B, the materials used in the ASME Code Class 2 and 3 portions of the MSSS and CFS must meet the requirements of Section III of the ASME Code and ASME Code Cases listed in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME, Section III." The guidance provided in RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," RG 1.71, "Welder Qualification for Areas of Limited Accessibility," RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ASME Code, Section III, Appendix D-1000, Article D-

1000, “Non-mandatory Preheat Procedures,” for low alloy steels and carbon steels may be used (or followed) to implement the requirements of GDC 1, 10 CFR 50.55a and 10 CFR Part 50, Appendix B.

DCD Tier 2, Tables 10.3.2-3, “Main Steam and Feedwater Piping Design Data,” and 10.3.2-4, “Main Steam Branch Piping Design Data,” list material specifications and grades for ASME Code, Class 2 and 3 main steam and feedwater piping. Table 10.3.2-5, “ASME Material Specifications with Filler Metal Specifications and Classification for each Welding Process,” lists weld filler material specifications and classifications. The staff reviewed the aforementioned tables and confirmed that the materials listed, meets ASME Code Section III requirements and are therefore acceptable.

The guidelines listed in RG 1.50 describe the staff’s endorsed methods to control preheat temperatures before post-weld heat treatment when welding low-alloy steel in accordance with ASME Code Section III. ASME Code, Section III, Appendix D, Article D-1000, provides recommended minimum preheat temperatures used to weld carbon steel and low-alloy steel components that are acceptable to the staff as noted in SRP, Section 10.3.6, “Steam and Feedwater System Materials.” DCD, Section 10.3.6.2, “Material Selection and Fabrication,” states that the welding of low-alloy materials conform to the guidance provided in RG 1.50 and the minimum preheat temperatures for carbon and low alloy steel materials conform to the recommendations in ASME Code, Section III, Appendix D, Article D-1000. The staff finds this acceptable because the applicant will conform to staff guidance for the preheating of carbon steel and low alloy steel materials used in ASME Code Class 2 and 3 portions of the MSSS and CFS. However, Table 1.9-1 “US-APWR Conformance with Division 1 Regulatory Guides,” does not list FSER, Section 10.3.6 in the line item for RG 1.50.

In **RAI 250-2143, Question 10.3.6-2**, the staff requested, in part, that the applicant list FSER, Section 10.3.6 in the line item for RG 1.50 in FSER Table 1.9-1. In its response to RAI 250-2143, Question 10.3.6-2, dated April 1, 2009, the applicant provided a proposed revision to Table 1.9-1 that lists FSER, Section 10.3.6.2 in the line item for RG 1.50 as requested by the staff. The staff will verify that the applicant makes the proposed modifications in the next DCD revision. The staff identifies this as **Confirmatory Item 10.3.6-2**

For the nondestructive examination of ferritic steel tubular products, compliance with applicable provisions of ASME Code meets the requirements of GDC 1 and 10 CFR 50.55a. The applicant stated in DCD, Section 10.3.6.2 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. The staff finds this acceptable because it meets ASME Code requirements and the acceptance criteria in SRP 10.3.6.

ASME Code, Section III, requires adherence to the requirements of ASME Section IX for welder qualification for production welds. However, there is a need for supplementing this section of the ASME Code because the assurance of providing satisfactory welds in locations of restricted direct physical and visual accessibility can be increased significantly by qualifying the welder under conditions simulating the space limitations under which the actual welds will be made. RG 1.71 provides guidance to supplement ASME Code, Section IX, in this respect. DCD Tier 2, Section 10.3.6.2 states that for welds in areas of limited accessibility, the qualification procedure is specified in conformance with the guidance of RG 1.71, “Welder Qualifications for Areas of Limited Accessibility.” The applicant also lists DCD Section 10.3.6.2 in the line for RG 1.71in Table 1.9-1. The staff finds this acceptable because the applicant will follow staff guidance regarding performance qualifications for welds that have limited accessibility.

RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” provides procedures acceptable to the staff for cleaning and handling Class 2 and 3 components in the MSSS and the CFS. DCD ,Section 10.3.6.2 states that the cleaning and handling of Class 2 and Class 3 components of the MSSS and CFS are conducted in accordance with the acceptable procedures described in RG 1.37. However, Table 1.9-1 does not list FSER, Section 10.3.6 in the line item for RG 1.37. In **RAI 250-2143, Question 10.3.6-4**, the staff requested, in part, that the applicant list FSER Section 10.3.6 in the line item for RG 1.37 in FSER Table 1.9-1. In its response to RAI 250-2143, Question 10.3.6-4, by letter dated April 1, 2009, the applicant provided a proposed revision to Table 1.9-1 that lists DCD Section 10.3.6.2 in the line item for RG 1.37 as requested by the staff. The staff will verify that the applicant makes the proposed modifications in the next DCD revision. The staff identifies this as **Confirmatory Item 10.3.6-04.**

10.3.4.4.2 Fracture Toughness of Class 2 and 3 Components

DCD, Section 10.3.6.1, “Fracture Toughness,” states that material specifications for pressure retaining components in the safety-related portion of the MSSS and CFS meet the fracture toughness requirements of ASME Code, Section III, Articles NC-2300 (Class 2) and ND-2300 (Class 3) for Quality Group B and Quality Group C components. The staff finds this acceptable because the applicant will meet ASME Code fracture toughness requirements for all ASME Code, Class 2 and 3 components in the MSSS and CFS and thus meets the requirements of GDC 35, “Emergency Core Cooling.”

10.3.4.4.3 Flow-Accelerated Corrosion

The staff notes that, historically, documents such as Generic Letter (GL) 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning,” have referred to FAC as erosion/corrosion. Therefore, FAC and erosion/corrosion are used interchangeably throughout this safety evaluation. In addition to design considerations to minimize erosion/corrosion, GL 89-08 stressed the importance of implementing formalized procedures or administrative controls to ensure continued long-term implementation of a FAC monitoring program for piping and components. Guidance provided by Electric Power Research Institute (EPRI) in NSAC-202L-R3, “Recommendations for an Effective Flow Accelerated Corrosion Program,” includes procedures and administrative controls to ensure that the structural integrity of all carbon steel lines containing high-energy fluids is maintained by minimizing FAC effects. The guidance in EPRI Technical Report NSAC-202L-R3 is a refinement of the guidance developed by the industry that was endorsed by the NRC in NUREG-1344, “Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants,” dated April 1989. The EPRI guidance is applicable to new reactors and remains acceptable to the staff.

The use of FAC resistant materials, specification of an adequate corrosion allowance and appropriate piping design measures are important tools to mitigate the effects of FAC. In addition, a final system design evaluation can be helpful in identifying areas that may require additional design modifications to mitigate the effects of FAC. Given the analytical tools available along with operating experience, system designs should be such that components will not degrade below their minimum design wall thickness for the design life of the systems. It is understood that components which could be potentially susceptible to FAC will be monitored throughout the life of the plant to ensure that, should FAC degradation occur, components will be replaced before they degrade below the minimum design wall thickness. After reviewing DCD, Section 10.3.6.3, “Flow-Accelerated Corrosion,” the staff requested, in **RAI 500-4012,**

Question 10.3.6-12, that the applicant modify the DCD to state that it will perform a final system design evaluation of all ASME Code Class 2 and 3 as well as non-ASME Code systems or portions of systems susceptible to FAC. The staff also requested that the applicant modify the DCD to discuss the design life of MSSS and CFS piping and components exposed to water or wet steam.

In its response to RAI 500-4012, Question 10.3.6-2, dated April 4, 2011, the applicant provided proposed modifications to DCD Section 10.3.6.3 to address the staff's concerns. The staff's evaluation below related to the applicant's design to mitigate FAC is based on the applicant's response to RAI 500-4012, Question 10.3.6-12 and proposed modifications to DCD Section 10.3.6.3 as stated in its April 4, 2011, letter.

DCD, Section 10.3.6.3 states that all safety related and non-safety related piping and components are designed to mitigate the effects of FAC as well as erosion, corrosion and cavitation. The applicant further stated that portions of the systems potentially susceptible to FAC are identified based on NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in US Nuclear Power Plants," dated April 1989, GL 89-08 and operational experience.

Most piping and components are carbon steel, however, the applicant stated that FAC resistant materials are used in portions of systems when required to mitigate the effects of FAC where the use of carbon steel will not provide sufficient protection. The staff reviewed DCD Tables 10.3.2-3, "Main Steam and Feedwater Piping Design Data," and 10.3.2-4, "Main Steam Branch Piping Design Data (2.5-Inch and Larger," and verified that piping segments that would be expected to be susceptible to FAC, such as portions of the feedwater system, are fabricated from 2.25 percent chromium material which has a high resistance to FAC degradation, which the staff finds acceptable. In addition to material selection, the US-APWR design and piping layout includes features that minimize FAC. These features include control of CFS water chemistry, the elimination of high turbulence points where possible, use of long radius elbows, smooth transitions at welds and selection of appropriate pipe diameters to reduce flow velocities to within industry recommended values. The staff finds the applicant's design and piping layout acceptable because the applicant has taken steps to minimize design attributes that create flow characteristics that can promote FAC.

The applicant stated that the required design wall thickness of piping and components includes an increase in wall thickness due to thinning during fabrication, FAC degradation and general corrosion. In addition, the final design will be evaluated when complete, and additional material upgrades will be made where necessary to provide reasonable assurance that the systems maintain their minimum design wall thickness for a design life of 40 years.

Based on the above evaluation, the staff finds that the applicant has taken appropriate measures in its design to mitigate the effects of FAC degradation through material selection, design and piping layout, consideration of wall thinning during fabrication, consideration of wall thinning due to general corrosion and the use of a final piping system design evaluation to provide reasonable assurance that systems maintain their minimum design wall thickness for a design life of 40 years. The staff's evaluation of the applicant's design to mitigate FAC is based on the proposed DCD modifications as stated in the applicant's April 4, 2011, response to **RAI 500-4012, Question 10.3.6-12.** The staff will verify that the applicant makes the appropriate changes in DCD, Revision 4. The staff identifies this as **Confirmatory Item 10.3.6-12**.

In addition to design considerations to minimize FAC as required by ASME Code Section III and described in GL 89-08, an appropriate long-term monitoring program must be implemented to detect the potential wall thinning of high-energy ASME Code, Section III, and non-safety-related piping caused by FAC, following the guidance provided in GL 89-08.

The applicant addresses long-term monitoring for FAC in COL Information Item 10.3(1) which is listed in DCD, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” and DCD, Section 10.3.7, “Combined License Information.” COL Information Item 10.3(1), states “The Combined License Applicant will provide a description of the FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam and are susceptible to erosion-corrosion damage. The description will address consistency with Generic Letter 89-08 and NSAC-202L-R2 and will provide a milestone schedule for implementation of the program.” Although the staff finds that following the guidance in EPRI Technical Report NSAC-202L, “Recommendations for an Effective Flow Accelerated Corrosion Program,” is an acceptable method to address the staff’s concerns described in GL 89-08, the applicant referenced Revision 2 of the EPRI report in lieu of Revision 3 which is the most current revision. In the applicant’s proposed modifications to the DCD, in its response to RAI **500-4012, Question 10.3.6-12**, dated April 4, 2011, the applicant modified DCD Sections 10.3.6.3, “Flow-Accelerated Corrosion (FAC),” and 10.3.7 to reference Revision 3 in lieu of Revision 2 of the EPRI Technical Report NSAC-202L. The staff finds this acceptable because the applicant will reference the most recent revision of the EPRI FAC program guidelines. The staff will verify that the applicant makes the appropriate changes in DCD Revision 4. The staff identifies this as part of **Confirmatory Item 10.0.6-12**.

10.3.4.5 Evaluation of COL Information Items

1. COL 10.3(1) Flow Accelerated Corrosion (FAC) monitoring program.

The COL applicant that references the US-APWR DC will address preparation of a FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam, as described in DCD, Section 10.3.5, “Water Chemistry,” and its subsections. This monitoring program is to address the industry guidelines and requirements in GL 89-08.

2. COL 10.3(3) Operating and maintenance procedures for water hammer prevention.

The COL applicant is to provide operating and maintenance procedures including adequate precautions to prevent water (steam) hammer, relief valve discharge loads and water entrainment effects in accordance with NUREG-0927, “Evaluation of Water Hammer Occurrence in Nuclear Power Plants,” dated March 1984, and a milestone schedule for implementation of the procedures.

The staff reviewed these COL information items and finds them acceptable with the incorporation of the proposed COL Information Item 10.3(3) for the COL applicants to address implementing plant procedures to prevent water (steam) hammer, as described earlier in the applicant’s response to US-APWR DCD RAI 329-1860, Question 10.3-1.

10.3.4.6 ITAAC Considerations

10 CFR 52.47, requires that, a DC application is to contain proposed ITAAC which are necessary and sufficient to provide reasonable assurance, a plant that incorporates the DC is built and will operate in accordance with the design commitments, and meets the provisions of the Atomic Energy Act of 1954 and the NRC regulations. DCD Tier 1, Section 2.7.1.2, “Main

Steam Supply System," of the US-APWR DCD identifies several ITAAC necessary to demonstrate the design commitments of the DCD. The numeric performance values for selected components have been specified in the identified ITAAC. Key parameters of the MSSS design that are used in the safety analysis are also identified in these ITAAC. The staff reviewed these ITAAC, verified them against the design features of the MSSS and finds that the as built system will comply with the approved system design of the DCD.

10.3.4.7 Initial Plant Test Program

The US-APWR DCD proposes the following plant test program with regards to the MSSS.

- Valve testing and inspection: The operability and setpoints of the MSSVs are verified at operating temperature using steam as the pressurization fluid. The operability of each MSRV and MSDV is verified. The MSIVs and MSBIVs are tested to check closing time prior to startup.

- System testing: The MSSS is designed to allow system operation testing for both normal and emergency operating modes. The safety-related components of the system are designed and located to permit pre-service and in-service inspection.

- Pipe testing: The safety-related main steam lines within the containment and main steam/feedwater piping area are visually and volumetrically inspected at installation per ASME code Section XI.

- In-service testing: The structural leaktight integrity and performance of the system components is demonstrated by operation.

10.3.5 COL Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the US-APWR DCD. The table was augmented to include "action required by COL applicant/holder."

Table 10.3-1
U.S-APWR Combined License Information Items

Item No.	Description	Section
10.3(1)	*Flow-Accelerated Corrosion (FAC) monitoring program: The COL applicant is to address preparation of a FAC monitoring program for carbon steel portions of the steam and power conversion systems that contain water or wet steam.*	
10.3(3)	*Operating and maintenance procedures for water hammer prevention. .*	

10.3.6 Conclusions

The NRC staff has reviewed and compared the information within the US-APWR DCD application to the existing licensing basis and relevant NRC regulations and acceptance criteria defined in NUREG-0800, Section 10.3. The staff has concluded that sufficient information has been provided by the applicant in the DCD, and DCD Tier 1 and Tier 2 sections of the MSSS, which supports that the system can perform its safety and non-safety functions.

Regarding the review of DCD Section 10.3.6, with the exception of acceptable resolution of the aforementioned confirmatory items, the staff concludes that the steam and feedwater system materials satisfy the relevant requirements of 10 CFR 50.55a; Appendix A to 10 CFR Part 50, GDC 1 and 35; and Appendix B to 10 CFR Part 50.

In conclusion, the applicant has provided sufficient information for satisfying GDC 2, GDC 4, GDC 5, GDC 34, 10 CFR 50.63, and 10 CFR 52.47(b)(1) criteria, and SRP guidance and acceptance criteria, as described in the regulatory basis of this SE.

10.4 Other Features of the Steam and Power Conversion System

10.4.1 Main Condensers

10.4.1.1 Introduction

The main condenser functions as the steam cycle heat sink, condenses and deaerates the exhaust steam from the main turbine and the turbine bypass system (TBS). The US-APWR main condenser (MC) is a three-shell, single-pass, divided water boxes, and rigidly supported unit. Each shell is located underneath its respective LP turbine. The condenser is equipped with titanium tubes.

10.4.1.2 Summary of Application

FSAR Tier 1: Final Safety Analysis Report (FSAR) Tier 1 Section 2.7.1.3.1, “Design Description,” of the US-APWR DCD, provides a brief description of the MC system. The DCD states that there are no safety-related interfaces with systems outside of the certified design. Also, in Tier 1 Section 2.7.1.3.2, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” the DCD identifies the ITAAC requirements for the MC system. These ITAACs are depicted in Tier 1 Table 2.7.1.3-1, “Main Condenser Inspections, Tests, Analyses, and Acceptance Criteria.”

FSAR Tier 2: The US-APWR FSAR Tier 2 Section 10.4.1, “Main Condensers,” provides the MC system design. The US-APWR MC is a nonsafety-related system and is located in the TB. It is a non-seismic category system and not designed to ASME code classifications. The NRC staff evaluation of the MC system is provided in Section 10.4.4.4, “Technical Evaluation” of this SE.

TS: There are no TS associated with the MC system.

10.4.1.3 Regulatory Basis

Conformance with the applicable requirements of 10 CFR 50, Appendix A, and the provisions of the following additional requirement constitutes an acceptable basis for a satisfactory MC design.

1. GDC 2, “Design Bases for Protection Against Natural Phenomena,” in that failure of the nonsafety-related system or component due to natural phenomena such

as earthquakes, tornadoes, hurricanes, and floods should not adversely affect the safety-related SSCs.

2. GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to provisions being included in the nuclear power unit design to suitably control the release of radioactive materials in gaseous and liquid effluents during normal operation, including anticipated operational occurrences (AOOs). GDC 60 is applicable to the design of the MC system because in PWRs, radioactive materials may be deposited in the main condensers if there is a primary-to-secondary SG tube leak.

3. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 194, and the NRC’s regulations.

In addition to the above GDCs, SRP, Section 10.4.1, “Main Condenser,” subsection II, “Acceptance Criteria”, “SRP Acceptance Criteria,” Item 1.B states that acceptance of the GDC 60 is based on meeting the following:

> “If there is a potential for explosive mixtures to exist, the MC is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, subsection II, “Acceptance Criteria,” SRP Acceptance Criteria, Item 6.”

10.4.1.4 Technical Evaluation

The staff reviewed the design of the US-APWR MC system in accordance with SRP, Section 10.4.1, Revision 3. The staff’s acceptability of the system is based on meeting the requirements of the following GDC criteria:

- GDC 2, as it relates to the failure of a nonsafety-related system or equipment, due to natural phenomena, does not adversely affect safety-related SSCs, and

- GDC 60, as it relates to ensuring that failures of the system do not result in excessive releases of radioactivity from the main condensers to the environment, as it relates to ensure that failures of the system do not result in excessive releases of radioactivity to the environment, do not cause unacceptable condensate quality, and do not flood areas of housing safety-related equipment.

The subsections of US-APWR FSAR Tier 2, Section 10.4.1 describe the MC system design basis; system and component description; operation; safety evaluation; tests and inspections; and instrumentation applications of the US-APWR MCs. Classification of equipment and components is provided in Tier 2, Section 3.2, “Classification of Structures, Systems, and Components” of the DCD. Table 10.4.1-1, “Main Condenser Design Data,” provides the design parameters of the condenser.

Tier 2, Section 10.4.1.1, "Design Basis," states that the US-APWR MC system has no safety-related function, and therefore has no safety-related design basis. The MC system is designed to perform the following functions:

- Receive and condense exhaust steam from the three LP turbines. Provide a collection point for steam, vents, and drains from various components.

- Accommodate up to 68 percent of the valves-wide-open main steam flow directly from the TBS.

- Condense steam without exceeding the maximum allowable condenser backpressure for the main turbine.

- Remove non-condensable gases through the MC evacuation system (MCES).

FSAR Tier 2, Section 10.4.1.2, "System Description," provides a brief description of the US-APWR MC, which is part of the CFS. The CFS is described in Tier 2, Section 10.4.7, "Condensate and Feedwater System," of the DCD. Figures 10.4.1-1 through 10.4.7-4, "Condensate and Feedwater Piping and Instrumentation Diagram," depicts the MC and CFS systems. The MC is a three-shell, single pass, single pressure unit. Each shell is located under their respective LP turbines, which are located in the TB. There is an equalizing pipe that connects each condenser shell at the condenser neck area. Therefore, all three condenser shells operate at the same pressure and temperature. The MCs deaerates the condensate and keeps the dissolved oxygen in the condensate below 10 parts per billion (ppb) during rated power operation. The condensate is drawn from the hotwell of each condenser to a single header which provides suction to the condensate pumps of the CFS. The condenser hotwells are designed for a five minute holdup time. The US-APWR MC system is designed to contain titanium tubes to maintain good corrosion and erosion resisting properties.

Regarding conformance to GDC 2 criteria, based on its design basis and above description, the staff finds that the US-APWR MC system is a nonsafety-related and non-seismically designed system, which is located in the TB. Failure of the MC or its components due to natural phenomena will have no adverse effects on safety-related SSCs, since such components are not located in the TB. Therefore, the staff finds that the US-APWR MC system meets the requirements of GDC 2.

Tier 2, Section 10.4.1.2.1, "System Operation," describes that during normal power operation, exhaust steam from the LP turbines, and also, feedwater heater drains and vents and gland steam condenser drains are directed into the MC shells. Further, during AOOs, the MC shells are designed to receive up to 68 percent of the rated steam from the main steam system. The DCD states that perforated distribution piping or baffle plates are installed to protect: 1) the condenser tubes, 2) LP feedwater heaters located in the condenser, and 3) other condenser components from the turbine bypass steam or high temperature drains entering the condenser shells. The staff finds that these design provisions conform to the SRP, Section 10.4.1, Subsection III, "Review Procedures," Item 3.D as it relates to incorporating provisions into the MC design that will preclude component or tube failures due to steam blowdown from the TBS. Also, the DCD states that in the event of high condenser pressure or trip of all circulating water pumps, or trip of all condensate pumps, the TBS valves are prohibited from opening. The staff finds this acceptable since it meets the SRP guidance provided in Item 3.C of the above SRP review procedure, as it relates to isolating the steam source on loss of condenser vacuum.

Additionally, the DCD states that steam relief blowout diaphragms are provided in the LP turbine outer casings to protect the condenser shells and turbine outer casings, which also, the staff finds, conforms to Item 3.C of the SRP review procedures.

Further, conforming to Item 2.A of the SRP Section III as it relates to controlling and collecting cooling water leakage into the condensate, the DCD states that the MC interfaces with the tube leak detection system to permit sampling of condensate in the condenser hotwell. Should circulating water in-leakage occur, these provisions permit determination of which tube bundle has sustained leakage. This is performed by isolating the CWS from the affected water box of the MC. However, the DCD did not provide further details regarding the provisions that permit detection of which tube bundle is affected by the leakage and what follow up actions will be performed. Therefore, in **RAI 245-2176, Question 10.4.1-1**, dated March 2, 2009, the staff requested the applicant to provide additional information in the DCD as related to the provisions to determine which MC tube bundle is affected by the leakage.

In its response to RAI 245-2176, Question 10.4.1-1, dated March 30, 2009, the applicant stated that the interface between the MC and the tube leak detection system is described in the eighth paragraph of DCD Section 10.4.1.2.1, "System Operation." Also, the applicant referred to Section 10.4.5.3.4.2, "Leakage from/into the System" of the SRP Section 10.4.5, "Circulating Water System," where it is stated that any leakage from the CWS due to tube leakage into the main condenser is detected by the secondary sampling system. Additionally, the applicant stated that the description in Section 10.4.1.2.1 is not clearly written, and therefore the applicant proposed to revise this FSAR section to add that, "The main condenser interfaces with the tube leak detection system as discussed in Subsection 9.3.2 (Process and Post-Accident Sampling Systems) to permit sampling of the condensate in the condenser hotwell." Further, the applicant referred to FSAR Table 9.3.2-4, "Secondary Side Sampling System," which shows that the sample points are located in each tube bundle, and therefore each tube leakage of the MC tube bundle can be detected. According to the DCD, Table 9.3.2-4, the sample points are located in each MC tube bundle hotwell; thereby making it possible to determine which tube bundle is leaking. The staff finds that the applicant's proposed revision resolves the concerns associated with RAI 245-2176, Question 10.4.1-1. Further, the staff verified that in Revision 2 of the DCD, the applicant revised FSAR, Section 10.4.1.2.1 and incorporated the above response.

In Tier 2, Section 10.4.3, "Safety Evaluation," the DCD provides its evaluation of the US-APWR MC system. The DCD states that during normal operation and shutdown, the MC contains insignificant quantity of radioactive contaminants. The DCD states that radioactive contaminants may enter through a SG tube leak. A discussion of these leaks is included in DCD Tier 2, Chapter 11, "Radioactive Waste Management." The DCD further states that no hydrogen buildup in the MC is anticipated. Also, in Section 10.4.1.2.1 (system operation), the DCD states that air leakage and non-condensable gases contained in the turbine exhaust steam are collected in the condenser and removed by the MCES.

However, conformance to GDC 60, as stipulated in SRP Section 10.4.1, Section II, "Acceptance Criteria," Item 1, requires that the design of the MC is acceptable if the integrated design of the system meets the requirements of GDC 60 as related to failures in the design of the system which do not result in excessive releases of radioactivity to the environment. Also, Item 1.B of the above SRP Acceptance Criteria states that the requirements of GDC 60 are met if the MC is designed to withstand the effects of any potential explosion and if instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures and dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures. The DCD did not provide adequate details to justify the requirements of

GDC 60, as it relates to failure of the MC system and potential explosion due to hydrogen buildup. Therefore, in RAI 245-2176, Question 10.4.1-2, dated March 2, 2009, the staff requested the applicant to provide additional information, with full justification, to conform to the GDC 60 criteria as described above, or provides a COL information item for the COL applicants to provide adequate details in this regard.

In its response to RAI 245-2176, Question 10.4.1-2, dated March 30, 2009, the applicant stated that under normal operating conditions, a pH controller and oxygen scavenger are injected into the SG secondary side water, as described in DCD, Section 10.4.10, “Secondary Side Chemical Injection System.” The applicant further stated that air, nitrogen, and ammonia are the main constituents in the non-condensable gases in the MC shells, and therefore hydrogen buildup is not expected in the MC. This is described in DCD Subsection 10.4.2.2.1, “General Description,” of Section 10.4.2, “Main Condenser Evacuation System (MCES).” The applicant justified that, due to this mixture, the potential for hydrogen buildup within the condenser shell does not exist.

Furthermore, the non-condensable gases are removed from the MC system by one of the two mechanical vacuum pumps, which are described in DCD Subsection 10.4.2.2.3, “System Operation,” of the MCES. If one pump fails, it gives an alarm in the MCR, and the standby pump is started. This further decreases any potential for hydrogen buildup within the condenser shells. The staff finds the applicant responses acceptable, since they conform to the GDC 60 criteria and SRP guidance; excessive radioactive releases to the environment and the buildup of hydrogen and explosive mixtures in the MC shells would be controlled. Therefore, the NRC staff’s concerns raised in RAI 245-2176, Question 10.4.1-2 are resolved. However, the explanation of how these concerns are addressed in its response was not described in the DCD, Section 10.4.1. Therefore, the staff requested the applicant in a supplemental RAI 434-3266, Question 10.4.1-1(a) to revise the FSAR Section 10.4.1 to reflect its responses to the above RAI 245-2176, Question 10.4.1-2.

The applicant provided its response to the supplemental RAI 434-3266 Question 10.4.1-1(a) in a letter dated August 26, 2009. In its response, the applicant proposed to revise the FSAR Section 10.4.1.3 of the DCD to reflect its RAI 245-2176, Question 10.4.1-2 response of March 30, 2009. In the proposed revision, the applicant referenced subsections of 10.4.2, “Main Condenser Evacuation System,” and 10.4.10, “Secondary Side Chemical Injection System,” which describe: a) the buildup and removal of noncondensable gases, and b) sources and decomposition of hydrogen in the MC during normal plant operation. The staff reviewed the proposed revision to the FSAR, Section 10.4.1.3 and finds it acceptable, because it reflected the RAI 245-2176, Question 10.4.1-2 response which resolved staff’s concern as stated earlier. Further, the staff verified that Revision 2 of the DCD incorporated the proposed revisions to Section 10.4.1.3. The staff’s evaluation of: a) the noncondensable gases and its removal, and b) chemical injection, oxygen scavenging, and thermal decomposition of hydrazine are evaluated in Sections 10.4.2 and 10.4.10 respectively, in this SE.

Regarding flooding, the DCD states that the failure of the MC and any resultant flooding will not preclude operation of any essential system since the water cannot reach safety-related equipment located in Category I structures. This statement meets the guidance of Item 3.A, Section III (SRP review procedures) of the SRP Section 10.4.3, as it relates to flood protection of the SSCs. However, the DCD did not provide details of the provisions recommended to keep the flood water from reaching the safety-related equipment. Therefore, in RAI 245-2176 Question 10.4.1-3 dated March 2, 2009, the staff requested the applicant to provide additional information in the DCD regarding flooding affects due to failure of the MC system and its components.

In its response to RAI 245-2176, Question 10.4.1-3, dated March 30, 2009, the applicant stated that in the yard area, the flood volume is directed away from the plant structures by virtue of the site grading and yard drainage system. In addition, water tight doors are installed in the doorways at the ground level, between the TB and the reactor building. This is described in FSAR Section 3.4.1.3, "Flood Protection from Internal Sources." Therefore, the applicant stated that the TB flooding does not affect the safety-related equipment in the reactor building. Also, in Section 3.4.1.3, the DCD states that there is no equipment to be protected from flooding in the TB. The staff reviewed FSAR, Section 3.4.1.3 and verified that the explanations provided by the applicant are justified and acceptable, and therefore finds the applicant's responses to RAI 245-2176, Question 10.4.1-3 acceptable. The applicant, in its response, further stated that Item 3.A in SRP Section 10.4.1.III (Review Procedures) is not applicable to Section 10.4.1 of the US-APWR DCD, which the staff also finds acceptable based on the above discussions. However, the staff requested the applicant, in supplemental RAI 434-3266, Question 10.4.1-1.b, to revise the FSAR, Section 10.4.1 to reflect these responses in the FSAR Section 10.4.1. In its response to the supplemental RAI 434-3266, Question 10.4.1-1(b), the applicant provided a mark-up of the FSAR Section 10.4.1.3 as part of its letter of August 26, 2009. The FSAR mark-up reflected its response of March 30, 2009, which the staff verified in Revision 2 of the DCD. Thus, the staff's supplemental RAI 434-4366 Questions 10.4.1-1(a) and (b) are considered closed.

Tier 2, Section 10.4.1.4, "Tests and Inspections," states that the condenser water boxes are hydrostatically tested after erection. Condenser shells are tested by completely filling them with water. Tube joints are leak tested during construction. Also, Tier 1, Table 2.7.1.3-1 of the DCD addresses the ITAAC, where it depicts the design commitment for the MC system. This adequately addresses the requirements of 10 CFR 52.47(b)(1) for the MC system that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Tier 2, Section 10.4.1.5, "Instrumentation Applications," of the DCD describes the MC protection devices, MC hotwell level, condenser pressure, and condenser temperature. Indicators and alarms are provided in the MCR. The MC hotwell is equipped with level control devices for control of automatic makeup and rejection of condensate from and to the condensate storage tank. Temperature indicators are provided for monitoring condenser performance. Condenser pressure is indicated in the MCR, and annunciates the high condenser pressure prior to reaching the turbine trip setpoint. The staff finds these protection devices acceptable, as they are consistent with SRP guidance of Item III.3, Section 10.4.3.

Based on its review and the above discussions, the staff finds that the US-APWR DCD MC system and its components meet the requirements of GDC 2 and GDC 60, as stated in the regulatory criteria and the SRP guidance.

10.4.1.5 Combined License (COL) Information Items

The staff reviewed the COL information items as listed in Tier 2 of the US-APWR DCD, Section 1.8.2, "Combined License Information", and found that there are no items relevant to the MC. The staff concluded that this is appropriate and that no COL information items are needed for the US-APWR MC system.

10.4.1.6 Conclusions

The NRC staff evaluated the MC for the US-APWR standard plant design in accordance with guidance that is referred to in the technical evaluation section of this SE. Based on its review of the information that was provided in the DCD, the staff has concluded that sufficient information has been provided by the applicant in the US-APWR DCD Tier 1, Section 2.7.1.3 and Tier 2, Section 10.4.1. In addition, the staff has compared the design information in the DC application to the relevant NRC regulations, acceptance criteria defined in NUREG-0800 - SRP Section 10.4.1, and other NRC RGs. In conclusion, the US-APWR design for the MC system is acceptable and meets the requirements of 10 CFR 52.47 (b)(1), GDC 2, GDC 60 and the guidelines of SRP Section 10.4.1 for protection against natural phenomena and control of releases of radioactive materials to the environment.

10.4.2 Main Condenser Evacuation System

10.4.2.1 Introduction

The MCES) is designed to remove air and non-condensable gases from the main condenser to establish and maintain a vacuum during startup and normal operation. The MCES performs its function of condenser evacuation with mechanical vacuum pumps. The MCES is located in the TB.

10.4.2.2 Summary of Application

FSAR, Tier 1: FSAR, Tier 1 Section 2.7.1.4.1, “Design Description,” of the US-APWR DCD, provides a brief description of the MCES. The DCD states that there are no safety-related interfaces with systems outside of the certified design. Also, DCD Tier 1, Section 2.7.1.4.2, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” describes the ITAAC requirements for the MCES. These ITAACs are depicted in Tier 1, Table 2.7.1.3-1, “Main Condenser Inspections, Tests, Analyses, and Acceptance Criteria.”

FSAR Tier 2: The US-APWR FSAR Tier 2, Section 10.4.2, “Main Condenser Evacuation System,” provides the MCES design. The US-APWR MCES is a nonsafety-related system and is located in the TB. It is a non-seismic category system and is not designed to the ASME code classifications. The NRC staff evaluation of the MCES is provided in Section 10.4.4.4, “Technical Evaluation” of this SE.

TSs: There are no TS requirements associated with the main condensers or the condenser evacuation system.

10.4.2.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800 - Standard Review Plan (SRP) Section 10.4.2, “Main Condenser Evacuation System,” which are summarized below. Review interfaces with other SRP sections can also be found in Item I, “Areas of Review,” of the SRP, Section 10.4.2, specifically with respect to detection of explosive gas mixtures and radiological monitoring.

1. GDC 2, “Design Bases for Protection Against Natural Phenomena,” in that failure of the nonsafety-related system or component due to natural phenomena such as

earthquakes, tornadoes, hurricanes, and floods should not adversely affect the safety-related SSCs.

2. GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to provisions being included in the nuclear power unit design to suitably control the release of radioactive materials in gaseous and liquid effluents during normal operation, including AOOs. GDC 60 is applicable to the design of the main condenser evacuation system because in PWRs, radioactive materials may be deposited in the main condensers if there is a primary-to-secondary SG tube leak.
3. GDC 64, "Monitoring Radioactivity Releases," as it relates to the MCES design for monitoring the releases of radioactive materials into the environment during normal operation, including AOOs.

4. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC regulations.

10.4.2.4 Technical Evaluation

The NRC staff reviewed the US-APWR MCES in accordance with SRP, Section 10.4.2. Acceptability of the MCES, described in US-APWR DCD, is based on meeting the requirements of: 1) GDC 2 for protecting the SSCs from natural phenomena, 2) GDC 60 for controlling the releases of radioactive materials to the environment, and 3) the requirements of GDC 64 for monitoring the release of radioactive materials to the environment.

The sections of US-APWR FSAR Tier 2 Section 10.4.2, "Main Condenser Evacuation System," are: MCES design bases, general description of the system and components, operation, safety evaluation, tests and inspections, and instrumentation applications of the US-APWR MCES. The classification of equipment and components is provided in Tier 2, Section 3.2, "Classification of Structures, Systems, and Components," of the DCD. FSAR Figure 10.4.2-1, "Main Condenser Evacuation System Piping and Instrumentation Diagram (P&ID)," depicts the MCES of the US-APWR. Tier 2, Table 10.4.2-1, "Main Condenser Evacuation System Design Data," provides the design features of the MCES vacuum pumps.

According to the DCD, the MCES has no safety design basis, since it performs no safety-related function. The non-condensable gases from the main condenser are exhausted to the environment in conformance with GDC 60 and GDC 64 criteria. The MCES removes the non-condensable gases from the main condenser by using three motor driven vacuum pumps.

The vacuum pumps are sized in accordance with Heat Exchange Institute (HEI), "Standards for Steam Surface Condensers." Also, the MCES piping and valves are designed in accordance with the ASME - B31.1 "Power Piping" code. The staff finds that the MCES is appropriately classified as non-safety related in accordance with the guidance in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and designed to HEI and ASME standards.

During plant startup, the three vacuum pumps are used for hogging (i.e., creating vacuum) in the main condenser, and during normal operation only one or two pumps are used to remove the non-condensable gases and to maintain the condenser vacuum. The non-condensable gases, primarily, consist of air, nitrogen, and ammonia. Dissolved oxygen will be present in the condensate in the condenser hotwell. The DCD states that only traces of oxygen will be released in the condenser, as compared to the large amounts of air that is evacuated from the condenser. Therefore, the DCD states that hydrogen buildup is not expected, and also the potential for explosive mixtures within the MCES does not exist.

Regarding conformance to GDC 2 criteria, based on its design basis and the above description, the staff finds that the US-APWR MCES system is a nonsafety-related system and is a non-seismically designed system, which is located in the TB. Failure of the MCES due to natural phenomena would have no adverse affects on safety-related SSCs, since such components are not located in the TB. Therefore, the staff finds that the US-APWR MCES meets the requirements of GDC 2.

Additionally, in Tier 2, Section 10.4.2.1.1, “General Description,” the DCD states that the non-condensable gases discharged by the MCES vacuum pumps are directed to the MCES vent. These non-condensable gases discharged from the MCES are not normally radioactive. However, the DCD states that, in the event of significant primary-to-secondary system leakage, it is possible for the non-condensable gases to become contaminated. These exhaust gases from the MCES are monitored for radioactivity prior to releasing to the environment. Also, in Section 10.4.2.5, “Instrumentation Applications,” the DCD states that a radiation detector is provided with an alarm in the MCES vent to monitor the discharge of the vacuum pumps. The DCD further states that upon detection of unacceptable levels of radiation, operating procedures are implemented.

Based on the above discussion, the staff finds that the MCES of US APWR conforms to the requirements of GDC 64, as related to the detection and monitoring of the radioactive materials in MCES vacuum pump exhaust to the atmosphere. However, in order to conform to GDC 60, as related to the control of releases of radioactive materials in the non-condensable gases from the vacuum pump exhaust, the DCD did not provide adequate details regarding its operating procedures identified earlier. Therefore, the staff requested additional information in RAI 246-2177, Question 10.4.2-1, the staff requested the applicant to provide further information regarding the key elements of these procedures. Also, the staff requested the applicant to provide additional information regarding unacceptable levels of radiation and alarm set points to preclude significant releases of radiation in the MCES vacuum pump effluents discharged to the atmosphere. Further, while it is clear that radiation monitoring is provided, based on a review of DCD Tier 2, Section 10.4.2 and the MCES P&ID (i.e., Figure 10.4.2-1), it is not clear, where the radiation monitors are located, and it is also not clear how the vacuum pump effluents are routed to the atmosphere. Therefore, the staff requested that the applicant provide further details with respect to the location of the detectors and routing of the MCES effluents.

In its response to RAI 246-2177, Question 10.4.2-1, dated March 30, 2009, the applicant stated that a statement regarding the key elements with respect to unacceptable levels of radiation and alarm setpoints to preclude significant radiation are addressed in DCD Subsection 11.5.2.4.2, “Condenser Vacuum Pump Exhaust Line Radiation Monitors.” Regarding the radiation monitoring, the applicant stated that the location of the detectors is shown in Figure 11.5-1i, “Typical Line Radiation Monitor Schematic,” and Figure 11.5-2C, “Location of Radiation Monitors at Plant.” The MCES effluents are directed to the vent of the roof of the TB via the MCES vent system, which is described in DCD Section 10.4.2.2.1. The staff reviewed the DCD Sections 10.4.2.2.1 and 11.5.2.4.2, and also reviewed the Figures 11.5-1i and 11.5-2c as described in the applicant’s response. In the response, the staff finds that the applicant adequately addressed the GDC 60 and 64 requirements, as related to the control releases and monitoring of the radioactive materials from the MCES to the environment. However, evaluation of the acceptable levels of radiation in the MCES and its monitoring is provided in Section 11.5 of the staff's SE. Further in its response, the applicant indicated no revisions to the DCD Section 10.4.1 to direct the reader to Section 11.5 for the noted discussion. Therefore, the staff issued a supplemental RAI 436-3267, Question 10.4.2-1 and requested the applicant to revise Section 10.4.2.2.1 to reflect the details provided in its response as described above.

In a letter dated August 26, 2009, the applicant provided its response to the supplemental RAI 436-3267, Question 10.4.2-1. In its response, the applicant provided a mark-up to the fifth paragraph in FSAR Section 10.4.2.2.1, where it reflected the details of its March 30, 2009, response which the staff already accepted as described above. Further, the staff verified that the mark up of the FSAR Section 10.4.2.2.1 was reflected in Revision 2 of the US-APWR DCD. Thus, the staff's concern in the supplemental RAI 436-3267, Question 10.4.2-1 is resolved,

Additionally, the DCD describes that the MCES has no direct impact on the reactor system. Should the MCES fail, condenser vacuum would gradually decrease as non-condensable gases would buildup. A decrease in turbine efficiency (due to a failing vacuum) would require an increase in reactor power, which is limited by the reactor control system. If the MCES remains inoperable and the vacuum continues to decrease, a turbine trip would be initiated, which the staff finds acceptable.

Tier 2, Section 10.4.2.5, "Instrumentation Applications," of the DCD provides the MCES with protection devices for pressure, temperature, and flow indications. Vacuum pump status and trip alarms are indicated in the MCR. More importantly, radiation monitoring of the MCES vacuum pump effluents and indicators with alarms are provided in the MCR.

The DCD states that the MCES is tested and inspected prior to plant startup. Each MCES vacuum pump is tested in accordance with the HEI standard. The DCD further identifies requirements for periodic in-service inspections and tests and also for continuous monitoring of the MCES components during plant operation to ensure satisfactory performance.

The staff reviewed the applicant's design description, system P&ID, and design criteria for the components of the MCES. The staff finds that the MCES is appropriately classified as non-safety related in accordance with the guidance in RG 1.26 and designed to HEI and ASME standards. The MCES includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment.

10.4.2.5 Combined License (COL) Information Items

The staff reviewed the COL information items as listed in Tier 2 of the US-APWR DCD, Section 1.8, Subsection 1.8.2, "Combined License Information," and finds no items that are relevant to the MCES. The staff finds it acceptable, since no COL information items are needed for the US-APWR MCES.

10.4.2.6 Conclusions

The NRC staff evaluated the MCES for the US-APWR standard plant design in accordance with guidance that is referred to in the technical evaluation section of this SE. Based on its review of the information that was provided in the DCD, the staff has concluded that sufficient information is provided by the applicant in the US-APWR DCD, Tier 1, Section 2.7.1.4 and Tier 2, Section

10.4.2. In addition, the staff has compared the design information in the DC application to the relevant NRC regulations, acceptance criteria defined in NUREG-0800 - SRP Section 10.4.1, and other NRC RGs. In conclusion, the US-APWR design for the MCES is acceptable and meets the requirements of 10 CFR 52.47 (b)(1), GDC 2, GDC 60, GDC 64 and the guidelines of SRP Section 10.4.1 for protection against natural phenomena and control of releases of radioactive materials to the environment.

10.4.3 Turbine Gland Seal System

10.4.3.1 Introduction

The turbine gland sealing system (GSS) is designed to provide a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings to prevent air leakage into and steam leakage out of these components. This includes the equipment to collect and route the system effluents to the appropriate destination. Review of the GSS is focused on the system features incorporated to monitor and control releases of radioactive materials in the effluents.

10.4.3.2 Summary of Application

FSAR Tier 1: Section 2.7.1.5, “Gland Seal System,” contains two subsections: Section 2.7.1.5.1, “Design Description,” and Section 2.7.1.5.2, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).” Section 2.7.1.5.1 describes the system purpose and functions, location and functional arrangement, key design features, seismic and ASME code classifications, system operation, alarms, logic, interlocks, class 1E electrical power sources and divisions, interface requirements, and numeric performance values. The GSS ITAACs are described in Tier 1, Table 2.7.1.5-1, “Gland Seal System Inspections, Tests, Analyses, and Acceptance Criteria.”

FSAR Tier 2: The GSS is shown in the DCD Tier 2, Figure 10.4.3-1, “Gland Seal System Piping and Instrumental Diagram.” The DCD Tier 2, Section 10.4.3, “Gland Seal System,” describes the design bases, system description, SE, tests and inspections, and instrumentation applications for the GSS of the US-APWR DCD, Revision 2.

In Section 10.4.3.1, the DCD states that the GSS performs no safety-related function and has no nuclear safety-related design basis. The system is designed to meet the following functional criteria:

- Prevent air leakage into and steam leakage out of the casings of the TG.
- Return condensed steam to the gland steam condenser and exhaust non-condensable gases into the atmosphere.
- Radioactive contamination in the non-condensable gases exhausted from the gland steam condenser is detected by a radiation monitor located in the exhaust piping line in conformance with GDC 60, “Control of Releases of Radioactive Materials to the Environment,” and GDC 64, “Monitoring Radioactivity Releases.”

ITAAC: The ITAACs for the GSS are provided in Table 2.7.1.5-1 of the Tier 1 Section 2.7.1.5.2.

TSs: There are no TS requirements associated with the GSS.

Initial Plant Test Program: The applicant indicates that the GSS testing and inspection conforms to the initial testing and operation program stipulated in US-APWR DCD Revision 2, Tier 2, Chapter 14, “Verification Program.”

10.4.3.3 Regulatory Basis

Conformance with the applicable requirements of 10 CFR 50, Appendix A, and the provisions of the following additional requirements constitutes an acceptable basis for a satisfactory GSS design.

- GDC 2, “Design Bases for Protection Against Natural Phenomena,” in that failure of the nonsafety-related system or component due to natural phenomena such as earthquakes, tornadoes, hurricanes, and floods should not adversely affect the safety-related SSCs.

- GDC 60, as it relates to the GSS design for the control of releases of radioactive materials to the environment.

- GDC 64, as it relates to the GSS design for monitoring of releases of radioactive materials to the environment during normal operation, including AOOs.

- CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC design is built and will operate in accordance with that DC design, the provisions of the Atomic Energy Act of 1954, and the NRC’s regulations.

10.4.3.4 Technical Evaluation

The staff reviewed the design of the US-APWR GSS in accordance with NUREG-0800 SRP, Section 10.4.3, “Turbine Gland Sealing System,” Revision 3. Acceptance of the GSS design is based on meeting the requirements of GDC 60 for controlling the releases of radioactive materials to the environment. Also, the acceptance of the GSS is based on meeting the requirements of GDC 64 for monitoring the releases of radioactive material to the environment.

The US-APWR DCD Section 10.4.3.1, “Design Basis,” states that the GSS has no safety-related function and therefore has no nuclear safety design basis, which the staff finds acceptable since the GSS is a nonsafety-related system. The GSS functional criteria are described in Tier 2, Section 10.4.3.1.2, “Non Safety Power Generation Design Basis,” which are discussed in the following sections.

Tier 2, Section 10.4.3.2.1, “General Description,” of the DCD provides a brief description of the GSS. The GSS is comprised of a sealing steam (steam-seal) supply header and associated piping, valves and seal steam pressure regulators, a gland steam condenser, two 100 percent motor-driven exhaust fans, and a radiation monitor in the exhaust line. Also, DCD Tier 2, Figure 10.4.3-1, “Gland Seal System Piping and Instrumental Diagram,” depicts a flow diagram for the system. The staff reviewed the P&ID and found that it conforms to the guidance provided in

Item 1, Section III, “Review Procedures,” of the SRP Section 10.4.3, in that the staff verified the source of sealing steam and the disposition of steam and non-condensable gases vented from the gland seal.

In Tier 2, Section 10.4.3.2.2, “System Operation,” the DCD describes that the GSS supplies sealing steam to the turbine shaft seals from a steam-seal header. This steam-seal header receives steam either from the auxiliary steam system or from the main steam system extracted from the main steam header. Each of these steam supply systems has a steam-seal control valve to regulate the steam flow and to maintain the steam-seal supply header pressure. During initial startup, the auxiliary boiler supplies steam to the auxiliary steam header, which in turn supplies steam to the steam-seal header. During normal operation, the main steam system supplies the sealing steam to the steam-seal header. Also, at times other than the initial startup, the sealing steam can be supplied from the auxiliary steam system as well. Additionally, the HP TGSS and each of the LP TGSS have separate steam pressure control valves in order to regulate the steam supply to the turbine glands. Excess supply of gland seal steam to the HP turbine is routed to feedwater heater No. 1 by a spillover control valve that opens automatically to bypass the excess steam.

As depicted in Figure 10.4.3-1, the mixture of air and excess steam collected from the HP and LP turbine glands is routed to a gland steam condenser. The steam-air mixture is discharged into the shell-side of the condenser. Cooling water from the condensate system flows through the tube side of the gland steam condenser as a cooling medium and condenses the steam from the steam-air mixture. Condensed steam from the gland steam condenser drains to the main condenser through a condensate recovery tank, and non-condensable gases are discharged to the atmosphere by two 100 percent exhaust fans. The gland steam condenser is maintained at a slight vacuum by the exhaust fans. Further, as depicted in Figure 10.4.3-1, the radiation monitor is located in the discharge line from the exhaust fans, and also shows that the non-condensable gases are discharged to atmosphere through total dissolved solids (TDS).

Regarding conformance to GDC 2, the staff finds that the GSS is a nonsafety-related and non-seismically designed system, which is located in the TB. Failure of the GSS or its components due to natural phenomena would have no adverse affects on safety-related SSCs, since such components are not located in the TB. Therefore, the effects of nonsafety-related over safety-related SSCs do not apply in this case, and therefore the staff finds that GDC 2 is not applicable to the GSS of US APWR.

Conformance to GDC 60 requires the GSS design to include means to control the releases of radioactive materials to the environment. Also, conformance to GDC 64 requires that the GSS design includes monitoring of the radioactive material releases to the environment during normal operation, including AOOs. In Section 10.4.3.2.2, the DCD states that the mixture of non-condensable gases discharged from the gland steam condenser is not normally radioactive. The DCD further states that, in the event of significant primary-to-secondary system leakage due to a SG tube leak, it is possible to discharge radioactively contaminated gases. According to the DCD, the GSS effluents are monitored by the radiation detector installed in the gland steam condenser exhaust fan discharge piping. Furthermore, in Section 10.4.3.5, “Instrumentation Applications,” the DCD also states that a radiation detector with an alarm is provided in the discharge piping to atmosphere to detect radiation associated with primary-to-secondary side leakage in the SGs. In Tier 2, Section 10.4.3.2.2, the DCD also states that upon detection of unacceptable levels of radiation, operating procedures are implemented. Based on the above discussion, the staff found that the GSS of the US-APWR conforms to the requirements of GDC 64, as related to the detection and monitoring of the radioactive materials in GSS effluents. However, in order to conform to GDC 60 as related to control the releases of radioactive materials in the effluents to the environment, the DCD did not provide adequate details regarding these operating procedures. Therefore, in RAI 236-2140, Question 10.4.3-1, dated February 26, 2009, the staff requested the applicant to provide further information regarding the key elements of these procedures. In addition, the staff requested the applicant to provide details on how the effluents are discharged to the environment via the TDS system identified in an earlier paragraph. Also, the staff requested the applicant to provide additional information regarding the operating procedures that will be implemented upon the detection of unacceptable levels of radiation and provision of alarms and corresponding set points to preclude significant releases of radiation in the effluents discharged to the environment.

In its response to RAI 236-2140, Question 10.4.3-1, dated March 24, 2009, the applicant stated that a description of radiation associated with primary-to-secondary side leakage is provided in DCD, Chapter 11, “Radioactive Waste Management.” The applicant committed to revising the DCD to include the following statement at the end of the fifth paragraph in Subsection 10.4.3.2.2:

> *A discussion of the radiological aspects of primary-to-secondary leakage, including anticipated releases from the system is addressed in Chapter 11.*

Also, in its response, the applicant described that the key elements (of the plant procedure that the staff requested) and the detection of the unacceptable levels of radiation and provision of the alarms and corresponding set-points to preclude significant release of radiation are described in DCD Section 11.5.2.4.3, “GSS (gland seal system) Exhaust Fan Discharge Line Radiation Monitors.” Further, the routing of the GSS effluents is shown in Figures 11.5-1j, “Typical Gland Steam Radiation Monitor Schematic,” and Figure 11.5-2g, “Location of Radiation Monitors at Plant (Power Block at Elevation 50’-2”.” The staff reviewed the DCD Section 11.5.2.4.3 and the above cited figures, and finds the applicant’s response acceptable as it addresses the GDC 60 and GDC 64 requirements and SRP guidance for the design of the GSS. However, in its response, the applicant did not revise Section 10.4.3 to direct the reader to refer to the DCD Section 11.5.2.4.3 and its associated figures. Therefore, the staff issued Supplemental RAI 437-3268, Question 10.4.3 -1 and requested the applicant to revise FSAR Section 10.4.3 to reflect the details of Section 11.5.4.2.3 and associated figures as described in its response.

In a letter dated August 26, 2009, the applicant provided its response to the Supplemental RAI 437-3268, Question 10.4.3-1. In its response, the applicant provided a mark-up to the fifth paragraph in FSAR Section 10.4.3.2.2, where it reflected the details of its March 30, 2009, response which the staff already reviewed and accepted as described above. Further, the staff verified that the FSAR Section 10.4.3.2.2 is revised to add the applicant’s response of March 30, 2009, in Revision 2 of the US-APWR DCD. Thus, the staff’s concern in the Supplemental RAI 437-3268, Question 10.4.3-1 is resolved,

A review and evaluation of the DCD Section 11.5.2.4.3, as it relates to the provisions for the process and effluent radiological monitoring of the GSS are included in Section 11.5 of this SE.

In Tier 2, Section 10.4.3.4, “Tests and Inspections,” the DCD states that the testing and the inspection will be performed based on the written procedures during the initial testing and operation program in accordance with the requirements of Chapter 14, “Verification Programs.” Also, in the Tier 1, Table 2.7.1.5-1 for the ITAACs, the DCD describes that inspections of the as-built system will be performed, and the as-built GSS will conform to the functional arrangement described in Tier 1, Section 2.7.1.5.1 of the DCD. However, as related to the ITAAC requirements and testing, the staff found no details in the Tier 1 or Tier 2 sections of the GSS-related portion of the DCD to indicate that the system will functionally operate as designed. Conformance to 10 CFR 52.47(b)(1) requires that a DC application will contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations. Also, the DCD did not: 1) identify the system or components allowing for periodic inspection and testing during the plant operation, and 2) demonstrate satisfactory functioning of the GSS equipment. Therefore, in order to satisfy the requirements of 10 CFR 52.47(b)(1) and the initial and periodic test programs, the staff requested the applicant, in RAI 236-2140, Question 10.4.3-2, dated February 26, 2009, to provide additional information to validate the functionality of the as-built GSS, with numeric performance values as applicable.

In its response to RAI 236-2140, Question 10.4.3-2, dated March 24, 2009, the applicant stated that the GSS is non-safety related, operating parameters are not critical to safety analysis, and the GSS is not identified as risk significant in probabilistic risk assessment (PRA) evaluations from DCD Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation." The applicant considers that the appropriate ITAAC and level of Tier 1 detail for GSS is provided by the general description of the system in Section 2.7.1.5 and identification of GSS exhaust line radiation monitors in Section 11.5.2.4.3. The applicant further stated that the GSS has no nuclear safety design basis, and therefore numerical parameters for monitoring GSS performance or details of tests and inspection are not critical to nuclear safety. The staff agrees that the numeric performance values and key parameters need not be specified unless there is a specific reason to include them. Based on the above discussion, the staff finds the applicant's response to RAI 236-2140, Question 10.4.3-2 acceptable. Therefore, the staff's concern described in RAI 236-2140, Question 10.4.3-2 is resolved.

Tier 2, Section 10.4.3.5, "Instrumentation Applications," describes the pressure controls, level controls, and alarms that are provided for proper operation of the system. A pressure controller signals a steam-seal control valve to control steam-seal supply header pressure. Additionally, pneumatic control valves provide "appropriate" pressure to the HP and LP turbine glands with excess steam controlled by the gland spillover control valve. The gland seal condenser is monitored for shell side pressure and also its internal liquid level. Alarms are provided for monitoring the system pressure and also to detect radiation associated with primary-to-secondary leakage in the SGs.

10.4.3.5 Combined License Information Items

There are no COL information items for the GSS.

10.4.3.6 Conclusions

The NRC staff has concluded that sufficient information has been provided by the applicant in the US-APWR DCD Tier 1, Section 2.7.1.5 and Tier 2, Section 10.4.3, "Gland Seal System". In addition, the staff has compared the design information in the DC application to the relevant NRC regulations, acceptance criteria defined in NUREG-0800, Section 10.4.3, and other NRC RGs. In conclusion, the US-APWR design for the GSS is acceptable and meets the requirements of 10 CFR 52.47 (b)(1), GDC 60 and GDC 64 and the guidelines of SRP Section 10.4.3 for controlling and monitoring of releases of radioactive materials to the environment.

10.4.4 Turbine Bypass System

10.4.4.1 Introduction

The TBS is part of the main steam system and provides capability to transport main steam from the SGs directly to the main condenser, bypassing the main turbine. This process is accomplished in a controlled manner and minimizes transient effects on the RCS during plant startup, hot standby, cooldown, and generator step-load reductions, turbine trips, or following a reactor trip.

10.4.4.2 Summary of Application

FSAR Tier 1: In FSAR Tier 1, Section 2.7.1.6.1, “Design Description,” of the US APWR DCD), the applicant provided a brief description of the TBS. The TBS is a nonsafety-related system and is located in the TB. It is a non-seismic category system and is not designed to ASME code classifications. Further, the DCD states that there are no safety-related interfaces with systems outside of the certified design. Also, in Tier 1 Section 2.7.1.6.2, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” the DCD describes the ITAAC requirements for the TBS.

FSAR Tier 2: In FSAR Tier 2 Section 10.4.4, “Turbine Bypass System,” of the US-APWR DCD, the applicant provided the TBS design bases, system and component description, system operation, SE, inspection and tests, instrumentation, and design parameters. The details of the TBS design and staff evaluation of the system are described in the Section 10.4.4.4, “Technical Evaluation,” of this SE.

ITAAC: The ITAACs for the TBS are depicted in Tier 1 Table 2.7.1.6-1, “Turbine Bypass System Inspections, Tests, Analyses, and Acceptance Criteria,” of the US-APWR DCD.

TSs: There is no TS requirement associated with the TBS.

10.4.4.3 Regulatory Basis

Conformance with the applicable requirements of 10 CFR 50, Appendix A, and the provisions of the following additional requirements constitute an acceptable basis for a satisfactory design of the TBS.

1 GDC 2, “Design Bases for Protection Against Natural Phenomena,” in that failure of the nonsafety-related system or component due to natural phenomena such as earthquakes, tornadoes, hurricanes, and floods should not adversely affect the safety-related SSCs.

2 GDC 4, “Environmental and Dynamic Effects Design Basis,” in that failure of the TBS due to a pipe break or malfunction of the TBS should not adversely affect systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).

3 GDC 34, “Residual Heat Removal,” as related to the ability to use the system for shutting down the plant during normal operations. The operation of the TBS eliminates the need to rely solely on safety systems.

4 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

10.4.4.4 Technical Evaluation

The design of the TBS is described in the FSAR Tier 1, Section 2.7.1.6 and the FSAR Tier 2, Section 10.4.4 of the US-APWR DCD, Revision 2. The staff reviewed the TBS, as described in the US-APWR DCD, Revision 2, in accordance with the NUREG-0800 SRP, Section 10.4.4, Revision 3, and the acceptability of the system is based on meeting the requirements of the following GDC criteria:

- GDC 2, as it relates to the failure of a nonsafety-related system or equipment, due to natural phenomena, does not adversely affect safety-related SSCs,

- GDC 4, as it relates to the system being designed such that a failure of the system due to a pipe break or system malfunction does not adversely affect safety-related systems or components, and

- GDC 34, as it relates to the ability to use the TBS for shutting down the plant during normal operations by removing residual heat without using the TG.

In Tier 2, Section 10.4.4.1, "Design Bases," the DCD provides the safety and power design bases for the TBS. The system has no safety design basis, since it performs no safety-related function. In Tier 2, Section 10.4.4.1.2, "Non-safety Power Generation Design Bases," the DCD describes that the TBS of the US-APWR is designed to bypass 67 percent of the main steam to the main condenser at full power operation. Also, the DCD states that the system is designed to sustain a 100 percent (electrical) load rejection, without generating a reactor trip, and without actuating a MSRV, MSSV, or pressurizer safety valve. The TBS is also designed to follow rapid turbine load reduction greater than 10 percent, but less than 100 percent with a reactor trip, which is discussed later in this SE. Further, the US-APWR TBS is designed to bypass steam to the main condenser during plant startup and also to permit a normal cooldown of the RCS from a hot shutdown to the point at which the RHR system can be placed in service. The staff finds these design features minimize transient effects on the RCS during plant startup, hot shutdown and cooldown, step load reductions in generator load, and following a reactor trip.

In Tier 2, Section 10.4.4.2, "System Description," the DCD provides a general description and also the component description of the APWR TBS. The system consists of a TBV header tapped off of the main steam equalization piping manifold upstream of the main turbine stop valves. Two individual sub-headers per condenser shell are extracted from this TBV header, and lines with the TBVs are connected to these sub-headers. A total of 15 TBVs are provided, and three sets of five valves each discharge to the three shells of the main condenser. FSAR Tier 2, Figures 10.3-2, "Main Steam Supply System Piping and Instrumentation Diagram (2/4)," and 10.3-3, "Main Steam Supply System Piping and Instrumentation Diagrams" of the DCD depict the TBS, and is shown as part of the MSSS.

Further, in FSAR Tier 2, Section 10.4.4.3, "System Operation," the DCD describes that the TBS has two modes of operation: 1)T_{avg} control, and 2) pressure control modes. The T_{avg} mode is used for at-power transients for generator load rejection and also for turbine trips. The system has independent controllers for load rejection and turbine trips. In the T_{avg} mode, the DCD states that the TBS operates to sustain a 100 percent load rejection, without generating a reactor trip or actuating a MSRV, MSSV, or pressurizer safety valve. Following a turbine trip, the turbine trip controller becomes active and activates the TBS. Further, the pressure control mode is used at no-load operations, and is used to remove decay heat during plant startup and cooldown.

The staff reviewed the information presented in the DCD as described above and evaluated the TBS against the GDC 2, GDC 4, and GDC 34 criteria as follows:

With respect to conformance to GDC 2, the staff finds that the TBS is a nonsafety-related and non-seismically designed system, which is located in the TB. Failure of the TBS or its components due to natural phenomena will have no adverse affects on safety-related SSCs since such components are not located in the TB. Therefore, the staff finds that the US-APWR TBS meets GDC 2 since its failure will not impact safety-related SSCs.

Conformance to GDC 4 requires that failure of the TBS due to a pipe break or malfunction of the system should not adversely affect essential systems or components that are necessary for safe shutdown or accident prevention or mitigation. In FSAR Tier 2, Section 10.4.4.4, the DCD states that the TBS serves no safety function and there is no safety-related equipment or components that exist in the vicinity of the TBS components. The DCD also states that all the high-energy lines of the TBS are located in the TB and failure of TBS high-energy lines will not disable the turbine speed control system. The staff verified that the TBS is located in the non-seismic TB, and that there are no safety-related SSCs in the TB or nearby the TBS. Further, the turbine speed control system is designed such that its failure will only trip the turbine and will have no adverse affect on those SSCs that allow a safe shutdown despite a failure in the TBS. Based on the above discussion, the staff finds that the TBS of the US-APWR meets the GDC 4 criteria, as it relates to the adverse affects of a pipe break or malfunction on those components of the system necessary for safe shutdown or accident prevention or mitigation since such components do not exist in the TB.

Conformance to GDC 34 requires the TBS to be designed to shutdown the plant during normal plant operations by removing residual heat without using the TG. In FSAR Tier 2, Section 10.4.4.1.2, the US-APWR DCD states that the TBS is designed to bypass steam to the main condenser during a plant shutdown to facilitate a manually controlled cooldown of the RCS to the point where the RHR system can be placed in service for further cooldown. Also, the DCD states that the TBS has the capacity to bypass 67 percent of the main steam flow to the main condenser at full power, and is designed to sustain a 100 percent load rejection, without generating a reactor trip, and without requiring actuation of the MSRVs, MSSVs, or pressurizer safety valve. The DCD further states that the TBS is designed to follow a rapid turbine load reduction greater than 10 percent, but less than 100 percent, with a reactor trip. The staff finds that the US-APWR TBS conforms to the GDC 34 requirements because it supports the RHR function for shutting down the plant during normal plant operation.

Further, Item 2, Section III, “Review Procedures” of SRP Section 10.4.4, “Turbine Bypass System,” recommends verification of the relation between the TBS and the MSRV capacity in terms of percentage of main steam flow, the maximum reactor power step change the system is designed to accommodate without a reactor or turbine trip, and the maximum electric load step change the reactor is designed to accommodate without reactor control rod motion or steam bypassing. However, Revision 1 of the DCD did not address this feature as recommended in the SRP guidance. Therefore, the staff requested additional information in RAI 159-1955, Question 10.4.4-1, dated January 21, 2009, regarding the TBS capacity for the maximum step change requirements in terms of percentage of the main steam conforming to the above SRP guidance as related to the GDC 34 requirement.

In a letter dated February 20, 2009, the applicant provided its response to RAI 159-1955, Question 10.4.4-1 and described that, with 15 TBVs, the TBS has a capacity of 68 percent of the rated power main steam flow. This is reflected in the DCD Section 10.4.4.1.2. The applicant also stated that the sum of the MSRV capacity is 10 percent of the rated power main steam flow. The applicant further stated that the reactor power is controlled following the electric load, and referred to the DCD, Section 10.4.4.3, where it is described that the TBS is designed to accommodate the maximum 100 percent step change of electric load without a reactor or turbine trip and without the actuation of the MSRVs. Additionally, in Section 10.4.4.3, it is described that the reactor is designed to be able to follow the maximum 10 percent step change of electric load with control rod motion and without using the TBS.

Based on its response, the staff finds that the applicant adequately addressed the SRP guidance in meeting the GDC 34 requirement, as it relates to the maximum reactor power step change the system is designed to accommodate without a reactor or turbine trip, and the maximum electric load step change the reactor is designed to accommodate without reactor control rod motion or steam bypassing. However, the relationship between the TBS capacity and the MSRV capacity in terms of percentage of rated maximum main steam flow was not clear to the staff. Therefore, the staff requested the applicant to provide clarification and/or additional information to meet the requirements of GDC 34, with respect to the relationship between the capacities of the MSRVs and the TBS. The staff further requested the applicant in a Supplemental RAI 430-3269, Question 10.4.4-1 to provide its response with proper justification and also to revise the FSAR to reflect its response(s).

In a letter dated August 28, 2009, the applicant provided its response to Supplemental RAI 430-3269, Question 10.4.4-1. In its response, the applicant stated that the three TBVs with 13.6 percent of rated main steam flow of 20,000 lb/h at the valve inlet pressure 777 psig perform adequate decay heat removal to keep the cooldown rate of RCS at 50 ^{O}F per hour during normal plant shutdown and thereby reduce the demands on the systems important to safety in meeting GDC 34. The staff finds the applicant’s response acceptable, since it provided sufficient details to ensure compliance with GDC 34; specifically it meets the underlying rationale of the GDC 34 requirement that using the TBS during normal plant shutdown reduces demands on systems important to safety. Also in its response, the applicant proposed a markup of the DCD to reflect this additional information. Further, the staff verified that Revision 2 of the DCD reflected the applicant’s response by adding this additional information to Tier 2, Section 10.4.4.4. Therefore, RAI 430-3269, Question 10.4.4-1 is resolved and closed.

Section 10.4.4.6,”Instrumentation Application,” of the DCD describes that the TBS controls are provided in the MCR for its operating mode selections. Also, the valve position indicators and the pressure indicators are located in the MCR. The DCD states that details of the instrumentation are described in Section 7.7, “Control Systems Not Required for Safety,” which is evaluated by the staff in Chapter 7, “Instrumentation and Controls,” of this SE.

ITAAC Considerations

The ITAAC requirement for the TBS of US-APWR is identified in Tier 1, Table 2.7.1.6-1 as described earlier in this SE. The staff reviewed this ITAAC requirement and finds it acceptable since it will ensure that the design features of the as-built system conform to the design commitments as described in the approved design in the DCD. This ITAAC also conforms to the 10 CFR 52.47(b)(1) requirement identified earlier in the "Regulatory Basis" section in this SE.

Initial Plant Test Program

In Section 10.4.4.5, "Inspection and Tests," the DCD states that before the system is placed in service, all the TBVs are tested for operability and the pipelines are hydrostatically tested to verify leak tightness. Also, all the piping and valves are accessible for inspection. The staff finds this acceptable.

10.4.4.5 Combined License Information Items

No COL information items are identified for the TBS in the DCD.

10.4.4.6 Conclusions

The staff has compared the information within the US-APWR application to the relevant NRC regulations and acceptance criteria defined in NUREG-0800, Section 10.4.4.

The NRC staff basis for acceptance of the TBS is conformance of the TBS to the Commission's regulations as set forth in 10 CFR 52.47 (b)(1), GDC 2, GDC 4 and GDC 34 criteria. The staff finds that the TBS of the US-APWR meets the requirements of GDC 4 with respect to the system being designed such that a TBS failure due to a pipe break or malfunction will have no adverse affect on the essential systems or components that are necessary for safe shutdown or accident prevention or mitigation. The system has also met the requirements of GDC 34 with respect to the ability of the TBS for shutting down the plant during normal operations by removing residual heat without using the TG. The staff concludes that the applicant has provided sufficient information for satisfying GDC 4.

10.4.5 Circulating Water System

10.4.5.1 Introduction

The CWS supplies cooling water to remove heat from the main condensers, under varying conditions of power plant operation and site environmental conditions. It does not have a safety-related function.

10.4.5.2 Summary of Application

Tier 1: The Tier 1 information associated with this section is found in Tier 1, Section 2.7.1.7, "Circulating Water System (CWS)," of the US-APWR DCD.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.4.5, "Circulating Water System," of the US-APWR DCD. It is a conceptual design based upon a cooling tower (CTW) approach. The COL applicant is to determine the site-specific final system configuration and system design parameters for the CWS including makeup water and blowdown. The staff's review discussed herein is based on the conceptual design presented in Section 10.4.5 of the US-APWR DCD, which is summarized here in part, as follows:

- The CWS draws water from the CWS cooling tower basin, and returns water to the CWS CTWs after passing through the main condenser. The CWS P&ID is shown in Figure 10.4.5-1, "Circulating Water System Piping and Instrumentation Diagram." The CWS has the following design functions:

  - The CWS supplies cooling water at a specified flow rate to condense the steam in the condenser.

  - The CWS is automatically isolated in the event of gross leakage in the TB condenser area to prevent flooding on the TB.

  - The CWS is designed such that a failure in a CWS component (piping, CTW, expansion joint, pump, etc.) does not have a detrimental effect on any safety-related equipment or component.

  - The CWS is composed of eight, 12.5 percent capacity circulating water pumps, CTWs, CTW basins, makeup water pumps, blowdown pumps, and associated piping, valves, strainers, and instrumentation.

  - The circulating water pumps are located in the CTW basins, and take suction from the CTW basin and pump water through the main condenser under varying conditions of power plant loading and design weather conditions. Conceptual design parameters for the major components are described in Table 10.4.5-1.

  - Makeup water is provided by the raw water system to compensate for the CTW evaporation, drift, and blowdown.

  - The CTW water chemistry is controlled by the CWS/raw water system chemical treatment system.

ITAAC: The ITAAC associated with Tier 2, Section 10.4.1 is given in Tier 1, Section 2.7.1.7.2 of the US-APWR DCD.

TSs: There is no TSs associated with this area of review.

10.4.5.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 10.4.5, "Circulating Water System," and are summarized below.

1. General Design Criteria (GDC) 2, "Design Bases for Protection Against Natural Phenomena," in that failure of the nonsafety-related system or component due to natural phenomena such as earthquakes, tornadoes, hurricanes, and floods should not adversely affect the safety-related SSCs.

2. GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to design provisions provided to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS.

3 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

10.4.5.4 Technical Evaluation

The staff reviewed the design of the US-APWR CWS in accordance with SRP Section 10.4.5. Acceptance of the CWS design is based on meeting the requirements of GDC 2 and GDC 4, as they relate to provisions in the US-APWR design to accommodate the effects of discharge water that may result from a failure of a component or piping in the CWS.

The US-APWR DCD Section 10.4.5.1, "Design Basis," states that the CWS has no safety-related function and therefore has no nuclear safety design basis, which the staff finds acceptable since the CWS is a nonsafety-related system. The CWS functional criteria are described in Tier 2, Section 10.4.5.1.2, "Non Safety Power Generation Design Basis," which are discussed in the following sections.

The US-APWR CWS as depicted in DCD Tier 2, Figure 10.4.5-1, "Circulating Water System Piping and Instrumentation Diagram," consists of circulating water pumps, CTWs, CTW basins, makeup water pumps, blowdown pumps, and associated piping, valves, and instrumentation. System design values are given in DCD Table 10.4.5-1, "Design Parameters for Major Components of Circulating Water System." The values in this table that are dependent on site-specific conditions will be replaced with site-specific values by COL applicants.

The CWS is composed of eight, 12.5 percent capacity circulating water pumps located in the CTW basins from which they take suction to pump water through the main condenser in the TB. Each CTW basin contains four circulating water pumps arranged in parallel. Two mechanical draft CTW assemblies provide 100 percent cooling for normal power operation. Makeup water to the CTWs is provided by the raw water system.

Regarding conformance to GDC 2 criteria, based on the above description, the staff finds that the US-APWR CWS is a nonsafety-related system and a non-seismically designed system. Failure of the CWS or its components due to natural phenomena will have no adverse affects on safety-related SSCs, since such components are not located in the TB. The staff's evaluation of the potential adverse effects of failure of the non-safety related raw water system on the safety related ultimate heat sink basin is discussed in Section 9.2.5 of this report. Therefore, the staff finds that the US-APWR CWS meets the requirements of GDC 2.

The requirements of GDC 4 are met when the CWS design includes provisions to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS. Specifically, means should be provided to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS.

The DCD states in Section 10.4.5.2.1, "General Description," that the CWS is automatically isolated if gross leakage into the TB condenser area occurs. The applicant also states that the CWS is designed in such a way that a CWS component failure will not have any detrimental effect on any safety-related equipment.

In DCD Section 10.4.5.3.4.1, “Circulating Water Piping/Expansion Joint Failures,” it is stated that large CWS leaks are indicated and alarmed in the control room in the event of a loss of vacuum in the condenser shell. The DCD also states that water from a system rupture will discharge from the TB through a relief panel in the TB wall before the water level rises to a point that can cause damage and that site grading will carry the water away from safety-related buildings.

A CWS piping failure in the yard area or a failure of the CTW basin does not impact any safety-related component due to the CTWs location being a sufficient distance from any equipment or structure important to safety.

Based on the above information, the staff concludes that the design of the US-APWR CWS meets the requirements of GDC 4 as it relates to design features to accommodate the effects of discharge water that may result from a failure of a component or piping in the CWS.

Technical Specifications

The staff reviewed Chapter 16, “Technical Specifications,” of the US-APWR DCD and found no technical specifications that are relevant to the CWS, and that the staff agrees no TSs are needed since it is a nonsafety-related system that does not affect any safety-related system or equipment.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

10 CFR 52.47(b)(1) requires a DC applicant to provide proposed ITAAC necessary to ensure that a plant incorporating the certified design is built and will operate in accordance with the DC, the Atomic Energy Act of 1954 and NRC regulations. Tier 1, DCD Section 2.7.1.7, “Circulating Water System (CWS),” describes the purpose of the CWS and its functions. Tier 1, DCD Table 2.7.1.7-1, “Circulating Water System Inspections, Tests, Analyses, and Acceptance Criteria,” states the design commitment; the ITAAC for the CWS. The staff reviewed this ITAAC and concluded the requirements of 10 CFR 52.47 (b) (1) for the US-APWR CWS are satisfied.

Initial Test Program

The staff reviewed Chapter 14, “Verification Programs,” of Tier 2 of the US-APWR DCD to ensure the applicant conformed to Initial Plant Test requirements. Chapter 14, of the US-APWR DCD lists one item for the CWS. Preoperational Test 14.2.12.1.33, “Circulating Water System Preoperational Test,” describes the objective, prerequisites, test method, and acceptance criterion for the CWS preoperational test. No additional preoperational testing is required. This test was reviewed by the staff and found to be an acceptable means to verify the system will perform as stated in DCD Section 10.4.5.

10.4.5.5 Combined License Information Items

The following COL information item is identified in DCD Tier 2, Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” as COL Information Item 10.4(1) for the COL applicants that use the CWS design as described in the US-APWR DCD:

- The Combined License Applicant is to determine the site specific final system configuration and system design parameters for the CWS including makeup water and blowdown.

The NRC staff finds the above listing to be complete. No further additional COL information items are identified that need to be included in DCD Tier 2, Table 1.8-2 for the CWS.

10.4.5.6 Conclusions

The staff evaluated the CWS for the US-APWR standard plant design in accordance with guidance that is referred to in the technical evaluation section of this SE. Based upon a review of the information that is provided and as discussed above in the technical evaluation section, the staff found that the applicant has met the requirements of GDC 2, GDC 4 and 10 CFR 52.47(b)(1) for Section 10.4.5, “Circulating Water System,” of the US-APWR DCD.

10.4.6 Condensate Polishing System (related to RG 1.206, Section C.III.1)

10.4.6.1 Introduction

The CPS is designed to remove dissolved ionic solids and impurities from the condensate. The CPS provides condensate cleanup capability and maintains condensate quality through demineralization. It does not perform a safety-related function. Also discussed in this section is secondary plant water chemistry as described in DCD Section 10.3.5, “Water Chemistry.”

10.4.6.2 Summary of Application

Tier 1: The Tier 1 information associated with this section is found in Tier 1, Section 2.7.1.8, “Condensate Polishing System (CPS),” of the US-APWR DCD.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.4.6, “Condensate Polishing System (CPS),” of the US-APWR DCD, summarized here in part, as follows:

- The CPS is designed with prefilters to remove corrosion products and with deep-bed, mixed-resin demineralizers to remove ionic impurities from the condensate during plant startup, hot standby, shutdown operations, and power operation. A condensate bypass valve is located in the condensate pump discharge header to bypass the condensate polishing vessels. The CPS consists of the following components: condensate prefilters and polishing vessels, resin traps, spent-resin holding vessels, a portable resin addition hopper and eductor, and instrumentation. The CPS is shown in DCD Figure 10.4.6-1, “Design Bases.”

- DCD Section 10.3.5 describes the secondary plant water chemistry, but its evaluation is included under Section 10.4.6 of this SE because the CPS is one of the principal means of effecting secondary water chemistry control.

ITAAC: The ITAAC associated with Tier 2, Section 10.4.6 are given in Tier 1 Section 2.7.1.8.2, “Inspections, Tests, Analyses, and Acceptance Criteria,” of the US-APWR DCD.

TSs: There is no TSs associated with this area of review.

10.4.6.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 10.4.6, “Condensate Cleanup System,” of

NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 10.4.6.I of NUREG-0800.

1. GDC 14, "Reactor Coolant Pressure Boundary," in Appendix A to 10 CFR Part 50, requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to ensure an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. GDC 14 applies to Section 10.4.6 of NUREG-0800, because the condensate cleanup system maintains water quality to avoid corrosion-induced failure of the reactor pressure boundary, specifically the SG tubing.

2. 10 CFR 52.47(b)(1), which requires that a DC application include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

10.4.6.4 Technical Evaluation

Secondary water chemistry involves many constituents that must be controlled to ensure materials integrity and safe operation. The principal industry standard is the "EPRI *PWR Secondary Water Chemistry Guidelines* (the EPRI Guidelines)," which is discussed in DCD Subsection 10.4.6.3.1, "Normal Operation." Chemistry control in the US-APWR is summarized in DCD Subsection 10.4.6.3.2, "Condenser Tube Leak," where some inconsistencies between the DCD and the EPRI Guidelines are noted. In response to inquiries by the staff, the applicant has sought to justify several of these differences. Because the EPRI Guidelines represent the industry standard, and are accepted by the NRC as such, any deviation should be documented and justified. Hence, these issues are discussed in detail in DCD Subsections 10.4.6.3.3, "Safety Evaluation," and 10.4.6.3.4, "Tests and Inspections."

10.4.6.4.1 Industry Standard

SRP acceptance criteria for Section 10.4.6 (II.2) refer to the acceptance criteria for SRP Section 5.4.2.1, and these refer to the latest version of the EPRI *PWR Secondary Water Chemistry Guidelines*.

Although the staff does not formally review or issue a SE of the various EPRI water chemistry guidelines (including the *PWR Secondary Water Chemistry Guidelines*), the guidelines are recognized as representing the industries best practices in water chemistry control. Extensive experience in operating reactors has demonstrated that following the EPRI Guidelines minimizes the occurrence of corrosion-related failures. Further, the EPRI Guidelines are periodically revised to reflect evolving knowledge with respect to best practices in chemistry control. Therefore, the staff accepts the use of the EPRI *PWR Secondary Water Chemistry Guidelines* as a basis for a recommended water chemistry program for a standard PWR design.

As is the case with primary system water, the EPRI Guidelines define certain measurements as control parameters and others as diagnostic parameters. The control parameters have a demonstrated link to SG degradation, and require close control of limits to ensure system integrity. Diagnostic parameters are important for diagnosis of problems, but are usually less stringent than control parameters.

The EPRI Guidelines also define three action levels for limit violations during normal operation, with increasing severity. If the Action-Level-1 limit is violated, power operation may continue while corrective action is taken. If the violation lasts longer than one week, this becomes an Action-Level-2 violation. An Action-Level-2 violation requires immediate power reduction and correction within 100 hours; otherwise an Action-Level-3 violation occurs. Action Level 3 requires immediate shutdown until the violation is corrected. The staff notes that while the terms “violation” and “requires” and other language that implies operational mandates are commonly used in this context, they are not meant, nor should be construed, as regulatory requirements, including TSs. Rather, they are technical provisions that, if adhered to, provide reasonable assurance that the regulatory requirement of maintaining the integrity of the reactor coolant pressure boundary (SG tubes) to the extent practicable through secondary water chemistry control will be met.

10.4.6.4.2 US-APWR Chemistry Control

Secondary water chemistry is focused on preventing corrosion in SGs, condensers, piping, and other equipment. Principal parameters that must be controlled are impurity ion concentrations, including sodium (Na^+), chloride (Cl^-), and sulfate (SO_4^{-2}) ions; pH; and dissolved oxygen. In the US-APWR, hydrazine is added to scavenge oxygen, and a morpholine/dimethylamine mixture is used to control pH.

Purification of secondary water is achieved through the CPS and the SG blowdown system (SGBS), which is capable of processing about one-third of the water from the condensers. The system runs at maximum capacity during startup, but is usually bypassed during normal operation. If water quality degrades, then flow through the CPS resumes until normal water quality is restored. Such intermittent operation is reasonable since, in the SGs, the less volatile impurities (metals and ions) stay predominantly in the liquid phase and are not transported with the steam to the condensers. The CPS is primarily needed to remove contaminants entering through the condensers themselves. DCD Section 10.3.5.2, “Containment Ingress,” indicates the condensers are a significant source of contaminant ingress. In addition, if leakage from the primary system occurs, then radioactive nuclides could also reach the condenser, and would be removed by the CPS.

The CPS purifies secondary water by passing it through mixed-resin (cation and anion) demineralizers. Each of the three demineralizers has its own resin trap, and all three are served by a single spent-resin holding vessel and a single resin mixing vessel, where fresh resin is prepared and stored. As described in DCD Table 10.4.6-1, “Condensate Polishing System Design Parameters,” all of these vessels are constructed of carbon steel, and all but the resin traps have a rubber lining. The rubber linings have been proven to provide adequate protection from corrosion of the carbon steel, and to prevent impurities from entering the secondary water. Design pressures for each vessel are shown in DCD Table 10.4.6-1; however, no operating details are included. Therefore, in **RAI 235-2134, Question 10.4.6-04**, the staff requested that the applicant provide a description of the equipment needed to ensure that the condensate, holding, and mixing vessels, as well as the resin traps, not to exceed the various design pressures. In its response to **RAI 235-2134, Question 10.4.6-04**, dated March 25, 2009, the applicant stated that it verified that all vessels conform to the applicable ASME design requirements. In addition, the applicant identified two possibilities for overpressure, the resin holding and mixing vessel and the spent-resin holding vessel. The applicant reported that it modified the design by including pressure relief valves in these units. The applicant also stated that DCD Figure 10.4.6-1 would be modified accordingly. Subsequently, the staff confirmed that this change was made in Revision 2 of the DCD. The staff finds that **RAI 235-2134, Question 10.04.06-04**, was adequately addressed by the applicant and is therefore resolved and closed.

As stated in DCD Section 10.4.10.2.2, "Component Description," morpholine and dimethylamine are used as pH controllers, and hydrazine is added to control dissolved oxygen. These additives are quite volatile and would normally travel with the steam through the turbines and to the condensers. Hence, depletion of these chemicals in the CFS will be detected in the main condensers (c.f. DCD 10.4.10.1). Thus, the injection of new chemicals occurs on the discharge side of the CPS. Additional injection points are the deaerator and SG feedwater. Concentrations are monitored using continuous analyzers (supplemented by grab samples), as described in DCD Section 9.3.2, "Process and Post-Accident Sampling Systems," and additive injections are determined either automatically or manually from the analytical data. Chemical mixing tanks, storage tanks, piping and valves are constructed of stainless steel to minimize corrosion and admission of impurities into the secondary water.

The EPRI Guidelines specify limits and sampling frequencies for concentrations of various aqueous species in order to assure materials integrity and safe operation. The DCD initially specified limits and frequencies, which were in some cases different than the EPRI values. In addition, the DCD definitions of Action Levels differed considerably from those in the EPRI Guidelines. A number of RAI's were issued, attempting to clarify the applicant's intentions, and to either establish consistency with EPRI Guidelines, or to thoroughly justify the differences. A summary description of the RAI's and answers can be found in the next section (10.4.6.4.3). Finally, in response to RAI 630-5044, Question 10.04.06-16 (Reference 10), dated October 6, 2010, the applicant has declared its intentions to eliminate all specification of chemistry control parameters and to place complete responsibility on the COL holder to provide specifications that are consistent with the EPRI Guidelines or are otherwise defensible. They issued revisions to the DCD that effectively accomplish such a change. These revisions eliminate several tables that were the subject of much discussion in previous RAI's, rendering such discussions (i.e., Section 10.4.6.4.3) obsolete. The applicant emphasized that the EPRI Guidelines are the principal water chemistry standard to be met; the staff has no objection to this plan as outlined in the applicant's revisions.

The EPRI Guidelines also describe diagnostic parameters, whose use is important in monitoring water purity. However, no limits are specified, since these parameters usually involve plant-specific conditions. The DCD mentions all the parameters noted in the EPRI Guidelines for feedwater, but does not include several diagnostic parameters for blowdown samples. These are discussed in more detail below.

1. **Action Levels**. As noted in SE Subsection 10.4.6.4.1 above, the EPRI Guidelines define concentration limits during power operation for three action levels. In its response to **RAI 235-2134, Question 10.4.6-2**, dated March 25, 2009, the applicant also mentioned Action Levels 1-3, but omitted the mandatory consequences and time limits for correction. In its response to **RAI 235-2134, Question 10.4.6-3**, also dated March 25, 2009, the applicant defended the lack of limits for Action Levels 2 and 3, since these values would require input from the plant operator. However, the applicant did not establish COL information items for a COL applicant to address this item. In its response to **RAI 441-3461, Question 10.4.6-9**, dated September 16, 2009, the applicant again mentioned Action Levels 1-3. However, the applicant's definitions of the action levels did not match those of the EPRI Guidelines, and continued to omit consequences for prolonged limit violations. In response to RAI 543-4368, Question 10.4.6-11 (Reference 9), dated April 26, 2010, the applicant has declared its intent to issue a COL

information item to ensure that the COL holder establishes secondary water chemistry guidelines based on the EPRI Guidelines, or provides adequate justification for deviating from them. This intent was confirmed by the response to RAI 630-5044, Question 10.4.6-16 (Reference 10).

2. **Concentrations of Impurities Na, Cl, SO_4 During Power Operation**. In its response to **RAI 235-2134, Question 10.4.6-1**, dated March 25, 2009, the applicant provided its basis for specifying concentration limits which are different from EPRI Guidelines. In its response to **RAI 383-3002, Question 10.4.6-6**, dated July 6, 2009, the applicant further explained this difference. This issue is addressed in detail in SE Section 10.4.6.4.3 below.

3. **Concentrations of Impurities Na, Cl, andSO_4 During Heatup**. Limits for control parameters sodium (Na), chloride (Cl), and sulfate (SO_4) in blowdown water are given in the EPRI Guidelines (Table 5-3, "EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines"), and the corresponding specifications in DCD Section 10.3.5, Table 10.3.5-3, are either consistent with or more conservative than the EPRI limits for escalation above 5 percent power. However, for power increases above 30 percent, the DCD is not consistent with the EPRI Guidelines, and this discrepancy is evaluated in SE Section 10.4.6.4.3 below.

4. **Sampling Frequency**. In the DCD Table 9.03.02-04, "Secondary Side Sampling System," the applicant supplied recommended sampling schedules for all control parameters, and several diagnostic parameters in feedwater, blowdown and condensate. These sampling frequencies meet or exceed the recommendations in the EPRI Guidelines in all cases in which sampling is actually performed (see SE Section 10.4.6.4.4, below, for the exceptions).

5. **Table Heading Change**. In its July 6, 2009, response to **RAI 383-3002, Question 10.4.6-6**, the applicant committed to revise DCD Table 10.3.5-3, so that the subheading ("Under 30%") wording is consistent with EPRI Guidelines Table 5-3 ("> 30%").

6. **Action-Level-1 Limits (Power Operation)**. During the power operation, the DCD is consistent with EPRI Action-Level-1 limits for condensate dissolved O_2 and feedwater dissolved oxygen (O_2), iron (Fe), and copper (Cu). The limits for hydrazine are approximately the same, although calculated differently. However, as noted in Item 1, the definitions of action levels are not consistent with the EPRI Guidelines. Also, the applicant's September 16, 2009, response to **RAI 441-3461, Question 10.4.6-9**, indicated that the Standard Value and Action-Level-1 values for condensate dissolved O_2 are different than the DCD values, the EPRI Guidelines, and a previous RAI response (RAI 383-3002, Question 10.4.6-7, dated July 06, 2009). In its response to **RAI 543-4368, Question 10.4.6-12** (Reference 9), dated April 26, 2010, the applicant declared this to be an error.

7. **Cold Shutdown/Wet Layup**. All DCD limits for control parameters during cold shutdown/wet layup (Table 10.3.5-2) are identical to the corresponding values in the EPRI Guidelines (Table 5-1).

8. **Blowdown Sampling**. This subject is addressed in more detail in SE Section 10.4.6.4.4 below.

9. **Action-Level-2 and -3 Limits (Power Operation)**. A substantial difference between the DCD and the EPRI Guidelines is that the DCD lacks Action-Level-2 and -3 limits for control parameters O_2 (in condensate and feedwater) and Na, Cl, and SO_4 (blowdown). For these last three impurities, the applicant specifies limits which do not correspond with the EPRI limits, and are discussed in detail in SE Section 10.4.6.3.3. In its September 16, 2009, response to **RAI 441-3461, Question 10.04.06-09**, the applicant stated that it would supply all the missing limits. However, the table that was provided in the RAI response did not contain Action-Level-3 values for Na, SO_4, and feedwater $O_{2;}$ and Action-Level-2 values for condensate O_2. As noted in Item 6 above, there is also a conflict in the Action-Level-1 value. Also in this table, the applicant gave Action-Level-2 limits for Cl and SO_4 that are higher than the limits provided by the EPRI Guidelines, but these limits are identical to values the applicant has defended (see SE Section 10.4.6.4.3 below). However, the DCD Action-Level-3 limit for Cl is nearly an order of magnitude higher than the corresponding value in the EPRI Guidelines. The applicant has not supplied any explanation or justification for this discrepancy. Combined with the definition of Action Level 3 proposed by the applicant, this would allow operation with greater than 2000-ppb Cl for up to 24 hours. Further, under the proposed DCD parameters, a US-APWR plant could also operate for up to 1 week with chloride between 100-2000 ppm given the definition of Action Level 2 and the proposed Cl action levels. In its response to RAI 543-4368, Question 10.4.6-13 (Reference 9), dated April 26, 2010, the applicant declared that the EPRI Guidelines do not define Action Level 2 limits for both (Table 5-4 for feedwater, Table 5-6 for condensate). In addition, the applicant failed to specify a COL information item, or to justify deviation from the EPRI Guidelines, as requested by RAI 543-4368, Question 10.4.6-13. Finally, the applicant claimed that corrective actions (also requested by RAI 543-4368, Question 10.4.6-13) were answered in a previous RAI response (Reference 5, Question 9). While this is true, restrictions and consequences for plant operation were not addressed in any of the RAI responses as requested. In fact, RAI 543-4368, Question 10.4.6-13 was prompted by the inadequacy of this previous response.

10.4.6.4.3 Impurity Concentration Limits

In response to RAI 235-2134, Question 10.4.6-1 (Reference 2), dated March 25, 2009, the applicant provided a basis for specifying concentration limits during power operation and heatup which are different from EPRI Guidelines. In its response to RAI 383-3002, Question 10.4.6-6 (Reference 4), dated July 06, 2009, the applicant further explains this difference. Since sodium (Na), chloride (Cl), and sulfate (SO_4) are control parameters, the EPRI Guidelines must be satisfied or other limits must be justified.

For power operation, the Action Level 2 limits recommended by the EPRI Guidelines are 50 ppb for Na, Cl, and SO_4 (Table 5-5). For heatup, the EPRI Guidelines indicate the same limits and consequences (Table 5-3), but list them in notes and do not label them "Action Level 2." Similarly, the Action Level 3 values for all three species are 250 ppb for power operation, with identical limits and consequences specified for heatup. The applicant specifies limits of 100 ppb for Cl and SO_4 for both heatup and power operation, in place of the Action Level 2 values from the EPRI Guidelines. The response to RAI No. 383-3002 Question No. 10.4.6-6 (Reference 4) directly addressed limits for heatup, but the applicant has also sought to establish the same limits for power operation [see the Table in the applicant's response to RAI 441-3461, Question 10.4.6-9 (Reference 5)]. The applicant mentions that it assisted in the experimental efforts yielding some of the data used by EPRI to establish the guidelines. In addition, the applicant has developed codes based on this data which the applicant has used to justify the limits which have been specified in the DCD. The use of one code to calculate crevice pH is compared with data in Figure 2 of Reference 4; the staff finds the code results correlate well with data through a range of pH = 3 through pH = 10. Judging from the figure, the standard error (standard deviation) appears to be about 1 pH unit.

The crucial part of the applicant's justification lies in the selection of a concentration factor (CF) of 10^5. (This quantity represents the increase in impurity concentrations in crevices and other isolated, flow-starved regions, when compared to the bulk liquid concentration.) In Figure 3 of Reference 4, the applicant shows a plot of CF for various operating conditions, and with two different types of tube support plates. For cylindrical-hole plates, the CF is in the range of 10^5-10^6, whereas for quatrefoil-shaped holes (labeled "BEC-type" in the figure), the CF is almost two orders of magnitude lower (range 10^3-10^4). The US-APWR intends to use a trefoil-shaped support plate, which should resemble the quatrefoil in CF. In establishing impurity limits, MHI has adopted a CF of 10^5, which is conservative by at least one order of magnitude.

The EPRI Guidelines (Figure 2-11) indicate acceptable pH ranges for different types of tube material. The acceptable range for 690TT (to be used in the US-APWR) is approximately $4.5 < pH < 11$. As a conservative measure, the applicant has reduced the range to $5 < pH < 10$, which is similar to the acceptable range of other types of tube material.

Using a CF of 10^5, the applicant calculated (using another of its codes) a correlation between Na bulk concentration and crevice pH at operating temperature, assumed to be 545°F (285°C). From the correlation (shown in Figure 4 of Reference 4), the upper limit (pH=10) is attained only when the bulk Na concentration increases to 50 ppb. For Na impurity, the lower limit is inconsequential. Hence, the applicant has selected 50 ppb as the limit for Na bulk concentration.

To obtain limits for Cl and SO_4, the applicant fixes the Na concentration at its maximum value of 50 ppb. However, this is not conservative, since the EPRI Guidelines (Figures 3-3 and 3-7) indicate that crevice pH increases as the ratios Na:Cl and Na:SO4 increase. Thus, for fixed values of Cl or SO4, raising Na to its maximum value will increase the crevice pH. However, limits for Cl and SO_4 will be determined by the lower limit for crevice pH; hence, a lower value of Na should be used. In its response to RAI 543-4368, Question 10.4.6-14 (Reference 9), dated April 26, 2010, the applicant notes that the primary mode for these ionic impurities to enter the system is from in-leakage of seawater or river water. From the applicant's data, the ratio of Na:Cl is approximately 1:2 (seawater) and 1:1 (river water); for the ratio of Na:SO_4 it is about 4:1 (seawater) and 1:2 (river water). From the sample calculations in the EPRI Guidelines, we note that for CF=10^5, the crevice pH stays above 5 for ratios Na:Cl=1:1 (Figure 3-5), Na:Cl=1:3 (Fig. 3-6), Na:SO_4=3:1, Na:SO_4=1:1, and Na:SO_4=1:3 (Figure 3-7). Thus, the staff finds this an acceptable explanation of appropriate ratios between Na and the anions. Calculations using the applicant's codes produce correlations between Cl or SO_4 and pH, shown in Figures 5 and 6 of Reference 4. From these figures, the minimum pH of 5 is reached when bulk Cl and SO_4 levels exceed approximately 110 ppb and 130 ppb, respectively. The applicant has set limits for these impurities at 100 ppb, slightly (and conservatively) lower than the concentration corresponding to the minimum pH. However, it should be noted that the applicant's limits are still near a region of rapid pH decrease, and there is considerable uncertainty associated with such rapid change. Nevertheless, the limit (100 ppb) for Cl is validated by independent experimental results, cited in Reference 4, Figure 7.

In its response to RAI 383-3002, Question No.10.4.6-6 (Reference 4), dated July 6, 2009, the applicant's analysis conflicts slightly with the DCD (Table 10.3.5-3). The applicant has rectified this inconsistency by proposing a minor change in DCD Table 10.3.5-3, described in the applicant's response to RAI 441-3461, Question 10.04.06-8 (Reference 5) dated April 26, 2010.

In summary, the analysis for the appropriate limits for the impurities Na, Cl, and SO_4 has been undertaken using the applicant's own code calculations. It has also involved conservative assumptions regarding CF, acceptable range for crevice pH, and maximum Na concentration (for calculation of Cl and SO_4 limits). There are uncertainties associated with the pH calculation and regarding the proximity of limits for Cl and SO_4 to regions of rapid change. The staff finds that these uncertainties are adequately covered by the conservative assumptions made by the applicant. In large part, the higher limits for the US-APWR are made possible because of the use of alloy 690TT tubing and trefoil-shaped holes in the support plates. The staff observed in NUREG-1841 "U.S. Operating Experience with Thermally Treated Alloy 690 Steam Generator Tubes," that there is significant operating experience using thermally treated alloy 690 for SG tubes. Thus, based on the analysis supplied by the applicant, the staff concludes that these impurity limits are adequate to assure integrity of secondary system components, even though they differ from the recommended limits in the EPRI Guidelines.

10.4.6.4.4 Sampling of Blowdown Water

The applicant proposes not to specify in the DCD measurement of pH, cation conductivity, hydrazine, morpholine, dimethylamine, and silica in SG blowdown water; although the staff notes that a COL applicant referencing the US-APWR design could choose to follow the EPRI Guidelines. In the following section, the staff discusses its review of the applicant's justification for this deviation from the EPRI Guidelines.

In its July 6, 2009, response to **RAI 383-3002, Question 10.4.6-7**, the applicant provided its justification for sampling frequencies for blowdown water (provided in Table 1 of the response) which differ from the EPRI Guidelines. Of the parameters listed (pH, cation conductivity, hydrazine, and silica), the applicant correctly noted that pH and hydrazine are not control values, so the continuous monitoring suggested by the EPRI Guidelines is not required.

In its response, the applicant did not consider cation conductivity to be a control value, although the EPRI Guidelines do regard it as such (Table 5-5, "EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines"). Cation conductivity indirectly measures the level of total anions in the system. The applicant argued that since the two principal anions (Cl and SO_4) are measured directly in blowdown samples, the cation conductivity is redundant unless other anions are present in significant amounts. The EPRI Guidelines list the three most likely additional anions: boric acid, organic acids, and fluoride. Since the US-APWR does not use boric acid in secondary water, the applicant contended that this species could only come through leaks from the primary system; in this case, other indicators (e.g. radioactive species) would require attention. Fluoride is not a component of any additive used in the US-APWR, although the applicant recognized that residual amounts from construction might be present at start-up; however, it argued that these would most likely be eliminated during early purification processes. The applicant also recognized that it is possible that degradation of morpholine and dimethylamine could alter the conductivity measurement slightly, but argued that this effect is minor and gradual, and therefore, would not necessitate cation conductivity being a control value. Indeed, the EPRI Guidelines (Table 5.5 notes) mention that conductivity is elevated in systems containing organic amines (as is the case here), requiring special consideration of sample results. Nonetheless, under the EPRI Guidelines, the operator would take into account this consideration to accommodate the higher values for Cl and SO_4 that are established in Section 10.4.6.4.3, "Safety Evaluation."

For pH, cation conductivity, and hydrazine, the applicant asserted that it is unlikely that these quantities would vary appreciably between blowdown effluent and feedwater. They are to be sampled continuously in feedwater, so the applicant reasoned that blowdown sampling for them is unnecessary and can be excluded. In its September 16, 2009, response to **RAI 441-3461, Revision 1, Question 10.4.6-10**, the applicant claimed that several operating Japanese plants follow such a sampling plan without complications. The staff finds merit in this argument. Hydrazine and morpholine both have boiling points near that of water (237°F (114°C) and 264 °F (129 °C), respectively), which would suggest that they distribute almost uniformly between the liquid and vapor phases. Similar behavior is observed at high temperatures as well. The vapor pressure of hydrazine is about 66 percent that of water at 549°F (28°C) (Reference 6), and morpholine has a critical temperature and critical pressure similar to those of water (Reference 7). However, both of these species should readily pass through the boiling-vapor transport-condensation pathway and eventually reenter the SG in the feedwater. Dimethylamine is even more volatile than water [boiling point (BP)=7°C at 1 atmosphere (atm), and critical temperature and pressure are considerably lower than those of water (Reference 8) [1], so dimethylamine would be expected to migrate preferentially to the vapor space. Of course, one of the reasons these compounds have found wide use is their favorable distribution in all phases. Thus, the staff finds that feedwater samples will provide reliable indication of distribution throughout the system.

The value of pH is, of course, directly related to the concentration of morpholine and dimethylamine, and is therefore a redundant measure of their presence or absence. If these chemicals are well distributed, then pH values should be fairly uniform throughout the system.

In Table 2 of its response to RAI 383-3002, Question No. 10.4.6-7 (Reference 4), dated July 6, 2009, the applicant has provided justification for the lack of continuous monitoring of cation conductivity in the blowdown water. However, in Table 4 of the same response, the applicant states that it will indeed measure cation conductivity continuously in the blowdown sample (along with Na). The applicant's position was clarified in its response to RAI 543-4368, Question 10.4.6-15 (Reference 9), dated April 26, 2010, where the applicant reiterated that there were no plans to measure cation conductivity in blowdown water, and that Table 4 (Reference 4) was in error.

The applicant has also eliminated the measurement of silica from the blowdown samples, justified by the fact that they do not use superheated steam, and there is virtually no source except for make-up water. The EPRI Guidelines give credibility to this reasoning, since the primary difficulty has occurred in facilities using superheated steam. They note (p. 5-20), "At the pressure range of PWR steam generators, however, silica is not likely to lead to a similar problem." The applicant's design will limit silica by monitoring silica in make-up water. In addition, the concentration of silica is a diagnostic parameter; hence, the EPRI Guidelines do not require it to be measured in blowdown samples.

On the basis of the above discussion and the need for COL applicants to establish sampling plans for feedwater and makeup water, the staff concludes that blowdown sampling of pH, cation conductivity, hydrazine, morpholine, dimethylamine, and silica is unnecessary.

[1] D. Shvedov and P.R. Tremaine, "Thermodynamic Properties of Aqueous Dimethylamine and Dimethylammonium Chloride at Temperatures from 283 K to 523 K: Apparent Molar Volumes, Heat Capacities, and Temperature Dependence of Ionization," J. Soln. Chem. 26(12), 1113-43 (1997).

10.4.6.5 Combined License Information Items

There is no COL information items associated with Section 10.4.6, "Condensate Polishing System," identified in the US-APWR DCD.

10.4.6.6 Conclusions

The applicant has described effective systems and procedures to maintain the purity of secondary side coolant. The applicant has justified a slight deviation from the EPRI Guidelines in sampling of blowdown water. Site specific details of the secondary side water chemistry will be provided by the COL holder. However, the site specific water chemistry program must satisfy the EPRI Guidelines or adequately defend other values. The staff therefore concludes, based on the information supplied by the applicant, that the requirements of GDC 14 and 10 CFR 52.47(b)(1) to maintain the Reactor Coolant Pressure Boundary (RCPB) will be satisfied.

10.4.6.7 References

1. *Pressurized Water Reactor Secondary Water Chemistry Guidelines*, Revision 6, Electric Power Research Institute, dated December 2004.

2. Letter from Yoshiki Ogata, MHI, to NRC dated March 25, 2009; Docket No. 52-021, MHI Ref: UAP-HF-09106; Subject: MHI's Response to US-APWR DCD RAI No. 235-2134 (ADAMS Accession No. ML090890519).

3. Letter from Yoshiki Ogata, MHI, to NRC dated April 14, 2009; Docket No. 52-021, MHI Ref: UAP-HF-09171; Subject: MHI's Response to US-APWR DCD RAI No. 280-2060 (ADAMS Accession No. ML091060217).

4. Letter from Yoshiki Ogata, MHI, to NRC dated July 6, 2009; Docket No. 52-021, MHI Ref: UAP-HF-09364; Subject: MHI's Response to US-APWR DCD RAI No. 383-3002 (ADAMS Accession No. ML091910255).

5. Letter from Yoshiki Ogata, MHI, to NRC dated September 16, 2009; Docket No. 52-021, MHI Ref: UAP-HF-09451; Subject: MHI's Response to US-APWR DCD RAI No. 441-3461 (ADAMS Accession No. ML092610532).

6. D. Giordano, "Survey of the Thermodynamic Properties of Hydrazine," *J. Chem. Eng. Data* **46**, 486-505 (2001).

7. P.R. Tremaine, D. Shvedov, and C. Xiao, "Thermodynamic Properties of Aqueous Morpholine and Morpholinium Chloride at Temperatures from 10 to 300°C: Apparent Molar Volumes, Heat Capacities, and Temperature Dependence of Ionization," *J. Phys. Chem. B* **101**, 409-19 (1997).

8. D. Shvedov and P.R. Tremaine, "Thermodynamic Properties of Aqueous Dimethylamine and Dimethylammonium Chloride at Temperatures from 283 K to 523 K: Apparent Molar Volumes, Heat Capacities, and Temperature Dependence of Ionization," J. Soln. Chem. 26(12), 1113-43 (1997).

10.4.7 Condensate and Feedwater System

10.4.7.1 Introduction

The CFS provides feedwater to the SGs. The CFS also provides condensate cleanup capability and maintains condensate quality through deaeration and interfacing with the main condenser, CPS, secondary side chemical injection system, and secondary sampling system. The safety-related portion of the system is required to function following a design basis accident to provide containment isolation for the main lines routed into containment.

10.4.7.2 Summary of Application

Tier 1: The Tier 1 information associated with this section is found in Tier 1, Section 2.7.1.9, “Condensate and Feedwater System (CFS),” of the US-APWR DCD. Figure 2.7.1.9-1, “Feedwater System,” illustrates the main feedwater lines, showing the arrangement of the safety-related CFS components.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.4.7, “Condensate and Feedwater System,” of the US-APWR DCD, summarized here in part, as follows:

- The CFS is composed of the condensate system (CDS) and the feedwater system (FWS). The CFS supplies the SGs with heated feedwater in a closed heat cycle using regenerative feedwater heating. The CDS takes suction from the main condenser hotwell and pumps condensate to the deaerator utilizing the condensate pumps. The FWS takes suction from the deaerator and pumps feedwater to the SGs using the feedwater booster/main feedwater pumps, through piping containing main feedwater regulation valves (MFRVs), main feedwater isolation valves (MFIVs), and main feedwater check valves (MFCVs). The SG water filling lines have a common flow measuring element and SG water filling control valve (SGWFCV).

- The CDS consists of the main condenser (described in Section 10.4.1, “Main Condensers”), CPS (described in Section 10.4.6, “Condensate Polishing System”), condensate pumps, five-stage, LP heaters including the deaerator, piping, associated valves and instrumentation.

ITAAC: The ITAAC associated with Tier 2, Section 10.4.7 are given in Tier 1, Section 2.7.1.9.2, “Inspections, Tests, Analyses, and Acceptance Criteria,” of the US-APWR DCD.

TSs: The TSs associated with Tier 2, Section 10.4.7 is given in Tier 2, Chapter 16, Section 3.7.3, of the US-APWR DCD for the following valves:

- MFIVs
- MFRVs
- Main Feedwater Bypass Regulation Valves
- SGWFCVs

10.4.7.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 10.4.7, “Condensate and Feedwater

System," of NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," and are summarized below. Review interfaces with other SRP sections can be found in SRP Section 10.4.7.

1. GDC 2 to 10 CFR 50, Appendix A as related to safety-related portions of the CFS designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

2. GDC 4 as related to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation, as well as during upset or accident conditions.

3. GDC 5 as related to the capability of shared systems and components important to safety to perform required safety functions.

4. GDC 44 as it relates to:

    a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

    b. Redundancy of components so that under accident conditions, the safety functions can be performed assuming a single active component failure. (This may be coincident with the LOOP for certain events.)

    c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

5. GDC 45, "Cooling Water System Inspection," as related to design provisions to permit periodic in-service inspection of system components and equipment.

6. GDC 46, "Cooling Water System Testing," as related to design provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

10.4.7.4 Technical Evaluation

The staff reviewed the US-APWR CFS design as described in the US-APWR DCD. The review, which was performed in accordance with SRP, Section 10.4.7, "Condensate and Feedwater System," Revision 4, dated March 2007, was based on the Tier 1 and Tier 2 information contained in Revision 2 of the US-APWR FSAR. Conformance with the acceptance criteria of SRP, Section 10.4.7 formed the basis for the evaluation of the CFS with respect to the applicable regulations. The results and conclusions of the staff's review of the CFS are discussed below. The evaluation addresses compliance with the SRP acceptance criteria listed in Section 10.4.7.3 of this SE.

The CFS provides feedwater at the required temperature, pressure, and flow rate to the SGs. It consists of the CDS which runs from the condenser hotwell outlet to the deaerator and the FWS which runs from the outlet of the deaerator to the SG nozzles. The CDS is located within the TB, and the safety-related portion of the FWS is located within the reactor building and inside containment. The major components of the CFS include condensate pumps, condensate polishers, a gland steam condenser, three strings of LP heaters, main feedwater pumps, feedwater booster pumps, two strings of HP feedwater heaters, condensate and feedwater regulating valves, MFIVs, and associated piping, valves instrumentation, and controls. The system description, including component design parameters and system flow diagrams are given in Section 10.4.7, “Condensate and Feedwater System,” of the US-APWR DCD.

The CFS has three 50 percent capacity condensate pumps, connected in parallel. During rated power operation, two pumps are operating; the third pump is on standby available for automatic start. All four feedwater booster/main feedwater pumps are operated during rated power operation. Each pump is designed to deliver 25 percent rated feedwater flow during rated operation. With an increase in pump speed, each pump is also capable of delivering 33 percent rated feedwater flow at rated operating pressure.

The CFS does not perform safety-related functions with respect to transferring heat from SSCs important to safety to the ultimate heat sink, and is not required to supply feedwater under accident conditions to affect plant shutdown or to mitigate the consequences of an accident. That function is performed by the EFW system, which is reviewed in Section 10.4.9, “Emergency Feedwater System,” of this SE. The safety-related function of the CFS is to provide containment and feedwater isolation following a design basis accident.

GDC 2, “Design Bases for Protection Against Natural Phenomena”

The staff reviewed the CFS for compliance with the requirements of GDC 2. Compliance with the requirements of GDC 2 is based on adherence to Position C.1 of RG 1.29, “Seismic Design Classification,” for the safety-related portions of the system, and Position C2 for the non-safety-related portions of the system.

The DCD indicates that the portion of the feedwater system from the SG inlets outward through the containment and up to and including the MFIVs is safety-related and has a safety-related function of providing containment and feedwater isolation following a design basis accident.

The safety-related portion of the CFS is required to remain functional after a design-basis accident to provide containment and feedwater isolation. As indicated in Table 3.2-2, “Classication of Mechanical and Fluid Systems, Components, and Equipment,” of the Tier 2 portion of the DCD, the CFS piping from the SGs inlet outward through the containment up to and including the MFIVs are seismic Category I and designed to Safety Class 2 (ASME Code Section III Class 2) requirements. Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment,” of Tier 2 of the DCD also indicates that the non-safety-related portion of the CFS from the MFIV inlets to the piping restraints at the interface between the auxiliary building and the TB is designed in accordance with the requirements of Section III of the ASME Code for Class 3 components, and it is also seismic Category I.

The system is designed such that adverse environmental conditions such as tornados and floods will not impair its safety function. The safety-related portions of the CFS are seismic Category I to withstand the effects of earthquakes and are housed inside the containment and inside the reactor building. Thus they are protected from the effects of wind and tornados, as described in Tier 2, Section 3.3, "Wind and Tornado Loadings," of the DCD; floods as described in Section 3.4, "Water Level (Flood) Design," of the DCD; and seismic events as described in Section 3.7, "Seismic Design," of the DCD. The review of the containment and reactor building as it relates to the adequacy of these buildings to protect against natural phenomena is reviewed in the corresponding sections of this SE. The safety related portions of the system are designed in accordance with RG 1.29 (Position C1). The non-safety related portions of the system are designed in accordance with RG 1.29 (Position C2). Therefore, the staff concludes that the CFS meets the requirements of GDC 2 as they relate to protecting the system against seismic and other natural phenomena.

GDC 4, "Environmental and Dynamic Effects Design Bases"

The staff reviewed the CFS for compliance with the requirements of GDC 4, "Environmental and Dynamic Effects Design Bases," as related to the dynamic effects associated with possible fluid flow instabilities, including induced water hammer and the effects of pipe breaks. Compliance with the requirements of GDC 4 is based on identification of the essential portions of the system as protected from dynamic effects, including internally and externally generated missiles, pipe whip and jet impingement due to high and moderate energy missiles and water hammer. The guidance in Branch Technical Position (BTP) 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators," specifically recommends that the CFS be designed to achieve the following provisions:

- Prevent or delay water draining from the feedring following a drop in SG water level.
- Minimize the volume of feedwater piping external to the SG which could pocket steam using the shortest horizontal run of inlet piping to the feedring.
- Perform tests, acceptable to the NRC, to verify that unacceptable feedwater hammer will not occur and provide test procedures for staff approval.
- Implement pipe refill flow limits where practical.

The applicant states in DCD Tier 2, Section 10.4.7.1.2, "Safety Design Basis," that the design includes suitable protection so that the dynamic effects, including internally generated missiles, pipe whipping, and discharge fluids due to equipment malfunctions; and external events does not pose a threat to system integrity. It is also indicated in this section that that the safety-related portions are protected from: missiles as addressed in DCD Tier 2, Section 3.5, "Missile Protection," is protected against dynamic effects of postulated pipe ruptures as addressed in DCD Tier 2, Section 3.6, "Protection Against Dynamic Effects Associated with Postulated Rupture of Piping," and environmental design as addressed in DCD Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment."

The applicant addresses water hammer prevention in DCD Tier 2, Section 10.4.7.7, "Water Hammer Prevention." In Section 10.4.7.7, the applicant states that the feedwater system and SG design minimize the potential for water hammer and subsequent effects. The applicant also states that system design features will prevent the formation of a steam pocket in the feedwater piping, and that prevention and mitigation of a feedline-related water hammer is accomplished through operation of the feedwater delivery system.

Design features identified in the DCD to prevent formation of stem pockets, and water hammer includes:

- The main feedwater connection on each of the SGs is the highest point of each feedwater heater line downstream of the MFIV, and is sloped so that it does not drain into the SGs.
- The feedwater lines contain no high-point pockets that could trap steam and lead to water hammer.
- The horizontal pipe length from the main nozzle to the downward turning elbow of each SG is minimized.
- The 91TT-1 SG feedwater ring has a welded thermal sleeve and top-discharge perforated nozzles so that water within the feedwater ring cannot drain out and so steam cannot become trapped. The perforated nozzles are located at the highest point in the feedwater system so any vapor within the feedwater ring can vent.

The US-APWR design has main feedwater control valves installed outside containment in each main feedwater line, which prevents reverse flow from the SG whenever the feedwater pumps are tripped. DCD Tier 2, Section 14.2 (14.2.12.1.29), "Preoperational Tests," includes preoperational testing that requires verification that water hammer does not occur in system components or piping, or inside the SGs, during normal system startup and operation.

SRP Section 10.4.7, Section IV.2 has guidelines for the COL applicant to review operating and maintenance procedures to ensure that precautions taken will minimize, or avoid water hammers.

Although the applicant stated in DCD Tier 2, Section 10.4.7.7, "Water Hammer Prevention," that water hammer prevention and mitigation are implemented in accordance with NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plant," the staff noted that there was no COL information item for the COL applicants to review operating and maintenance procedures to ensure that they include precautions to minimize or eliminate water hammer. Consequently, the staff issued RAI 124-1638, Question 10.4.7-1, dated December 4, 2008, requesting that the applicant propose a COL information item to provide operating and maintenance procedures to address water hammer issues.

In its response to RAI 124-1638, Question 10.4.7-1, dated June 1, 2009, the applicant proposed adding COL Information Item 10.4(6), requiring COL applicants to develop operating and maintenance procedures for the CFS to aid in reducing the frequency of water hammer events. The applicant also proposed adding to the DCD, supplemental descriptions and a list of key elements regarding water hammer and mitigation to be included in the CFS operating and maintenance procedures.

The staff has confirmed that Revision 2 FSAR Tier 2, Section 10.4.7 was revised as committed in the RAI response. The staff reviewed the COL item and the revision of the DCD text and determined that the concerns raised by the staff in RAI 124-1638, Question 10.4.7-1 were resolved since the COL applicants will develop operating and maintenance procedures for the

CFS to minimize water hammer

The staff finds that the CFS design meets the requirements of GDC 4, “Environmental and Dynamic Effects Design Bases,” as related to the dynamic effects associated with possible fluid instabilities, including induced water hammer and the effects of pipe breaks.

GDC 5, “Sharing of Structures, Systems, and Components"

The US-APWR is designed as a single facility, so the requirement of GDC 5 for sharing of systems between units does not apply.

GDC 44, “Cooling Water”

The requirements of GDC 44, as related to the capability to transfer heat from SSCs important to safety to an ultimate heat sink are met by demonstrating that the CFS is capable of providing heat removal under both normal operating and accident conditions.

Each main feedwater line to the SG contains a feedwater flow element, a MFIV, a MFRV, and a MFCV, all located in the reactor building. The MFIVs, installed in each of the four feedwater lines outside the containment, are used to prevent uncontrolled blowdown from the SGs in the event of a feedwater line break (FLB). The MFRVs are used to control feedwater flow rate to the SG during normal operation. During normal and upset conditions, the MFCV prevents reverse flow from the SG whenever the feedwater pumps are tripped. In addition, the closure of the valves prevents more than one SG from blowing down in the event of a FLB. The MFCV is designed to limit blowdown from the SG and to prevent water hammer due to sudden valve closure.

The safety-related isolation function of the CFS is accomplished by redundant means. Automatic isolation of the main FWS is provided when required to mitigate the consequences of an accident. A single active component failure of the safety-related portion of the system does not compromise the safety function of the system. DCD Tier 2, Table 10.4.7-3, “Condensate and Feedwater System Failure Modes and Effects Analysis,” provides the failure modes and effects analysis of the safety-related active components of the FWS. As indicated in DCD Tier 2, Section 10.4.7.2.5, “Emergency Operation,” the CFS is not required to supply feedwater under accident conditions to affect plant shutdown or to mitigate consequences of an accident. In the event of an accident, the SGs are fed with water from the EFW system which is connected to CFS piping through an EFW system pump suction tie line located on the safety-related portion of the CFS inside the reactor building.

Based on the discussion above, the staff finds that the US-APWR CFS meets the requirements of GDC 44, “Cooling Water,” by providing a redundant and isolable system capable of transferring heat loads from the reactor system to a heat sink under both normal and upset conditions.

GDC 45, “Inspection of Cooling Water System,” and GDC 46, “Testing of Cooling Water System”

The staff reviewed the CFS design to ensure design provisions are provided for periodic inspection of systems, components and equipment, as required by GDC 45. The FSAR states that in-service testing is performed in accordance with the requirements of ASME Section IX. Therefore, GDC 45 is satisfied with respect to permitting periodic inservice inspection (ISI) of system components and equipment.

The design of the safety-related portions of the CFS was reviewed by staff to ensure there are provisions for the performance of periodic functional testing of the system and components, as required by GDC 46. In DCD Tier 2, Section 10.4.7.1.2, “Safety Design Basis,” it is stated that the portion of the FWS from the SG inlets outward through the containment up to and including the MFIVs is constructed in accordance with the requirements of ASME Code, Section III, Class 2 components and is designed to seismic Category I requirements. The piping upstream of MFIVs to the first piping restraint at the interface between the reactor building (main steam/feedwater piping area) and TB is constructed in accordance with the requirements of ASME Code, Section III Class 3 components and is designed to seismic Category I requirements. Section 10.4.7.1.2 also states that the portion of the FWS to be constructed in accordance with ASME Code, Section III Class 2 requirements allows access to welds and uses removable insulation for inservice inspection, in accordance with ASME Code, Section XI. The portion of the FWS to be constructed in accordance with ASME Code, Section, Class 3 requirements is also designed and configured to accommodate inservice inspection in accordance with ASME Code, Section XI. Based on the above, the staff concludes the CFS design meets the requirements of GDC 46 since the design includes provisions for the performance of periodic functional testing of the system and components.

Technical Specifications

The staff reviewed FSAR Tier 2 Chapter 16, TS 3.7.3 for applicability to the Main Feedwater System (MFWS). TS 3.7.3 provide limiting conditions for operation and surveillance requirements for the MFWS valves. TS Bases 3.7.3 background description is consistent with the FSAR Tier 2 description of MFWS valves. The staff concludes that TS 3.7.3 appropriately addresses the limiting conditions for operation and surveillance requirements for the MFWS valves.

10.4.7.5 Combined License (COL) Information Items

Table 10.4.7-1
U.S. APWR Combined License Information Items

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
10.4(6)	Provide operating and maintenance procedures for water hammer prevention. The Combined License Applicant will develop a milestone schedule for implementation of the procedure.	10.4.7	Y	

10.4.7.6 Conclusions

The staff finds that the review of the DCD application supported that the CFS functional design is acceptable for the reasons set forth above and because it meets appropriate regulatory requirements including GDC 2 regarding protection from natural phenomena, GDC 4 regarding protection against missiles and effects of pipe break, GDC 5 regarding shared systems, GDC 44 regarding transferring heat to the ultimate heat sink, GDC 45 regarding inspections, GDC 46 regarding periodic testing, and 10 CFR 52.47(b)(1) regarding ITAAC.

10.4.8. Steam Generator Blowdown System (PWR)

10.4.8.1 Introduction

The SGBS assists in maintaining secondary side water chemistry within acceptable limits during normal plant operation and during AOOs due to main condenser in-leakage or primary to secondary SG tube leakage. This is done by removing impurities concentrated in SGs by continuous blowdown of secondary side water from the SGs. The SGBS has a safety-related function of isolating the secondary side of the SG. This provides a heat sink for a safe shutdown or to mitigate the consequences of a design basis accident.

10.4.8.2 Summary of Application

Tier 1: The Tier 1 information associated with this section is found in Tier 1 Section 2.7.1.10, “Steam Generator Blowdown System,” of the US-APWR DCD. Figure 2.7.1.10-1, “Design Description,” illustrates the SGBS.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.4.8, “Steam Generator Blowdown System,” of the US-APWR DCD, summarized here in part, as follows:

- The SGBS consists of a flash tank, regenerative heat exchangers, non-regenerative coolers, filters, demineralizers, piping, valves, and instrumentation. The flash tank, regenerative heat exchangers and non-regenerative coolers are provided to cool the blowdown water with heat recovery, while the filters and demineralizers are provided to purify the blowdown water. One blowdown line per SG is provided. The blowdown line from each SG is provided with two flow paths, a line for purifying blowdown water during normal plant operation and a line for discharging water to the waste water system (WWS) or the condenser used during startup and abnormal water chemistry conditions. A blowdown sample line with a safety-related containment isolation valve is provided for each SG for sampling. Two safety-related SG blowdown isolation valves in the series are provided on each of the blowdown lines. The SGBS flow diagrams are shown in Figures 10.4.8-1, “Steam Generator Blowdown System Piping and Instrumentation Diagram (1/2),” and 10.4.8-2, “Steam Generator Blowdown System Piping and Instrumentation Diagram (2/2).” Component design parameters are provided in Table 10.4.8-1, “Steam Generator Blowdown System Major Component Design Parameters.”

ITAAC: The ITAAC associated with Tier 2 Section 10.4.8 are given in Tier 1, Section 2.7.1.10.2, “Inspections, Tests, Analyses, and Acceptance Criteria,” of the US-APWR DCD.

TSs: There is no TSs for this area of review.

10.4.8.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 10.4.8, “Steam Generator Blowdown System,” of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 10.4.8.I, “Design Bases,” of NUREG-0800.

1. GDC 1, as it relates to system components being designed, fabricated, erected, and tested for quality standards.

2. GDC 2, as it relates to system components designed to seismic Category 1 requirements.

3. GDC 13, “Instrumentation and Control,” as it relates to monitoring system variables that can affect the reactor coolant pressure boundary and maintaining them within prescribed operating ranges.

4. GDC 14, as it relates to secondary water chemistry control to maintain the integrity of the primary coolant pressure boundary.

5. 10 CFR 52.47(b)(1), which requires that a DC application address the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC has been built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

10.4.8.4 Technical Evaluation

The staff reviewed the SGBS in accordance with NUREG-0800 Section 10.4.8, “Steam Generator Blowdown System.” Acceptance of the SGBS is based on meeting the requirements of GDC 1, GDC 2, GDC 13, and GDC 14.

The principal function of the SGBS is to maintain the secondary-side water chemistry in the SGs within permissible limits by removing particulate and dissolved impurities. The SGBS is designed to continuously perform a blowdown of each of the four SGs, purifiy the blowdown, and return it to the steam cycle through the deaerator and condenser. Each blowdown line contains two isolation valves in series in order to maintain the SG inventory. The system is designed to close the valves automatically under any of the following conditions: a containment isolation signal, high radiation from the SG blowdown water or condenser, actuation of the EFW pump, and HP or water level in the blowdown flash tank. The blowdown sample lines are isolated under the same conditions except the flash tank signals.

During normal power operation the blowdown flow rate is designed to be approximately 0.5 percent to 1 percent of the maximum steaming rate (MSR). A higher blowdown flow rate of up to 3 percent of MSR can be used to reduce solids content in the SGs during plant startup. The blowdown water is drawn from a location above the tube sheet of each SG where impurities are expected to accumulate. In its response to RAI 251-2146, Question 10.4.8-1, dated April 01, 2009, the applicant clarified the DCD by adding Figure 10.4.8-3, “Concept of Peripheral Blowdown,” and a more detailed description of the blowdown nozzle arrangement. The US-APWR has a peripheral blowdown system design, with the nozzle located about seven inches below the top of the tubesheet in the bottom of a peripheral groove.

Each SG has two flow paths, one to the flash tank and then water purification, and a second path to the condenser or WWS during startup or abnormal conditions. Each line has a throttle valve to control the blowdown rate. Steam returns to the steam cycle through the deaerator (CFS). Prior to purification, the water in the flash tank is further cooled in the regenerative and non-regenerative heat exchangers. The heat exchangers are designed to cool the blowdown water before it enters the demineralizer resin beds. The demineralizers have an inlet filter for particulates and an outlet filter that prevents resin fines from entering the feedwater. The demineralizers remove dissolved anion and cation impurities. The demineralizers also remove the pH control additives, as discussed in DCD Section 10.4.10.2.3, "Secondary Side Chemical Injection System." The purified water is returned to the feedwater system at the condenser. Each of the two demineralizer trains can process 100 percent of the maximum blowdown rate. In its response to RAI 251-2146, Question 10.04.08-02, dated April 01, 2009, the applicant clarified that processing 100 percent of the maximum blowdown rate means each demineralizer train can process all of the liquid phase flowing from the SG blowdown flash tank. At the maximum blowdown rate of 202,000 lb/hr to the flash tank, the liquid phase flow rate is approximately 159,000 lb/hr, or approximately 320 gpm at the temperature and pressure conditions of the demineralizers.

The heat exchanger trains are 50 percent capacity each. At less than 0.5 percent MSR, one set of Heat Exchanger is needed. At 1 percent MSR, both trains are used. During plant startup, the blowdown rate is approximately 3 percent of MSR at rated power, and the liquid flows to the WWS or to the condenser for processing in the CPS. With abnormal water chemistry the blowdown rate is 3 percent and the blowdown is discharged to the WWS.

A blowdown sampling line from each SG leads to a secondary water quality monitoring station. Water sampling capabilities for the SG blowdown water are discussed in DCD Subsection 9.3.2, "Process and Post Accident Sampling Systems." NRC staff review of the systems under SRP, Section 9.3.2 is discussed in Section 9.3.2 of this Final SE Report. The SG Blowdown Sampling System provides a means for controlling secondary water chemistry and detecting a SG tube leak or failure. System components include the blowdown lines, blowdown sample lines, sample coolers, pressure reducing valves, radioactive process monitor, instruments, piping, and other valves.

According to Table 1.9.2-5, "US-APWR Conformance with Standard Review Plan Chapter 5 Reactor," in the DCD, Tier 2, the US-APWR secondary water chemistry is based on EPRI Guidelines. The staff considers this an acceptable method of meeting secondary water chemistry requirements, as discussed in BTP MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," (the latest revision of which is called BTP 5-1). However, DCD Table 1.9.2-5 notes some differences between the EPRI Guidelines and the US-APWR requirements. This includes the sampling requirements for SG blowdown water. The details of this issue were reviewed by the staff mainly under SRP Section 10.4.6, "Condensate Cleanup System," and that review is discussed in the corresponding section of the staff's safety evaluation.

In its response to RAI 251-2146, Question 10.4.8-4, dated April 1, 2009, the applicant revised DCD Section 10.4.8.3, "Safety Evaluation," to clarify a statement relating secondary water chemistry control to SG tube integrity. The original statement implied tube integrity could be ensured through secondary water chemistry control. Since there is potential degradation mechanisms unrelated to secondary-side chemistry, such as wear, the revised statement indicates more realistically that secondary water chemistry control is an important contributor to tube integrity.

Conformance to GDC 13 requires the SGBS design to include provisions to monitor the system parameters and maintain them within a range that allows the system to perform its impurity removal function and thereby assist in maintaining the integrity of the reactor coolant boundary. DCD Tier 2, Section 10.4.8.5, "Instrumentation Applications," states that pressure, flow, temperature, and radiation instrumentation both monitor and control system operation, and it refers to FSAR Tier 2, Section 9.3.2, "Process and Post-Accident Sampling Systems," for a description of how the system monitors the secondary water chemistry. DCD Figures 10.4.8-1, "Steam Generator Blowdown System Piping and Instrumentation Diagram (1/2)," and 10.4.8-2, "Steam Generator Blowdown System Piping and Instrumentation Diagram (2/2)," show the location of these monitors and controls. FSAR Section 10.4.8.3, "Safety Evaluation," refers to FSAR Section 10.3.5, "Water Chemistry," for details on the secondary water chemistry specifications. Table 9.3.2-5, "Steam Generator Blowdown Sampling System," in Section 9.3.2 specifies continuous monitoring of cation conductivity, sodium, chloride, and sulfate in samples from each blowdown line. The sample system also monitors specific conductivity and pH. Sampling points at the demineralizer outlets provide a means for monitoring the system performance and maintaining the specified water chemistry.

The staff's review concludes that the SGBS design meets the requirements of GDC 13 because the instrumentation described is capable of monitoring system parameters and maintaining them within a range that allows the system to perform its impurity removal function. Instrumentation and process controls are provided to control flashing, liquid levels, and process flow through the proper components. Therefore, the instrumentation and controls allow the SGBS to maintain secondary water chemistry in a range protective to the reactor coolant pressure boundary. In addition, the system is designed with radiation monitors for isolating the blowdown lines in the event of a high radiation level, such as from SG tube leakage. The staff's detailed review of I&Cs for the US-APWR is documented in Chapter 7, "Instrumentation and Controls," of this SE.

In order to meet the requirements of GDC 14, the system must be designed to control the secondary water chemistry to support the integrity of the primary coolant pressure boundary. Therefore, the system must be sized to accommodate the blowdown flow rate needed to maintain the specified secondary water chemistry under both normal operation and AOOs. Under normal operating conditions, the APWR blowdown flow rate of 1 percent of the MSR provides adequate assurance that the SG water quality will be within specifications. This blowdown rate is consistent with the design of current United States operating PWRs, and the equipment capacities and system flow rates are comparable to those of operating reactors.

The capability to blow down a SG at up to 3 percent provides additional assurance that under abnormal conditions involving one SG with an unusually high impurity ingress rate the affected SG can be cleaned up rapidly. Under abnormal conditions such as contamination from a condenser leak or condensate polisher failure, protection of the SGs is provided by conformance with the EPRI Secondary Water Chemistry Guidelines. The EPRI Guidelines require that power be reduced or the plant shuts down, if parameters exceed the specified action levels. Therefore, the SGBS is not solely relied upon to protect the integrity of the RCPB under abnormal conditions. The blowdown capacity and purification capabilities of the SGBS are, however, adequate to provide interim control of the SG water chemistry while the plant power level is reduced or the plant is shut down as required by the EPRI secondary water chemistry guidelines. In addition, the design of the SGBS includes provisions to bypass individual components and provisions to route the effluent to the waste management system if the SGBS cannot remove the impurities from the blowdown. Therefore, even if the purification function does not operate properly, the SGBS design prevents an adverse impact on the secondary water quality.

In RAI 251-2146, Question 10.4.8-5, dated March 02, 2009, the staff asked the applicant to describe how the US-APWR cleanup capability is adequate for maintaining the specified secondary water chemistry under conditions of condenser tube leaks and radioactivity from primary-to secondary SG tube leakage. In its response to RAI 251-2146, Question 10.4.8-5, dated April 01, 2009, the applicant revised DCD Section 10.4.8.2.2.5, “Abnormal Operation,” to clarify that the CPS maintains condensate water quality in the case of a condenser tube leak, but the SGBS demineralizers support purification by the CPS. The response also stated that with respect to primary-to-secondary tube leakage, the SGBS is designed to maintain the decontamination factor for radioactive materials in accordance with DCD Chapter 11, “Radioactive Waste Management.” This is acceptable because it clarifies how the applicant determined that the system can assist in maintaining optimum secondary water chemistry during these AOOs.

As stated in SRP Section 10.4.8, “Steam Generator Blowdown System,” in order to comply with GDC 14, temperature limits should not be exceeded for heat sensitive processes. The US-APWR SGBS contains both cation and mixed bed (anion/cation) demineralizers that contain temperature-sensitive resins. According to various sources, including the handbook on water treatment from the United States Department of Energy and data sheets from resin manufacturers, the temperature of typical ion exchange resins should be less than about 140°F to prevent thermal decomposition. In its response to RAI 251-2146, Question 10.4.8-6, dated April 1, 2009, the applicant confirmed that this is the recommended temperature limit for the US-APWR SGBS resin. The APWR blowdown passes through the regenerative and non-regenerative coolers, which reduce the temperature of the blowdown water prior to entering the demineralizer inlet filter. The design includes appropriate sensors and controls so that a high temperature signal at the non-regenerative cooler outlet isolates blowdown flow upstream of the demineralizer inlet filter. The isolated flow is diverted to the wastewater system or condenser. In response to RAI 251-2146, Question 10.04.08-06, the applicant changed DCD Section 10.4.8.5, “Instrumentation Applications,” to identify the setpoint temperature as 130°F. This is acceptable because it is below the 140°F limit. The system is thus designed to prevent the temperature from exceeding the upper limit of the demineralizer resins.

FSAR Tier 1, Section 2.7.1.10, “Steam Generator Blowdown System (SGBDS),” and Tier 2, Section 10.4.8.1, “Design Bases,” discuss the design bases for the SGBS. Conformance to GDC 1 and GDC 2 is based on SGBS components and piping from the connection inside the primary containment up to and including the first isolation valve outside the containment being designed as Seismic Category I and Quality Group B. This is based on conformance with RG 1.29 “Seismic Design Classification” and RG 1.26, “Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants,” as stated in SRP Section 10.4.8. The staff reviewed DCD Tier 1, Section 2.8.7, DCD Tier 2, Table 3.2-2, “Classification of Mechanical and Fluid Systems, Components, and Equipment,” and Figure 10.4.8-1, “Steam Generator Blowdown System Piping and Instrumentation Diagram (1/2),” and Figure 10.4.8-2, “Steam Generator Blowdown System Piping and Instrumentation Diagram (2/2),” which show the piping and components up to and including the first isolation valve outside the containment are designed to ASME Section III, Class 2, and Seismic Category I. This is in accordance with RG 1.26, which lists ASME Section III, Class 2 as the quality standards applied to Quality Group B.

In RAI 251-2146, Question 10.4.8-7, dated March 02, 2009, the staff asked the applicant to discuss how the system is designed to prevent FAC or describe the controls in place to ensure it is included in COL applicants' FAC programs. In its response to RAI 251-2146, Question 10.4.8-7, dated April 1, 2009, the applicant explained that the system is designed to preclude FAC in most locations through the use of low-alloy steel and stainless steel in most portions of the system. In portions less susceptible to FAC, where carbon steel may be used, FAC is addressed through water chemistry control and inclusion in the FAC monitoring program. The FAC monitoring program must be provided by COL applicants according to COL Information Item 10.3(1). The applicant also revised DCD Section 10.4.8.2.3, accordingly. This is acceptable because the design includes FAC prevention, while areas of susceptibility are addressed in a FAC monitoring program reviewed by the staff.

For the portion of the system downstream of the outer containment isolation valves, SRP 10.4.8, "Steam Generator Blowdown System," states that RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" Position C.1.1, specifies the quality group standards. RG 1.143 identifies ASME B31.3 (Process Piping) as the design and construction code for piping and valves. The staff identified a discrepancy in that, DCD Tier 2, Table 3.2.2-1, "Classification of Mechanical and Fluid Systems, Components, and Equipment," indicates piping and valves in this portion of the APWR SGBS will be designed to ASME B31.1 (Power Piping). In its response to RAI 251-2146, Question 10.04.08-07, dated April 01, 2009, the applicant revised DCD, Section 3.2, "Classifications of Structures, Systems, and Components," to make the design consistent with RG 1.143. This includes using ASME B31.3 as the design code for piping and valves downstream of the outer containment isolation valves. These proposed changes to the design classifications will be included in the staff's review of DCD, Section 3.2. Considering all portions of the system as discussed above, the staff therefore determined that the design conforms to the requirements of GDC 1 and GDC 2.

Tier 1

FSAR Tier 1, Section 2.7.1.10, "Steam Generator Blowdown System (SGBDS)," describes the SGBS, including Table 2.7.1.10-1, "Equipment Characteristics," Table 2.7.1.10-2, "Piping Characteristics," and Table 2.7.1.10-3, "ITAAC."

ITAAC

ITAAC for the SGBS are provided in FSAR Tier 1 Table 2.7.1.10-3, "Steam Generator Blowdown System Inspections, Tests, Analyses, and Acceptance Criteria." The ITAAC addresses the functional arrangement of the as-built SGBS, ASME Code Section III requirements, seismic category requirements, and division separation requirements. Therefore, the staff determined that these ITAAC are adequate to ensure future plants will be built in accordance with the DC related to the SGBS.

Technical Specifications

There are no limiting conditions for operation for the SGBS because it does not meet any of the four criteria of 10 CFR 50.36(c)(2)(ii).

Preoperational Testing

DCD Tier 2, Section 14.2.12.1.83, "Steam Generator Blowdown System Preoperational Test," describes the SGBS preoperational testing. The objectives of the testing are to demonstrate the operation of the SGBS sample monitoring system and to demonstrate that the SGBS routes and processes blowdown water correctly. The testing verifies manual and automatic system controls, flow paths, flow rates, temperatures, indications, and alarms. In addition, the test verifies system isolation using simulated signals.

10.4.8.5 Combined License (COL) Information Items

Table 1.8-2 of the DCD contains the following information.

Table 10.4.8-1: US APWR Combined License Information Item

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
10.4(2)	Steam Generator Blowdown System: The Combined License applicant is to address the discharge to the Waste Water System, including site specific requirements.	10.4.12	Y	N

10.4.8.6 Conclusions

Pending resolution of Open Item 10.04.06-5 on adherence to industry guidelines for secondary water chemistry, the staff will conclude that the SGBS satisfies the requirements of GDC 1, GDC 2, GDC 13, and GDC 14, and the guidelines of SRP Section 10.4.8. This conclusion is based on the SGBS design being adequate to control the concentration of chemical impurities and radioactive materials in the secondary coolant. In addition, the SGBS design meets the primary boundary material integrity requirements of GDC 13 and GDC 14 related to monitoring and maintaining acceptable secondary water chemistry, respectively. Also, the SGBS meets the quality standard requirements of GDC 1 and the seismic requirements of GDC 2.

10.4.9 Emergency Feedwater System (EFWS)

10.4.9.1 Introduction

The EFWS supplies feedwater to the SGs when the main feedwater system is not in operation for transient conditions or postulated accidents. It is designed to remove reactor core decay heat and reactor coolant sensible heat through the SGs following transient conditions or postulated accidents such as a reactor trip, loss of main feedwater, main steam line breaks, or FLBs, LOOP, small break loss-of-coolant accident (LOCA), SBO, anticipated transient without scram (ATWS), and SG tube rupture (SGTR). It is a safety-related system.

10.4.9.2 Summary of Application

Tier 1: The Tier 1 information associated with Section 10.4.9, “Emergency Feedwater System,” is found in Tier 1, Section 2.7.1.11, “Emergency Feedwater System (EFWS),” of the US-APWR DCD. Figure 2.7.1.11-1, “Emergency Feedwater System,” illustrates the arrangement of the EFWS components. Table 2.7.1.11-1 provides a tabulation of the location of EFWS equipment.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.4.9 of the US-APWR DCD, summarized here in part, as follows:

- The EFWS consists of two motor-driven EFW pumps, two steam-driven EFW pumps, two EFW pits, piping, valves, and associated instrumentation. The EFWS flow diagram is shown in Figures 10.4.9-1, “Emergency Feedwater System Piping and Instrumentation Diagram (1/2),” and 10.4.9-2, “Emergency Feedwater System Piping and Instrumentation Diagram (2/2).” Each EFW pump takes suction from one of two EFW pits and the discharge is directed to one of the four SGs. Both EFW pits together contain the minimum water volume required for maintaining the plant at hot standby condition for eight hours and performing plant cooldown for six hours until the RHR system can start to operate.

- Normally open, motor-operated EFW control valves are provided in the EFW pump discharge lines to each SG for controlling EFW flow. Two normally open, motor-operated isolation valves are provided in the EFW lines routed from the EFW pump to each SG for isolation of the system from a faulty SG. One steam isolation valve is provided in each line from the SG that provides steam to the EFW pump turbine. This is a containment isolation valve. One normally closed valve is provided for each EFW pump turbine in the common line that provides steam to the EFW pump turbine. The design parameters of the EFW components are provided in Table 10.4.9-1.

ITAAC: The ITAAC associated with Tier 2 Section 10.4.9 are given in Tier 1 Section 2.7.1.11.2, “Inspections, Tests, Analyses, and Acceptance Criteria,” of the US-APWR DCD.

TSs: The TSs associated with Tier 2 Section 10.4.9 are given in Tier 2 Chapter 16, Section 3.7.5 of the US-APWR DCD for the four EFW trains, and Chapter 16, Section 3.7.6 of the US-APWR DCD for the EFW pits.

10.4.9.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 10.4.9, “Auxiliary Feedwater System (PWR),” of NUREG-0800, the SRP, and are summarized below. Review interfaces with other SRP sections can be found in Section 10.4.9.I of NUREG-0800.

1. GDC 2, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes.

2. GDC 4, with respect to the capability of the system and the structure housing the system to withstand the effects of pipe breaks and external missiles.

3. GDC 5, with respect to the capability of shared systems and components important to safety to perform required safety functions.

4. GDC 19, “Control Room,” with respect to the design capability of system I&Cs for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.

5. GDC 34, "Residual Heat Removal Systems," and GDC 44, "Cooling Water," with respect to the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions, assuming any single active failure, coincident with the LOOP for certain events, and the capability to isolate components, subsystems, or piping if required to maintain system safety function.

6. GDC 45, as it relates to design provisions made to permit periodic ISI of system components and equipment.

7. GDC 46, as it relates to design provisions made to permit appropriate functional testing of the system and components.

8. GDC 60 as it relates to the GSS design for the control of releases of radioactive materials to the environment.

9. 10 CFR 50.62, as it relates to the design provisions for automatic initiation of the EFWS in an ATWS event.

10. 10 CFR 50.63, as it relates to the design provisions for withstanding and recovering from a SBO.

11. 10 CFR 52.47(b)(1), as it relates to requirement that a DC application address the proposed ITAAC. Acceptance for meeting this criterion is based on the provision of information describing the proposed ITAAC, including appropriate design bases. Acceptance criteria are identified and addressed as appropriate in accordance with Section 14.3 of NUREG-0800.

10.4.9.4 Technical Evaluation

The staff reviewed the EFWS and EFW pits in accordance with the review procedures in Standard Review Plan (SRP) Section 10.4.9, "Auxiliary Feedwater System (PWR)," Revision 3, for the EFWS, and SRP Section 9.2.6, "Condensate Storage Facilities," Revision 3, for the EFW pits. Conformance with the acceptance criteria of SRP Section 10.4.9, and SRP Section 9.2.6, formed the basis for the evaluation of the EFWS, and its associated EFW pits with respect to the applicable regulations. The results of the staff's review are provided below. The evaluation addresses compliance with the SRP acceptance criteria listed in Section 10.4.9.3 above.

The staff's evaluation of the EFWS is based on Revision 2 of the DCD. All references made in this section to the DCD or individual DCD sections represent Revision 2 of the DCD.

The EFWS is a safety-related system designed to remove reactor core decay heat and RCS sensible heat through the SGs following transient conditions or postulated accidents, including the following: reactor trip; LOOP; loss of main feedwater; small break LOCA; FLB; MSLB; SBO; ATWS; and SGTR. It consists of two motor-driven pumps, two steam turbine-driven pumps, and two EFW pits, along with associated piping, valves and instrumentation. The two EFW pits are the preferred source of water for the EFWS and serve as the safety-related water supply source for the EFWS.

Descriptions of the EFWS are provided in the DCD Tier 1, Section 2.7.1.11, "Emergency Feedwater System:" DCD Tier 2, Section 1.2.1.5.3.4, "Emergency Feedwater System:" DCD Tier 2, Section 10.4.9, "Emergency Feedwater System:" DCD Tier 2 Chapter 16, "Technical Specifications:" Section 3.7.5, "Emergency Feedwater (EFW) System:" and Section 3.7.6, "Emergency Feedwater Pit (EFW Pit):" and Bases 3.7.5 and 3.7.6. The functional arrangement of the system is provided in DCD Tier 1 Figure 2.7.1.11-1, "Emergency Feedwater System Location of Equipment and Piping:" and DCD Tier 2, Figure 3E-8, "Emergency Feed Water System Flow Diagram (1/2);" Figure 3E-9, "Emergency Feedwater System Flow Diagram (2/2);" Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumental Diagram (1/2);" and Figure 10.4.9-2, "Emergency Feedwater System Piping and Instrumental Diagram (2/2)."

The EFWS is designed to supply water to the SGs whenever the reactor coolant temperature is above 177°C (350°F) and the main feedwater system is not in operation. The EFWS removes reactor core decay heat and RCS sensible heat through the SGs during design basis transient and accident conditions. It is comprised of four 50 percent capacity pumps, and each pump takes suction from the EFW pits and supply feedwater to only one SG. Each of the four EFWS pumps is sized to supply the feedwater flow required for removal of 50 percent of the reactor decay heat. The EFWS is designed with an emergency water supply capacity sufficient to remove decay heat and to provide feedwater for cooldown of the RCS at an average temperature of approximately 27.78°C (50°F) per hour for a period of 14 hours. This 14-hour period allows for 8 hours at hot standby followed by a 6-hour cooldown of the primary system at an average rate of approximately 27.78°C (50°F) per hour.

The EFWS provides several important safety-related functions:

- Flow to the SGs for cooldown and depressurization of the RCS from a condition of full power to an RCS temperature at which the residual heat removal system (RHRS) may be placed in operation

- Isolation of the EFWS flow to the affected SG following a MSLB to prevent overcooling the RCS

- Isolation of EFWS pump flow to the SG with a tube rupture upon water level increase in the faulted SG to prevent SG over-fill and mitigate potential radiological consequences.

<u>10.4.9.4.1 System Design Considerations</u>

A. GDC 2, "Design Bases for Protection Against Natural Phenomena"

The staff reviewed the EFWS (including the EFW pits) for compliance with the requirements of GDC 2 with respect to its design for protection against the effect of natural phenomena such as earthquakes, tornados, hurricanes and floods. Compliance with the requirements of GDC 2 is based on adherence to Position C.1 of RG 1.29, "Seismic Design Classification," for the safety-related portions of the system, and Position C.2 for non-safety-related portions of the system.

The location of the EFWS components and equipment is given in DCD Tier 2 Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," and Table 3D-2, "Design of Structures, Systems, Components, and Equipment." As stated in DCD Tier 2, Section 10.4.9.2 the EFWS components are located in the reactor building and therefore safety-related equipment items associated with the EFWS are located inside the reactor building. The reactor building is a seismic Category I structure that is designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other natural phenomena. The basis for the adequacy of the design for the reactor building is provided in DCD, Tier 2 Sections 3.2, 3.3, 3.4, 3.5 and 3.8, which are reviewed in the corresponding sections of this SE.

In its review of the EFWS the staff found that Sheet 1 of DCD Tier 1, Table 2.7.1.11-1, "Emergency Feedwater System Location of Equipment and Piping ," indicated that the "A-emergency feedwater isolation valve" was located inside containment, while the corresponding "B," "C," and "D" train valves were located in the reactor building. Also, in DCD Tier 2 Section 10.4.9.1, "Design Basis," the third bulleted item referenced "buildings where the EFWS components are located," thus implying that EFWS components may be located in more than one building. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-1, dated January 1, 2009, that the applicant address these discrepancies regarding the location of EFWS components.

In its response to RAI 160-1848, Question 10.4.9-1, dated February 20, 2009, the applicant stated that all of the EFWS components are located inside the reactor building. The applicant proposes to revise the DCD to address the discrepancies identified in RAI 160-1848, Question 10.4.9-1. Specifically, DCD Tier 1 Table 2.7.1.11-1 will be modified so that the location of the "A-emergency feedwater isolation valve" is listed as the "reactor building." Also, in the third bulleted item in DCD Tier 2 Section 10.4.9.1, "Design Basis," the word "buildings" will be changed to "building." The staff has confirmed that Revision 2 of DCD (dated October 2009) Tier 1, Table 2.7.1.11-1, and Tier 2, Section 14.9.1 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue since the application now correctly identifies the location of the EFWS components. Therefore, the staff considers RAI 160-1848 Question 10.4.9-1 to be resolved.

As indicated in DCD Tier 2, Section 3.2.1.1.3, "Non Seismic," non-seismically rated SSCs are located outside of safety-related buildings or segregated from seismic Category I SSCs so that the failure of their structural integrity would not impact the seismic Category I SSCs and cause adverse system interactions. The applicant stated that should it be determined that the potential exists for a non-seismically rated SSC to adversely impact a seismic Category I SSC, the non-seismically rated SSC is designed to seismic Category II criteria so that it could not fail during a SSE and cause an adverse impact or interaction with the seismic Category I SSC.

In addition to the building housing the EFWS components (reactor building), the staff also reviewed classifications of the EFWS components and equipment provided in DCD, Tier 2, Table 3.2-2, "System Quality Group Classification," and found them to be appropriately classified in accordance with RG 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive waste Containing Components of Nuclear Power Plants." The EFWS containment isolation valves are classified as Quality Group B, which corresponds to seismic Class 1 and ASME III Class 2 (part of the RCS pressure boundary). Except for flow, pressure, and temperature transmitters, and piping and non-safety valves associated with the turbine steam drain pots, the remaining EFWS equipment items are classified as Quality Group C, which corresponds to seismic Class 1 and ASME III Class 3 (safety-related). Transmitters for EFW flow, EFWS pit water level, and pump discharge pressure are designed to seismic Class 1 (a "Quality Grouping" assignment does not apply to these components). The basis for the EFWS equipment classification is provided in DCD, Tier 1 Table 2.7.1.11-2, "Emergency Feedwater System Equipment Characteristics," DCD Tier 2 Table 3.2-2, DCD Tier 1, Figure 2.7.1.11-1, and DCD Tier 2 Figures 10.4.9-1 and 10.4.9-2. With the exception of the EFW pit breather lines (see below), the staff found the EFWS system piping and components to be designed to quality standards commensurate with the importance of its safety functions.

Some United States PWRs use a CST as the preferred source of water for the EFWS pumps. For the US-APWR design, there is no CST source of water for the EFWS pumps. Instead, two EFW pits ("A" and "B") serve as the water supply source. The EFW pits are connected by a tie line with two normally closed manual valves. Pit "A" is normally aligned to feed EFWS pumps "A" and "B", while pit "B" is normally aligned to feed EFWS pumps "C" and "D." The EFW pits are located in the reactor building, as shown in DCD Tier 2, Figure 1.2-9, "Power Block at Elevation 76'-5" - Plan View." As indicated in DCD Tier 2, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components and Equipment," the pits are designed as seismic Category I.

The staff noted that the seismic categorization of the EFW pit breather lines (vent lines) was not explicitly identified in the DCD. Therefore, the staff requested in RAI 160-1848, Question 10.4.9-2, dated January 21, 2009, that the applicant identify and justify the seismic category associated with the EFW pit breather lines.

In its response to RAI 160-1848, Question 10.4.9-2, dated February 20, 2009, the applicant stated that the EFW pit breather lines are designed to seismic Category 1. In DCD Tier 2 Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumentation Diagram (1/2)," there are no equipment classification boundary symbols between the EFW pits and breather lines. Thus, the EFW pits and breather lines share the same equipment classification. Given that the EFW pits are seismic Category 1, the EFW pit breather lines are also seismic Category 1. The staff finds that the concerns identified in RAI 160-1848, Question 10.4.9-2 are resolved since the applicant adequately clarified the seismic categorization of the EFW pit breather lines.

The safety-related EFWS and EFWS pit components are designed in accordance with RG 1.29 Position C.1. In addition, the non-safety components are designed in conformance with RG 1.29 Position C.2. Based on the above review, the staff concludes that the EFWS design conforms to the guidelines of Positions C.1 and C.2 of RG 1.29 and the requirements of GDC 2, as they relate to protecting the system against phenomena.

B. GDC 4, "Environmental and Dynamic Effects Design Bases"

The staff reviewed the EFWS for compliance with the requirements of GDC 4 with respect to the capability of the system and the structures housing the system to withstand the effects of pipe breaks and internally and externally generated missiles, and pipe whip and jet impingement due to high and moderate energy pipe breaks. Compliance with the requirements of GDC 4 is based on identification of the essential portions of the system as protected from dynamic effects including internal and external missiles and meeting the guidance in BTP 10-2, "Design Guidelines to Avoid Water Hammer in Steam Generators."

In the design, the EFWS, and EFW Pits are located inside the reactor building, which is a seismic Category I structure designed to provide protection against tornadoes, floods, missiles, and other natural phenomena. In accordance with DCD Tier 1 Table 2.7.1.11-5, "Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria," each of the EFWS mechanical divisions is physically separated by structural and/or fire barriers, except for the EFW pump suction tie line and discharge tie lines. Isolation valves are installed in the tie lines to help assure that tie line rupture would not lead to the unavailability of the entire system. In accordance with DCD Tier 2 Section 10.4.9.3, "Safety Evaluation," each EFW pump is located in a separate compartment surrounded by structural concrete walls. Complete physical and electrical separation is maintained for the pump controls, control signals, electrical power supplies, and instrumentation for each EFW pump.

Section 10.4.9.3 of the Tier 2 DCD states that "…safety-related portions of the EFWS are protected from missiles as described in Section 3.5,…." and "…against dynamic effects associated with the postulated rupture of piping as described in Section 3.6." The staff did find, in DCD Tier 1 Table 2.2-4, "Inspection, Tests, Analyses, and Acceptance Criteria," ITAAC Item # 17, that the applicant made the following commitment with regard to pipe breaks: "Safety-related SSCs are designed to withstand the dynamic effects of pipe breaks." Section 3.6.1 of this SE addresses the staff's acceptance of this position with regard to the dynamic effects from pipe breaks. The staff also noted that COL Information Item 3.6(1) is used to verify the consequences of failures of site-specific piping with regard to plant shutdown. However, based on the review of the information in these two DCD sections, the staff could not find sufficient information in regard to the provisions and plant design features to ensure adequate protection against the effects of pipe breaks and internally and externally generated missiles, and pipe whip and jet impingement due to high and moderate energy pipe breaks. The staff was unable to find a corresponding ITAAC that would ensure missile protection for the EFWS. Therefore, the staff requested in RAI 160-1848, Question 10.4.9-3, dated January 21, 2009, that the applicant describe the provisions and design features used to ensure adequate protection from missiles and compliance with GDC 4.

In its response to RAI 160-1848, Question 10.4.9-3, dated February 20, 2009, the applicant stated that changes will be made to DCD Tier 2, Section 3.5. This revised DCD section will reference RG 1.117, "Tornado Design Classification," and will list SSCs to be protected from postulated missiles, consistent with Appendix A of RG 1.117. This list of SSCs to be protected includes EFW systems. The revised DCD section will also include a summary of potential missile sources, the probabilistic analysis methodology used to statistically evaluate missile sources, and the protection features for preventing the generation of missiles and/or protecting potentially targeted SSCs.

The staff has confirmed that Revision 2 DCD Tier 2, Section 3.5 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue since the application now clearly identifies the EFW system SSCs that are protected from postulated missiles. Therefore, the staff considers RAI 160-1848, Question 10.4.9-3 resolved.

The staff reviewed the EFWS against the guidance in BTP 10-2, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," and NRC Generic Letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." These three documents contain design guidelines and recommendations to reduce, or eliminate piping damage caused by water hammer transients. The discussion of design provisions to mitigate water hammer is presented in DCD Tier 2, Section 5.4.2.1.2.11, "Water Hammer at Feedwater Ring," Section 10.4.7.2.2, Section 10.4.7.3, "Safety Evaluation," and Section 10.4.7.7, "Water Hammer Prevention," for the main feedwater system. The DCD does not separately list design guidelines and recommendations to address water hammer for the EFWS. Instead, for the EFWS, the DCD refers back to the water hammer discussions for the main feedwater discussions that are provided in DCD Tier 2, Section 10.4.7.7. In DCD Tier 2 Section 14.2, "Initial Plant Test Program," the applicant includes instructions for the COL holder to check for water hammer during normal system startup and operation conditions during motor-driven EFWS preoperational testing (14.2.12.1.24) and during turbine-driven EFWS preoperational testing (14.2.12.1.25). The COL holder is also instructed to check for unacceptable water hammer during restoration of normal SG level from low water level as part of feedwater preoperational testing (14.2.12.1.29).

Based on the staff's review, the above test provisions described in DCD Tier 2, Section 14.2 are considered appropriate for minimizing water hammer events related to the EFWS. However, there was no discussion in the DCD as to what extent the design measures discussed for the main feedwater system, apply to the EFWS, and what measure of applicability the feedwater prevention measures have for the EFWS. Also there was no information presented in the DCD that will ensure development of operating and maintenance procedures by the COL applicant that will minimize the potential for water hammer in the EFWS during operation. In addition, the DCD did not mention that the EFWS piping network should be sufficiently filled with water while on standby during plant operation to prevent excessive gas accumulation. As discussed in GL 2008-01, gas accumulation can in some circumstances result in water hammer. The staff requested in **RAI 160-1848, Question 10.4.9-4**, dated January 21, 2009, that the applicant provide an explanation of how the procedure requirements to prevent or minimize water hammer will be included in the DCD, and how the lines will be maintained water-solid.

In its response to RAI 160-1848, Question 10.4.9-4, dated June 1, 2009, the applicant proposed adding a new COL Information Item 10.4(6) requiring COL applicants to develop operating and maintenance procedures for the EFWS that address minimization of potential water hammer. The applicant also proposed adding to the DCD, a description of water hammer prevention and mitigation measures for the EFWS, and a list of the key elements to be included in the EFWS operating and maintenance procedures related to water hammer prevention.

In its response to RAI 160-1848, Question 10.4.9-4, the applicant also stated that DCD Tier 2 Section 10.4.9.3 will be revised to describe a restoration procedure for addressing situations where bypass leakage through EFW check valves is detected. This restoration procedure will require the isolated area to be filled with water prior to returning to service. The applicant further indicated how the TSs require restoration of a train outage be completed within 72 hours. The staff finds that the applicant's restoration procedure is acceptable. The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9 was revised as committed in the RAI response. The staff reviewed the COL Information Item 10.4(6) and the revision to DCD Section 10.4.9 and found that with the addition of the new COL item and the revision of the DCD text will ensure that the applicant will develop procedures to prevent or minimize water hammer. Therefore, the staff considers RAI 160-1848, Question 10.4.9-4, to be resolved.

The flood protection and protection against pipe whip and jet impingement due to high and moderate energy pipe breaks are evaluated in Section 3.4.1, "Flood Protection," and Section 3.6.1, "Plant Design for Protection against Postulated Piping Failure in Fluid Systems Inside and Outside Containment," of this SE. The general review of protection against the effects of internal and external missiles is evaluated in Section 3.5.1.1, Section 3.5.1.4, and Section 3.6.1 of this SE.

Based on the above review, the staff finds that the applicant has identified the essential portions of the EFW system as protected from dynamic effects including internal and external missiles. The staff also considers the plant design and test provisions to be appropriate for minimizing water hammer events, because they conform to the guidance in BTP 10-2, GL-2008-01, and NUREG-0927. The staff concludes that the EFWS and EFW pits meet the requirements of GDC 4 with respect to the environmental and dynamic effects design basis.

C. GDC 5, "Sharing of Structures, Systems, and Components"

The staff reviewed the EFWS design for compliance with the requirements of GDC 5 with respect to sharing of SSCs. Acceptance is based on the failure of any component including a pipe break and single active failure not preventing the safe shutdown and cool down of either unit (together or singularly). As stated in DCD Tier 2, Section 3.1.1.5, “Criterion 5 - Sharing of Structures, Systems, and Components,” the US-APWR is a single plant, and does not share safety-related SSCs with other units or plants. Thus, the requirement of GDC 5 for sharing systems between units does not apply.

D. GDC 19, “Control Room”

The staff reviewed the EFWS design for compliance with the requirements of GDC 19 as related to the design capability of system I&Cs for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown. An applicant may demonstrate compliance with the requirements of GDC 19 by meeting BTP 5-4, “Design Requirements of the Residual Heat Removal System,” with regards to cold shutdown from the control room using only safety grade equipment.

The US-APWR has a MCR, from which actions can be taken to operate the plant under normal conditions and to maintain it in a safe manner under accident conditions, including LOCAs. In Tier 2, Section 10.4.9.2 of the US-APWR DCD, the applicant states that the status of active components during system operation is displayed for the operator in the MCR and at the remote shutdown console. A more detailed discussion of safety-related display instrumentation related to the EFWS is included in Tier 2, DCD Section 7.5, “Information Systems Important to Safety.” The applicant further states in Tier 2, DCD Section 10.4.9.2, “System Description,” that information indicative of the readiness of the EFWS prior to operation and the status of active components during a system operation is displayed for the operator in the MCR and at the remote shutdown console. DCD, Tier 2 Section 10.4.9.1, “Design Bases,” states that the EFWS has the capability to permit operation at hot standby for eight hours followed by six hours of cooldown to the RHR cut-in temperature from the MCR using only safety related equipment with a single active failure. The indication and controls provided for the EFWS are summarized in Table 10.4.9-5, “Emergency Feedwater System Failure Modes and Effects Analysis (FMEA).” Based on the above described capability to monitor and control the EFWS from the MCR using only safety-related equipment, the staff finds that the plant design provides an acceptable means of compliance with GDC 19 and BTP 5-4.

E. GDC 34, “Residual Heat Removal,” and GDC 44, “Cooling Water”

The staff reviewed the EFWS for compliance with the requirements of GDC 34 and GDC 44, with respect to the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions, assuming any single active failure, coincident with the LOOP for certain events, and the capability to isolate components, subsystems, or piping if required to maintain system safety function. To demonstrate compliance with GDC 34 and GDC 44, SRP Section 10.4.9 states, in part, that the system design should conform to the guidance of BTP 10-1, “Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants,” as it relates to EFWS pump drive and power supply diversity.

The staff reviewed the EFWS for compliance with the requirements of BTP 10-1, as related to EFW pump drive and power supply diversity. Guideline B.1 in BTP 10-1 states that the EFWS should have at least two full-capacity, independent systems with diverse power sources.

The US-APWR EFWS design provides diversity by using two types of pump drives (electric motors and steam turbines). It is stated in DCD Tier 2, Section 10.4.9.2, “System Description,” that the EFWS has two motor-driven and two turbine-driven EFW pumps, with different power supplies. The two motor-driven EFW pumps connect to each different safety ac bus to achieve the specific safety function in case of off-site power loss, and each bus is backed by a redundant emergency power source. DCD Tier 2, Table 10.4.9-6 identifies the specific power sources for EFWS components. The turbine driven EFW pumps are connected to main steam lines from two SGs. Steam supplying drivers to the A-EFW pump is taken from main steam line A and B, and steam supplying drivers to the D-EFW pump taken from main steam lines C and D. Based on a review of the EFW design the staff finds that the guidance in of BTP 10-1, as related to EFW pump drive and power diversity, is satisfied

The staff reviewed the EFWS for its capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions, assuming any single active failure and its ability to maintain required system safety function.

DCD, Tier 2, Section 10.4.9.3 states that the EFWS and supporting systems are designed to provide the required flow to the SGs with a LOOP, assuming a single active component failure in one train and a maintenance outage of one active component in another train during on-line maintenance.

The essential portions of the system consist of four redundant 50 percent trains feeding the four SGs. The EFW pumps automatically start on receipt of LOOP signal, Emergency Core Cooling System (ECCS) actuation signal, main feedwater pumps trip (all pumps) signal, or low SG water level signal in any one of SGs. Furthermore, the motor-operated, EFW feedwater control and isolation valves are normally-open and verified to be in their fully open position at startup of the EFW pump on receipt of an open check signal such as LOOP signal, ECCS actuation signal, main feedwater pumps trip (all pumps) signal, or low SG water level signal in any SG. The two normally closed, motor-operated EFW turbine pump actuation valves automatically open upon receiving an EFW pump actuation signal. In accordance with Tier 2, Section 10.4.9.1, “Emergency Feedwater System Piping and Instrumentation Diagram (1/2),” and Table 10.4.9-5, “Emergency Feedwater System Summary of Indication and Controls,” the system is also capable of manual actuation from the control room.

The EFW pump is designed to develop adequate head to supply the design flow of at least 1514 L/min (400 gpm) to each SG, when the SG pressure is equivalent to the set pressure of the first stage of the main steam safety valve (safety valve with lowest set pressure) plus 3 percent of accumulation. As stated in DCD, Tier 2, Section 10.4.9, the EFW is designed with two 50 percent EFW pits and provides sufficient heat removal capability for EFWS operation for 8 hours at hot standby, followed by 6-hour cooldown of the primary system at an average rate of approximately 27.78°C (50°F) per hour. The EFW pump capacity is based on providing sufficient feedwater supply to prevent the reactor coolant discharge from the pressurizer safety valve even when only two EFW pumps and two SGs are available due to single failure of one EFW pump and one SG failure. The EFW pump capacity satisfies the required feedwater flow to the SG to prevent the reactor coolant release from the pressurizer safety valve with loss of the main feedwater due to a main FLB. The pump head is sufficient to establish the minimum necessary flow rate against the SG pressure corresponding to the first stage main steam safety valve set pressure plus 3 percent accumulation pressure. With the low-low water level in the EFW pits, the available net positive suction head (NPSH) to the motor-driven (M/D) EFW pumps is 29.57 meters (97 feet), while the required NPSH at maximum operating flow is 22.25 meters (73 feet) providing adequate margin. The available NPSH to the turbine-driven (T/D) EFW pumps is 30.48 meters (100 feet), while the required NPSH at maximum operating flow is 23.17

meters (76 feet) providing sufficient margin. Confirmation that the as-built plant has sufficient NPSH for the EFW pumps is accomplished with ITAAC 14 in DCD Tier 1; Table 2.7.1.11-5, “Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria.”

DCD Tier 2 Section 10.4.9.2.2, Item A (b), “Normal Plant Operation,” the second paragraph, stated the following: “The manual valves in the suction line flow paths from the EFW pits to the M/D and T/D EFW pumps are normally closed.” However, in Figure 10.4.9-1, “Emergency Feedwater System Piping and Instrumentation Diagram (1/2),” it appeared that these pump suction valves are normally open. If the suction valves are normally closed, it is not clear that the EFWS can operate in a timely manner to provide heat removal given that local operator action would be required to open the valves prior to establishing injection flow from the EFWS. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-5 that the applicant address this item.

In its response to RAI 60-1848, Question 10.4.9-5, dated February 20, 2009, the applicant stated that the manual valves in the pump suction line flow paths are normally open. The applicant proposed to revise DCD Tier 2, Section 10.4.9.2.2, “System Operation,” to state that these valves are normally open. The staff has confirmed that Revision 2 DCD Tier 2, Section 3.5 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue. Therefore, the staff considers RAI 160-1848, Question 10.4.9-5 to be resolved, since the applicant now clearly identifies the suction line flow valves as normally open and local operator action is not required to establish injection flow from the EFWS.

The EFWS can be operated for approximately 14 hours with the water 1.55 x 10^6 L (409,700 usable gallons) in the two EFW pits. The combined volume of water in the two EFW pits exceeds the minimum water volume required for maintaining the plant at hot standby condition for eight hours and performing plant cooldown for six hours until the RHRS can start to operate. In DCD, Tier 2, Section 10.4.9.3, “Safety Evaluation,” the total required EFW volume needed to accommodate the decay heat removal requirements for 8 hours of operation at hot standby followed by 6 hours cooldown is given as 1.41 x 10^6 L (372,000 gallons), or 7.05 x 10^5 L (186,200 gallons) per pit. The useable volume of water is specified as 7.75 x 10^5 L (204,850 gallons) per pit in DCD, Tier 2 Section 10.4.9.3, and verification that each EFW pits contains a minimum volume of 7.75 x 10^5 L (204,850 gallons) is required by TS surveillance requirement (SR) 3.7.6.1. The demineralized water storage tank (DWST) provides a non-safety-related backup water source for the EFWS. In accordance with DCD Tier 2 Table 9.2.6-1, the DWST has a capacity of 1.89 x 10^6 L (500,000 gallons).

Based on the staffs review as detailed above, the staff finds that the EFWS and EFW pits to be capable of transferring heat loads from the reactor system to a heat sink under both normal operating and accident conditions assuming single active failure and thus satisfies the requirements of GDC 34 and GDC 44.

The EFWS satisfies the recommendations of RG 1.62, “Manual Initiation of Protective Actions,” regarding the capability of manual initiation of protective actions. As documented in DCD Tier 2 Sections 10.4.9-2, Table 10.4.9-5, Section 7.3.1.5.9, Section 7.3.1.5.10, and Table 7.3-5, the plant design provides means to manually actuate and control EFW pumps and motor-operated EFW isolation and control valves.

In accordance with SRP Section 9.2.6, Section III, Item 1.C, the applicant should discuss methods to protect the purity and cleanliness of the EFW pit inventory. Methods might include pit coatings, covers, and other passive components. The DCD did not appear to describe methods used to protect the purity and cleanliness of the EFW pit inventory. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-6, dated January 21, 2009, that the applicant address this topic.

In its response to RAI 160-1848, Question 10.4.9-6, dated February 20, 2009, the applicant stated that the EFW pits are completely enclosed structures. Stainless steel plates are used to line the interior surfaces of the pits. Water supplied to the pits is clean, demineralized water from the DWST. The applicant further stated that sampling of EFW pit inventory is performed during each “regular inspection” to ensure turbidity does not exceed one ppm. “Feed and bleed” drainage and replenishment of the pit inventory would be used to correct deviations from the acceptable turbidity threshold. The applicant proposed to add this information to DCD Tier 2, Section 10.4.9.2 “System Description,” though without mention of the stainless steel liner. Also, the applicant did not specify the time interval between each “regular inspection.”

The staff finds that the applicant’s response to RAI 160-1848, Question 10.4.9-6 is acceptable, except that the stainless steel liner should be mentioned in the revised DCD. Also, the applicant should specify the time interval between regular inspections and include this interval in the revised DCD. Accordingly, the staff requested in RAI 408-3170, Question 10.4.9-24, dated June 24, 2009, that the applicant also include these information items in the revised Tier 2 DCD.

In its response to RAI 408-3170, Question 10.4.9-24,dated July 28, 2009, the applicant stated that DCD Tier 2, Section 10.4.9.2.1 D will be revised to state that the EFW pits, including their linings, are constructed of stainless steel. The revised DCD text will further state that monthly sampling of the EFW pits will be performed, with periodic inspections of the pits (after they are drained) performed in conjunction with the ISI program. The staff reviewed the proposed revisions to DCD Tier 2, Section 10.4.9.2.1 D and found that with the revisions of the DCD text, the concerns raised by the staff in RAI 408-3170, Question 10.4.9-24 were resolved. The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed the concerns raised by the staff in RAI 408-3170, Question 10.4.9-24 and therefore, the staff considers RAI 160-1848, Question 10.4.9-6 and RAI 408-3170, Question 10.4.9-24 to be resolved, because the application now clearly discusses the methods to protect the purity and cleanliness of the EFW pit inventory.

The staff reviewed the EFWS with regard to the generic recommendations of NUREG-0611, “Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plants,” and NUREG-0635, “Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering – Designed Operating Plants.” The staff found that the design to be consistent with the recommendations except as noted below:

<u>Generic Short Term Recommendation No. 3 (GS-3)</u>: GS-3 recommends that measures be taken to eliminate or reduce the potential for water hammer from EFWS discharge. A review of measures taken to eliminate or reduce the potential for water hammer was discussed above in this section of the SE, with regard to the applicant’s compliance with the requirements of GDC 4. As part of the review associated with GDC 4, the staff requested in RAI 160-1848, Question 10.4.9-4, dated January 21, 2009, that the applicant provide an explanation of how the procedure requirements to prevent or minimize water hammer will be included in the DCD, and how the lines will be maintained water-solid. The review of RAI 160-1848, Question 10.4.9-4 is discussed in Section 10.4.9.4.1B of this SE. Based on the addition of COL Information Item

10.4(6) to require plant operating and maintenance procedures, address water hammer and the revision to DCD Section 10.4.9, the concerned raised by this RAI was resolved.

Generic Short Term Recommendation No. 4 (GS-4): GS-4 recommends emergency procedures to be available for transferring to alternate sources of EFW supply. In accordance with DCD Tier 2, Figure 10.4.9-1, “Emergency Feedwater System Piping and Instrumentation Diagram (1/2),” and DCD Tier 2, Chapter 16, the DWST provides a direct backup source for EFWS. If the water level of the EFW pit reached low-low level, operators are given an alarm in the MCR. The EFW pumps subsequently will be stopped or the water source will be switched to DWST manually to keep the sufficient EFW flow, if necessary. DCD Tier 2, Section 13.5.3, “Plant Procedures,” states that the COL applicant is to describe the program for developing and implementing emergency operating procedures by means of COL Information Item 13.5(6). However, the staff could not find a specific commitment that the COL applicant would develop emergency procedures that specifically address the switchover of water to the DWST. Accordingly, the staff requested, in RAI 160-1848, Question 10.4.9-7, dated January 21, 2009, that the applicant demonstrate how it will be assured that emergency procedures will be developed for switchover of water to the DWST. In accordance with DCD Tier 2 Table 9.2.6-1, the DWST has a capacity of 1.89×10^6 L (500,000 gallons).

In its response to RAI 160-1848, Question 10.4.9-7, dated February 20, 2009,the applicant proposed to revise DCD Tier 2, Section 10.4.9.2.1 to indicate that switchover to the DWST will be performed following an accident or transient if (a) the reactor has not been stabilized at Mode 4 (hot shutdown) and (b) both EFW pits have reached low-low level. The revised DCD will further indicate that switchover will be accomplished by means of operator actions to open manual valves from the DWST to the suction of the EFW pumps, and that prior to opening these isolation valves, the operator must verify that the DWST has sufficient water level to ensure adequate NPSH for the EFW pumps.

The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.1, “Design Bases,” was revised as committed in the RAI response. On the basis of its review of the applicant’s response, and the revised DCD, the staff finds that the concerns identified in RAI 160-1848, Question 10.4.9-7 are resolved, given that (a) the revised DCD Tier 2 include specific information related to operator actions associated with switchover to the DWST and (b) the program for developing and implementing emergency operating procedures is described by means of COL Information Item 13.5(6).

Generic Short Term Recommendation No. 5 (GS-5): GS-5 recommends the plant be capable of providing required EFW flow for at least two hours from one EFWS pump train independent of any ac power source. Two of the four EFWS trains have turbine-driven pump trains that can operate during SBO conditions. The remaining two EFWS trains have motor-driven pumps that can be powered from special AAC GTGs. As described in DCD Tier 2 Section 8.4.2.1.2, “Offsite Power System,” and Section 10.4.9.3, “Safety Evaluation,” the plant can be maintained in a safe condition during the time needed for the AAC GTGs to restore Class 1E ac electrical power. Only one AAC GTG train is needed. With credit for AAC-generated electrical power, the plant can withstand a SBO condition for at least eight hours. However, the staff could not find, in the DCD, a statement regarding the amount of time that the turbine-driven EFWS pump trains could supply flow to the plant in the absence of all ac power. Considerations related to extended operation without ac power include, for example, the continued availability of (I&C and pump room cooling. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-8, dated January 21, 2009, that the applicant demonstrate that the plant is capable of providing required EFW flow for at least two hours from one EFWS pump train independent of any ac power source.

In its response to RAI 160-1848, Question 10.4.9-8, dated February 20, 2009, the applicant stated that the turbine-driven EFWS pumps and associated valves are capable of operating from Class 1E batteries for at least two hours, except that ventilation for the pump rooms may be required after one hour of pump operation, given the design temperature of the pumps and associated support equipment is 79.4°C (175°F). Room cooling is provided by air handling units that are operated by ac power. The ac power needed to support room cooling can be provided by a single unit of the AAC GTG system. The applicant stated that the AAC GTG would be started and operated within one hour to provide pump room ventilation. The AAC GTG also powers battery chargers for the Class 1E batteries.

On the basis of its review of the applicant's response, the staff finds that the applicant has adequately responded to RAI 160-1848, Question 10.4.9-8 with regard to providing sufficient information. However, based on the applicant's response, it appeared that the design did not meet the criteria of GS-5, given that EFWS flow can be maintained during a SBO for only one hour without having to rely on ac power to provide pump room ventilation. Accordingly, the staff requested in RAI 408-3170, Question 10.4.9-24, dated June 24, 2009, that the applicant provide justification as to why the EFWS capability for providing the required EFWS flow for one hour instead of the two hours recommended by GS-5, is acceptable.

In its response to RAI 408-3170, Question 10.4.9-24, dated July 28, 2009, the applicant stated that the US-APWR design meets the GS-5 criterion, because ac power from the AAC GTG system would be available within one hour to provide ventilation for the turbine-driven EFW pump. The applicant further stated that the design of the AAC GTG system conforms to RG 1.155, "Station Blackout," a topic that is also discussed in DCD Tier 2, Section 8.4.1.3, "Alternate AC Power Sources." Per RG 1.155, an ACC power source credited for satisfying the requirements for a SBO should be independent and diverse from onsite emergency power sources, so as to minimize the potential for CCF among the ACC and onsite emergency power sources. RG 1.155 also specifies that there should not be a single-point vulnerability (from a weather-related event or single active failure) that could simultaneously disable the ACC system and any portion of the emergency ac power sources. DCD Tier 2, Section 8.4.1.3, "Alternate AC Power Sources," states that the potential for CCFs associated with the Class 1E GTGs and AAC GTGs are minimized, given that these two sets of GTGs have different ratings and diverse starting systems, along with independent and separate support systems. The adequacy of the US-APWR design with regard to a SBO is evaluated in Section 8.4.1.3 of this SE.

In its original response to RAI 160-1848, Question 10.4.9-8, dated January 21, 2009, the applicant proposed to revise DCD Tier 2, Section 10.4.9.2.2 B, "System Operation," to discuss the availability of the turbine-driven EFW pump trains during a SBO. As part of its response to RAI 408-3170, Question 10.4.9-24, the applicant proposed to further revise DCD Tier 2 Section 10.4.9.2.2 B by making a reference to DCD Tier 2, Section 8.4.1.3, "Alternate AC Power Sources," and its discussion regarding design features that limit CCFs between the AAC and onsite emergency power sources. The applicant also proposed to further revise DCD Tier 2, Section 10.4.9.2.2.B by noting the availability of AAC units during a SBO event.

NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of -Coolant Accidents in Westinghouse - Designed Operating Plants,," and NUREG-0635, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering - Designed Operating Plants," were issued in 1981, prior to the development and issuance of 10 CFR 50.63 "Loss of all alternating current power." GS-5 and GL-3 were recommendations to address the challenges to Auxiliary Feedwater System that arises as a result of loss of all AC power, which for the current fleet of operating plants at that time was a LOOP concurrent with the unavailability of onsite emergency power (usually two diesel generators and no AAC source). 10 CFR 50.63 allows for the use of an AAC source for a SBO. It states that the AAC source will constitute acceptable capability to withstand a SBO provided an analysis is performed which demonstrates that the plant has this capability from the onset of the SBO until the AAC source and required shutdown equipment is started and lined up to operate.

As noted above, the US-APWR design includes an onsite AAC power source to deal with SBO conditions. The AAC power sources (two full capacity, 4000KW, GTGs) are discussed in Section 8.4, "Station Blackout," of the DCD, and reviewed in the corresponding section of this SE. The SBO copying analysis in DCD Section 8.4.2.1.2, "Station Blackout Coping Analysis," states that the power supply from AAC GTG to one of the Class 1E buses can be restored within 60 minutes, and that the availability of power from the AAC GTG to one Class 1E bus within 60 minutes is verified by actual field testing. It is also indicated in DCD Section 8.4.2.1.2 that after the AAC GTG has restored power to the Class 1E power system, action to power heating, ventilation, and air conditioning will be performed to establish the plant in a safe shutdown condition for the long term. Based on the information regarding the AAC GTG system provided by the applicant in the RAI responses and in DCD Section 8.4, the staff concludes that the AAC GTG would be available within one hour to provide ventilation for the turbine-driven EFW pump. The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.1, "Design Bases," was revised as committed in the RAI response. On the basis of its review of the applicant's response, and the revised DCD, the staff finds that the concerns identified in RAI 160-1848, Question 10.4.9-7 are resolved.

Based on the above, the staff concludes that the US-APWR design meets GS-5 since the design is capable of providing EFW flow for at least two hours without any onsite emergency power.

<u>Generic Short Term Recommendation No. 6 (GS-6):</u> GS-6 recommends confirmation of the availability of an EFW flow path that has been taken out of service to perform periodic testing or maintenance, including TS requirements and procedures that require an operator to verify proper alignment of the flow path. The staff identified that TS SR 3.7.5.5 in Chapter 16 of the Tier 2, DCD requires a flow test to verify the EFWS flow paths if the reactor has been in cold shutdown or refueling for an extended period of time, consistent with the recommendations of GS-6. The procedures should include an independent check by a second operator to verify the flow path alignment. However, the staff could not find a specific commitment that the COL applicant would develop procedures that specifically require confirmation of the availability of an EFW flow path that has been previously taken out of service to perform periodic testing or maintenance, including independent verification by a second operator. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-9, dated January 21, 2009, that the applicant provide the procedure that demonstrates how verification of the proper flow path alignment will be accomplished, consistent with GS-6.

In its response to RAI 160-1848, Question 10.4.9-9, dated February 20, 2009, the applicant stated that this procedure would be included as part of COL Information Item 13.5(5). However, to more specifically address this issue, the applicant proposes to add a sentence at the end of DCD Tier 2, Section 10.4.9.2.3, that following periodic testing of the EFWS pumps, an operator will determine that EFWS valve alignments have been properly made, with independent verification by a second operator.

The staff reviewed the applicant's response to RAI 160-1848, Question 10.4.9-9 along with the revisions to DCD Tier 2, Section 10.4.9.2.3. The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2.3 was revised as committed in the RAI response. The staff also reviewed COL Information Item 13.5(5). Since COL Information Item 13.5(5) instructs COL applicants to provide site specific procedures, and DCD Section 10.4.9.2.3, "Testing and Inspection Requirements," specifically states that following periodic testing of the EFWS pumps, an operator will determine that EFWS valve alignments have been properly made, with independent verification by a second operator. The information in DCD Section 10.4.9.2.3, along with COL Information Item 13.5(5) will ensure that procedures developed by the COL applicant will result in confirmation of the availability of an EFW flow path, including independent verification by a second operator. Accordingly, the staff finds that the applicant has adequately addressed the concerns raised by the staff in RAI 160-1848, Question 10.4.9-9 and therefore, the staff considers RAI 160-1848, Question 10.4.9-9 to be resolved.

<u>Additional Short-Term Recommendation (Primary EFW Water Source Low Level Alarm)</u>: In accordance with this additional short-term recommendation, the plant should provide redundant level indication and low level alarms in the control room for the EFWS primary water supply. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming the largest capacity EFW pump is operating. In accordance with DCD Tier 2, Section 10.4.9.2.3, "Testing and Inspection Requirements," and Table 10.4.9-5 "Emergency Feedwater System Summary of Indication and Controls," each pit is provided with two channels of the level transmitters to provide control room indication of the EFW pit water level during normal plant conditions, monitor water level following an accident, and annunciate abnormal water level. However, it did not appear that the applicant had specifically demonstrated that the low level alarm setpoint allows at least 20 minutes for operator action, assuming the largest capacity EFW pump is operating. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-10, dated January 21, 2009, that the applicant demonstrate that the low level alarm setpoint allows at least 20 minutes for operator action.

In its response to RAI 160-1848, Question 10.4.9-10, dated February 20, 2009, the applicant stated that for each pit, there is least 186,200 gallons of water available for EFWS pump suction once the pit low level alarm setpoint is reached. In order for the water level to drop from the low level setpoint to the pump stop water level within 20 minutes, each of the two pumps connected to a given pit would have to pump at about 17,600 L/min (4,650 gpm), which exceeds the maximum possible flow rate even if the SGs are at atmospheric pressure conditions. The applicant proposed to revise DCD Tier 2, Section 10.4.9.2.1, "Description of Major Components," Subsection D to specifically state that the pit capacity and low level alarm setpoint allow operators at least 20 minutes for action following a low level alarm.

The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2.1 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed the concerns raised by the staff in RAI 160-1848, Question 10.4.9-10 because the application now clearly states that the low level alarm setpoint allows at least 20 minutes for operator action, and therefore, the staff considers RAI 160-1848, Question 10.4.9-10 to be resolved.

<u>Additional Short-Term Recommendation (EFW Pump Endurance Test)</u>: In accordance with this additional short-term recommendation, it is requested that a one-time 72-hour endurance test be performed on all EFWS pumps. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted for one hour. In accordance with SRP 10.4.9 Section

III, Item 3, a 48-hour test is acceptable rather than the 72-hour test. However, the staff could not find a commitment regarding a pump endurance test. Accordingly, the staff requested in RAI 160-1848, Question 10.4.9-11, dated January 21, 2009, that the applicant assure that a pump endurance test will be performed in accordance with NUREG-0611 and NUREG-0635, and the incorporation of the test requirement into the inspection and testing section of the US-APWR DCD.

In its response to RAI 160-1848, Question 10.4.9-11, dated February 20, 2009, the applicant stated that the DCD Tier 2 will be modified to include requirements for endurance testing of the EFWS pumps. Specifically, DCD Tier 2, Section 10.4.9.2.3, "Testing and Inspection Requirements," will be revised to include a statement that pump endurance testing will be performed in accordance with the generic recommendations of NUREG-0611 and NUREG-0635. In addition, DCD Tier 2, Section 14.2.12.1.24, "Motor-Driven Emergency Feedwater System Preoperational Test," and Section 14.2.12.1.25, "Turbine-Driven Emergency Feedwater System Preoperational Test," will be revised to include specific requirements for 48-hour endurance tests of the motor and turbine-driven EFWS pumps.

The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2.3, Section 14.2.12.1.24, and Section 14.2.12.1.25 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed the concerns raised by the staff in RAI 160-1848, Question 10.4.9-11 because the applicant clearly specifies the performance of a pump endurance test. Therefore, the staff considers RAI 160-1848, Question 10.4.9-11 to be resolved.

<u>Generic Long Term Recommendation No. 3 (GL-3):</u> The GL-3 recommendation is the same as GS-5 discussed above in this section of this SE. The staff concluded above that the EFWS is in compliance with recommendation GL-5. Therefore, the staff also concludes that the EFWS is in compliance with GL-3.

Based on the above review, the staff finds that the EFWS design is in compliance with generic recommendations of NUREG-0611 and NUREG-0635, and satisfies the requirements of GDC 34, and GDC 44, in that it has the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions, assuming any single active failure, coincident with the LOOP, and the capability to isolate components, subsystems, or piping if required to maintain system safety function.

F. GDC 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System"

The staff reviewed the EFWS for compliance with the requirements of GDC 45 as related to design provisions to permit periodic ISI of system components and equipment, and GDC 46 regarding provisions to permit appropriate functional testing of the system and components.

The EFWS design was reviewed to determine its compliance with the requirements of GDC 45 with regard to designing and locating systems pumps, valves, heat exchangers, and piping, to the extent practicable, to facilitate periodic inspection by providing adequate accessibility. DCD, Tier 1, Section 2.7.1.11.1, "Emergency Feedwater System," states that all of the EFWS components are located in the reactor building. Also, in accordance with DCD Tier 2, Section 3.4.1.2, "Flood Protection from External Sources," the EFWS is located in the non-radiological controlled area (NRCA) of the reactor building, which helps to facilitate inspection.

The staff found the EFWS pumps and the appropriate system valves were included in the plant Inservice Testing (IST) program as described in DCD Tier 2 Section 3.9.6. As indicated in DCD Tier 1 Figure 2.7.1.11-1 and DCD Tier 2, Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumentation Diagram (1/2)," design provisions are provided to permit periodic ISI and operational testing of the EFWS during normal plant conditions.

The staff reviewed the design of the EFW pits for compliance with the requirements of GDC 45 with respect to ISI. As indicated in DCD Tier 2, Figures 1.2-9, 1.2-10, 9A-8, and 9A-9, each pit is located in the reactor building within its own cubicle. However, these figures do not indicate doorways or other means of entry to these cubicles to facilitate inspections of the pits. Accordingly, the staff requested, in RAI 160-1848, Question 10.4.9-12, dated January 21, 2009, that the applicant explain how pit inspections will be accomplished.

In its response to RAI 160-1848, Question 10.4.9-12, dated February 20, 2009, the applicant stated that the COL applicant is responsible for in-service testing and inspection of seismic Category 1 structures, including the EFW pit liner, per DCD Tier 2, Section 3.8.4.7, "Testing and Inservice Requirements." This monitoring requirement is also stated in COL Information Item 3.8(22).

The applicant further stated that all structural components inside the pit, including the pit liner, are constructed of stainless steel. An access hatch located above the 100 percent water level is available for inspections of the pit interior areas. The pits do not contain any equipment. Once construction and installation is complete, an inspection will be performed to assure the integrity of the pit liner. Further, completed inspections with the pits drained will be performed per the ISI program. Given that the EFW pits are completely enclosed structures, intrusion of foreign materials is not anticipated.

The staff finds that the applicant's response to RAI 160-1848, Question 10.4.9-12 is acceptable, except that the applicant does not propose to include any of this information in the DCD. Accordingly, the staff requested in RAI 408-3170, Question 10.4.9-25, dated June 24, 2009, that the applicant include this response into the revised Tier 2 DCD.

In its response to RAI 408-3170, Question 10.4.9-25, dated July 28, 2009, the applicant stated that DCD Tier 2, Section 10.4.9.2.1 D will be revised to state that the pits are completely enclosed, do not contain any operating equipment, are filled with clean demineralized water, and that inspections of the pit interior can be accomplished by means of an access hatch located above the water level. The revised DCD text will also state that complete inspections of the pits (after they are drained) will be periodically performed in conjunction with the ISI program. The staff has confirmed that Revision 2 DCD Tier 2, Sections 10.4.9.21 was revised as committed in the RAI response. With the revision to DCD Tier 2, Section 10.4.9.2.1 D the applicant has adequately addressed the concerns raised by the staff in RAI 160-1848, Question 10.4.9-25 since the application now clearly explains how pit inspections will be accomplished. Therefore, the staff considers RAI 160-1848, Question 10.4.9-25 to be resolved.

The staff found the EFWS pumps and the appropriate system valves were included in the plant IST program as described in DCD Tier 2 Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." As indicated in DCD Tier 1 Figure 2.7.1.11-1, "Emergency Feedwater System Location of Equipment and Piping," and DCD Tier 2, Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumentation Diagram (1/2)," design provisions are provided to permit periodic ISI and operational testing of the EFWS during normal plant conditions. A full flow line with a normally closed valve and an orifice allows pump testing during normal plant operation at the pump design flow rate without injection to the SGs. The system is also capable of manual actuation to provide for operational testing independent of the automatic signal. Each full flow line routes pump discharge flow to the EFW pits. Instrumentation identified on DCD Tier 2, Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumentation Diagram (1/2)," includes pump flow rate and differential pressure developed by each pump, which are necessary for testing.

The EFWS, including its signals and circuits are capable of being tested periodically while the plant is at power, in accordance with the frequency specified in the TSs (Chapter 16), Section 3.7.5. However, the DCD did not appear to confirm that the testing will include transfer between normal and emergency buses, as specified in GDC 46 and SRP 10.4.9 Section IV, Item 9. Accordingly, the staff requested, in RAI 160-1848, Question 10.4.9-13, dated January 21, 2009, that the applicant demonstrate how the EFWS is tested with regard to transfer between normal and emergency buses.

In its response to RAI 160-1848, Question 10.4.9-13, dated February 20, 2009, the applicant stated that the testing described in DCD Tier 2, Section 14.2.12.1.45, "Class 1E Bus Load Sequence Preoperational Test," will include the transfer between normal and emergency buses. The applicant proposes to revise DCD Tier 2, Section 10.4.9.2.3, "Testing and Inspection Requirements," to state that testing of the EFWS will include testing of the transfer between normal and emergency buses.

The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2.3 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue since the application now clear states that the Chapter 14 initial testing of the EFWS will include testing of the transfer between normal and emergency buses. Therefore, the staff considers RAI 160-1848, Question 10.4.9-13 to be resolved.

Based on the above review, the staff finds that the EFWS satisfies the requirements of GDC 45 and GDC 46, since design provisions are provided to permit periodic ISIs of EFWS components and equipment, and operational testing of the EFWS during normal plant conditions.

G. GDC 60, "Control of Releases of Radioactive Materials to the Environment"

The staff reviewed the design of the EFW pits for compliance with the requirements of GDC 60 with respect to control of releases of radioactive materials. In accordance with SRP 9.2.6 Section III, Item 3.E, condensate tank overflow should be connected to the radwaste system. GDC 60 requires that a means be provided to control the release of radioactive materials in liquid effluents. As indicated in DCD Tier 2, Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumentation Diagram (1/2)," overflow from the EFW pits are routed to the non-radioactive sump system. Pit overflow would not be subject to radioactive contamination, given that the pit inventory is provided by means of the DWST. Since the EFW pits will not contain liquid effluents containing radioactive material, the staff concludes that there is no potential for the release of radioactive material to the environment from the EFW pits and GDC 60 does not apply.

H. 10 CFR 50.62, "Requirements for Reduction of Risk from ATWS Events for Light-Water-Cooled Nuclear Power Plants"

The staff reviewed the EFWS for compliance with the requirements of 10 CFR 50.62 regarding provisions for automatic initiation in an ATWS. The applicant states in DCD Tier 2 Section 7.8, "Diverse Instrumentation and Control," and Section 10.4.9.2.2, "System Description," that the US-APWR design includes a DAS that is capable of automatically actuating the EFWS under conditions indicative of an ATWS. The DAS is supported by conventional hardware circuits (analog circuits, solid-state logic processing, and relay circuits) that are diverse, separate, and isolated from the safety-related protection and safety monitoring system (PSMS) and its reactor trip (RT) function. As indicated in DCD Tier 2, Table 7.8-4, "Combined License Information," a diverse low SG level EFWS actuation signal is provided for ATWS mitigation. Therefore, the design was found to satisfy 10 CFR 50.62 regarding provisions for automatic initiation in an ATWS.

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

The staff reviewed the EFWS for compliance with the requirements of 10 CFR 50.63 regarding the capability for responding to a SBO. An applicant may demonstrate compliance with this requirement by meeting Positions 3.2.2, 3.3.2 and 3.3.4 of RG 1.155 "Station Blackout", which is based on the EFWS design providing for sufficient decay heat removal in a SBO
Two of the four EFWS trains have turbine-driven pump trains that can operate during SBO conditions. The remaining two EFWS trains have motor-driven pumps that can be powered from special AAC GTGs. The electrical power system compliance with 10 CFR 50.63 is reviewed in Section 8.4, "Station Blackout," of this SE. As described in DCD Tier 2, Section 8.4.2.1.2, "Station Blackout Coping Analysis," and Section 10.4.9.3, "Safety Evaluation," the plant can be maintained in a safe condition during the time needed for the AAC GTGs to restore Class 1E ac electrical power. Only one AAC GTG train is needed. With credit for AAC-generated electrical power, the plant can withstand a SBO condition for at least eight hours. DCD Tier 2, Section 10.4.9.3 indicates that the total usable water inventory in the EFW pits 1.55 x 10^6 L (409,700 gallons) is sufficient for decay heat removal during the eight hour SBO duration

J. Miscellaneous

The US-APWR has design provisions that detect and mitigate steam binding of the EFWS pumps due to back-leakage from the SGs to the EFWS. Steam leakage from the SGs to the EFWS pumps during standby conditions is prevented by a series arrangement of two check valves in each pump train, as shown in DCD Tier 2, Figure 10.4.9-1. The applicant states in Tier 2, Section 10.4.9.3, that temperature monitoring is performed in the EFW discharge lines as a means to detect back leakage. The EFW system design for recognizing the effects of steam binding of EFW pumps is consistent with guidance in Generic Safety Issue (GSI)-93, "Steam Binding of Auxiliary Feedwater Pumps," and associated GL 88-03, "Resolution of Generic Safety Issue 93." GL 88-03 specifically recommends that procedures be in place for recognizing steam binding and for restoring the EFWS to operable status if steam binding is detected. However, the staff could not find any information in the DCD to ensure that the COL applicant develops operating and maintenance procedures to address steam binding issues. Accordingly, the staff requested, in RAI 160-1848, Question 10.4.9-14, dated June 24, 2009, that the applicant provides the operating and maintenance procedures that address steam binding issues.

In its response to RAI 160-1848, Question 10.4.9-14, dated February 20, 2009, the applicant stated that DCD Tier 2, Section 10.4.9.3 will be revised to describe a restoration procedure for addressing situations where bypass leakage through the EFW check valves is detected. This restoration procedure will require the isolated area to be filled with water prior to returning to service. The applicant further indicated how the TSs require restoration of a train outage be completed within 72 hours. The staff finds that the applicant's restoration procedure is acceptable because they incorporate the recommendations in GSI-93.

However, the staff notes in the applicant's response to RAI 160-1848, Question 10.4.9-14, dated February 20, 2009, that the applicant does not specifically discuss operating procedures that would help prevent or lead to recognition of steam binding issues. Accordingly, the staff requested, in RAI 408-3170, Question 10.4.9-26, dated June 24, 2009, that the applicant describes the pertinent operating procedures.

In its response to RAI 408-3170, Question 10.4.9-26, dated July 28, 2009, the applicant stated that this issue has been addressed as part of an amended response to RAI 160-1848, Question 10.4.9-4. The amended response to RAI 160-1848, Question 10.4.9-4 is described in a letter dated June 1, 2009. In its amended response to RAI 160-1848, Question 10.4.9-4, the applicant proposed adding a new COL Information Item 10.4(6) requiring COL applicants to develop operating and maintenance procedures in accordance with NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," for the EFWS that address minimization of potential water hammer. The applicant also proposed to revise DCD Tier 2, Section 10.4.9.2.2, "System Operation," to ensure that the procedures specifically address important elements, for example the introduction of steam, heated water, or voids into water-filled lines, and proper warm-up and drainage of steam-filled lines. The staff reviewed the proposed COL Information Item 10.4(6) and the revised DCD Tier 2, Section 10.4.9.2.2 and found that with the addition of the new COL item and the revision of the DCD text, the concerns raised by the staff in RAI 408-3170, Question 10.4.9-26 were resolved.

The staff reviewed design provisions that have been incorporated to provide minimum flow for EFWS pump cooling. Minimum flow check valves for each EFWS pump are depicted in DCD Tier 2, Figure 10.4.9-1, "Emergency Feedwater System Piping and Instrumentation Diagram (1/2)." The pump minimum flow recirculation lines discharge recirculation water back into the EFW pits. However, there did not appear to be a discussion in the DCD about pump minimum flow requirements addressed in NRC IE Bulletin IEB 88-04, "Potential Safety-Related Pump Loss." This bulletin discusses, in part, pump minimum flow requirements as they relate not only to pump cooling due to fluid temperature rise, but also to hydraulic instability due to insufficient minimum flow, resulting in pump cavitation and potential damage of the impeller. This bulletin recommends that the limitations associated with these hydraulic phenomena be considered when specifying minimum flow capacity. Given that there does not appear to be any mention of IEB 88-04 in the DCD, the staff requested in RAI 160-1848, Question 10.4.9-15, dated January 21, 2009, that the applicant provide an explanation of how the pump minimum flow requirements of IEB 88-04 were addressed.

In its response to RAI 160-1848, Question 10.4.9-15, dated February 20, 2009, the applicant stated that the minimum flow rate of each EFW pump is based on hydraulic phenomena determined by the pump vendor. The applicant further stated that DCD Tier 2, Section 10.4.9.2.1, Subsection A, "Emergency Feedwater Pumps," will be revised to explicitly indicate that the requirements of IEB 88-04 will be followed to ensure that the EFWS pump minimum flow lines have sufficient capacity.

The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2.1 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue since the EFWS will be designed in accordance with IEB 88-04. Therefore, the staff considers RAI 160-1848 Question 10.4.9-3 to be resolved.

<u>10.4.9.4.2 Technical Specifications</u>

The staff reviewed the TS requirements for the EFWS as presented in DCD Tier 2 Chapter 16. In MODES 1, 2, and 3, the EFWS, including the EFW pits, are required to be operable. For MODE 4, the EFWS is not required to be operable because SG heat removal is not credited in any MODE 4 safety analysis (see DCD Tier 2, Section B 3.4.6 and Technical Report MUAP-07036 (R1), "Justification for Deviations between NUREG-1431 Rev. 3.1 and US-APWR Technical Specifications." The EFWS is addressed in proposed TS Section 3.7.5 and Bases 3.7.5. Similarly, the EFW pits are addressed in proposed TS Section 3.7.6 and Bases 3.7.6. In MODE 5 or 6, the SGs are not normally used for heat removal, and the EFWS is not required.

Applicable EFWS Limiting Conditions for Operation (LCO) are provided in DCD Tier 2 Chapter 16 LCO 3.7.5. For applicability to MODES 1, 2, and 3, the LCO and the associated Bases were reviewed and found to be acceptable.

SRs for the following parameters are provided: (1) valve alignment confirmation, (2) pump developed head, (3) verification that automatic valves actuate to correct position on actual or simulated actuation signal, (4) pump start on actual or simulated actuation signal, and (5) verification that the EFWS is properly aligned from EFW pits to the SGs prior to entering MODE 2 after more than 30 days in MODE 5 or 6. EFW alignment is also verified following extended outages.

The staff determined that the surveillance parameters listed above for the EFWS are reasonable, since they provide for pump operability, proper system alignment, and correct automatic response of the EFWS pumps and valves. However, there was no TSs that addresses water quality associated with the EFW pit inventory. Accordingly, staff requested in RAI 160-1848, Question 10.4.9-16, dated January 21, 2009, that the applicant add a SR to ensure that the EFW pit water quality is appropriately maintained.

In its response to RAI 160-1848, Question 10.4.9-16, dated February 20, 2009, the applicant stated that the TSs for the US-APWR are based on NUREG-1431, "Standard Technical Specifications Westinghouse Plants," which does not include SRs for water chemistry related to EFWS suction sources. The applicant provided additional rationale for excluding SRs, though the staff found the related discussions to be unclear. Specifically, the applicant stated that with regard to PRA or safety analysis, the water chemistry "expects nothing." In addition, with regard to the 40.6°C (105°F) maximum design temperature of the EFW pit, the applicant stated that "…it is not the condition the SCC occurs." Accordingly, the staff requested in RAI 408-3170, Question 10.4.9-27, dated June 24, 2009, that the applicant clarify the rationale used to exclude SRs for water chemistry related to EFWS suction sources.

In its response to RAI 408-3170, Question 10.4.9-27, dated July 28, 2009, the applicant stated that the EFWS heat removal function does not depend on the water quality, but only on the amount of water available in the EFW pits. The applicant further stated that water for the EFW pits is supplied by the CST. Because water transferred into the pits is not deaerated, and because the pits have direct communication with the atmosphere, the pit inventory will contain dissolved oxygen. However, the dissolved oxygen would not cause stress corrosion cracking of the pit surfaces, given that the pit design temperature is 40.6 °C (105°F), which is well below temperatures needed to initiate stress corrosion cracking. The applicant proposed to revise DCD Tier 2 Section 10.4.9.2.1 to explain why the dissolved oxygen does not lead to stress corrosion cracking. The staff reviewed the proposed revision to DCD Section 10.4.9.2.1 and found that with the revision of the DCD text, the concerns raised by the staff in RAI 10.4.9-27 were resolved because the applicant justified the exclusion of surveillance requirements for water chemistry for EFW pits based on the fact that water chemistry does not impact the EFWS heat removal function. The staff has confirmed that Revision 2 DCD Tier 2, Section 10.4.9.2.1 was revised as committed in the RAI response. Accordingly, the staff considers RAI 408-3170 Question 10.4.9-27.

The staff reviewed the LCO and the associated Bases for the EFW pits and found them to be acceptable. The applicable LCO for the EFW pits is provided in DCD Tier 2 Chapter 16; LCO 3.7.6. The LCO includes two action levels for a condition in which one or both EFW pits become inoperable.

There is one surveillance requirement for the EFW pits, namely that the pit level be maintained at or above 7.75×10^5 L (204,850 gallons) (SR 3.7.6.1).

As described in DCD Tier 2 Section 10.4.9.2.1, Item D, “Emergency Feedwater Pits,” the EFW pits are connected by a tie line with two normally closed manual valves. If these valves are not maintained closed, it might be possible for a fault in one pit (e.g., a leak) to drain inventory from the remaining pit. However, a surveillance requirement for maintaining the EFW pit cross tie valves in the closed position was not provided. Accordingly, staff requested in RAI 160-1848 Question 10.4.9-17 that the applicant add a surveillance requirement to ensure that the EFW pit cross connect valves are normally maintained in the closed position.

In its response to RAI 160-1848 Question 10.4.9-17 in a letter dated February 20, 2009. In that response, the applicant stated that a surveillance requirement related to the positioning of the EFW pit cross connect valves is not required, given that an existing surveillance requirement (SR 3.7.6.1) requires periodic verification of the water level for each pit. The applicant further states that this surveillance would serve as a means to detect pit leakage. As described in DCD Tier 2 Section 16, “Technical Specifications,” the level of each pit is to be verified with a frequency of 12 hours or in accordance with the Surveillance Frequency Control Program.

On the basis of its review of the applicant’s response, the staff finds that the concerns identified in RAI 160-1848 Question 10.4.9-17 are resolved, given that the pit levels are periodically verified.

Based on a detailed review of proposed TS Sections 3.7.5 and 3.7.6 and Bases 3.7.5 and 3.7.6, the staff finds the EFWS will be operated in accordance with its design bases requirements.

10.4.9.4.3 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

Proposed ITAAC for the EFWS are given in DCD Tier 1 Table 2.7.1.11-5 (EFW ITAAC). Table 2.7.1.11-5 contains test and inspection requirements for the EFWS. These tests and/or inspections confirm: (1) components designated as Class 1E will be qualified to perform in harsh environment [Item 6.a], (2) adequate NPSH to the system pumps [Item 14], (3) design flow rates to SGs for design conditions [Item 12], (4) adequate EFW pit volume [Item 13], (5) the functional arrangement of the EFWS, which includes cross-connections that allow alignments of EFWS pump suction to both EFW pits and EFWS pump discharge to any SG [Item 1.a], (6) remotely operated valves can be opened and closed from the main control room [Item 8.a], and (7) the ability of system valves to reposition in accordance with the design [Item 9.a].

Section 2.7.1.11.1 of the Tier 1 DCD indicates that the EFWS is designed to remove decay heat and sensible heat during various transient and accident conditions, including MSLB and SGTR.

The EFWS should be designed to limit the maximum amount of feedwater that can be discharged following a MSLB to prevent excessive SG feedwater flow and pump runout. Furthermore, the EFWS should be designed to limit the maximum amount of feedwater that can be discharged into a failed SG so that SG overfill is prevented. Limitations on maximum EFW flow rates are discussed in DCD Tier 2 Sections 10.4.9.2 and 10.4.9.2.1. However, the staff could not find an ITAAC entry or DCD Tier 1 discussion that specifically addresses limitations on maximum flowrates. Therefore, the staff requested in RAI 160-1848 Question 10.4.9-18 that the applicant demonstrate how it will be assured that limitations on flowrates will be addressed as part of the ITAAC process.

In its response to RAI 160-1848 Question 10.4.9-18 in a letter dated February 20, 2009. In that response, the applicant stated that DCD Tier 1, Table 2.7.1.11-5, "Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria," will be revised to include an ITAAC entry (no. 15) related to related limiting the maximum flow to a depressurized SG. However, the acceptance criteria portion of this ITAAC entry includes the following statement: "…sum of maximum flow to each SG is less than 3463 L/min (915 gpm) with pumps running against a faulty SG pressure of 0 psig." The applicant did not provide a basis for the 3463 L/min (915 gpm) cited in this statement. Accordingly, the staff requested in **RAI 408-3170 Question 10.4.9-28** that the applicant provide the basis for the 3463 L/min (915 gpm).

In its response to RAI 408-3170 Question 10.4.9-28 in a letter dated July 28, 2009. In that response, the applicant stated that EFW control valves are adjusted during pre-operational testing to limit the maximum EFW flow rate into each SG. The control valves limit the maximum flow into the SGs during a main steam line break (MSLB) or SGTR. The maximum flow into each SG will be set at 91 m^3/h (400 gpm) at a SG pressure of 8,418 kPa gauge (1221 psig). The pre-operational testing to confirm this flow limit must be performed during hot shutdown conditions. This flow limit will in turn preserve SG flow assumptions used in the transient and accident analyses associated with MLSB and SG overfill during a SGTR. The applicant proposed to revise ITAAC Item 15 in DCD Tier 1 Table 2.7.1.11-5, "Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria" to state the revised flow limit and summarize its basis. The staff has confirmed that Revision 2 DCD Tier 1, Table 2.7.1.11-5 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 408-3170 Question 10.4.9-28 resolved.

Section 2.7.1.11.1 of the Tier 1 DCD describes flow recirculation lines from each EFW pump that permit testing of each EFW pump at full flow. Figure 2.7.1.11-1 of the Tier 1 DCD displays flow recirculation lines that are connected to pump discharge paths. SRP 14.3, Appendix C, Item I.A.xiv states that normally, all design commitments in Tier 1 should be verified by an ITAAC entry, unless there are specific reasons why this is not necessary. SRP 14.3, Appendix C, Item II.B.iv, states that online test features should be verified by ITAAC. However, the staff could not locate supporting information that specifically demonstrates how the capability of EFW pump flow test features will be verified through the ITAAC process (e.g., functional flow tests). Accordingly, the staff requested in RAI 160-1848 Question 10.4.9-19 that the applicant include the capability of the pumps to be tested at flow during plant operation as part of the ITAAC process.

In its response to RAI 160-1848 Question 10.4.9-19 in a letter dated February 20, 2009. In that response, the applicant stated that revisions will be made to DCD Tier 1, Table 2.7.1.11-5, "Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria," will be revised to include an ITAAC entry (no. 16) to verify the capability to perform full flow testing of

EFWS pumps by means of the pump recirculation lines.

The staff has confirmed that Revision 2 DCD Tier 1, Table 2.7.1.11-5 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 160-1848 Question 10.4.9-19 resolved.

SRP 14.3, Appendix C, Item I.B.ix states that Tier 1 figures for safety-related systems should include most of the valves on the DCD Tier 2 P&ID except for items, such as fill, drain, test tees, and maintenance isolation valves. The scope of valves to be included on the figures are those motor-operated valves (MOVs), power-operated valves (POVs), and check valves with a safety related active function. Also, SRP 14.3, Appendix C, Item II.B.i states that, typically, the system ITAAC specify functional tests or tests and analyses, to verify the direct safety functions for the various system operating modes.

Figure 2.7.1.11-1 of the Tier 1 DCD illustrates the arrangement of EFW components by means of a flow diagram. A more detailed version of the EFW configuration is provided in the Figures 10.4.9-1 and 10.4.9-2 of the Tier 2 DCD. The set of additional details provided in Figures 10.4.9-1 and 10.4.9-2 of the Tier 2 DCD includes various check valves. By comparison, Figure 2.7.1.11-1 of the Tier 1 DCD does not include any check valves. It appears that at least some of the check valves shown in Figures 10.4.9-1 and 10.4.9-2 of the Tier 2 DCD have a safety related function (e.g., some check valves would prevent flow diversion of water through an inactive pump). Also, EFWS check valves are not explicitly identified in the ITAAC shown in Table 2.7.1.11-5 of the Tier 1 DCD. Information provided by the applicant was insufficient for the staff to determine why EFWS check valves have been excluded from the Tier 1 EFWS flow diagram and from the ITAAC. Accordingly, the staff requested in **RAI 160-1848 Question 10.4.9-20** that the applicant addresses the exclusion of check valves from the Tier 1 EFWS flow diagram and ITAAC.

In its response to RAI 160-1848 Question 10.4.9-20 in a letter dated February 20, 2009. In that response, the applicant stated that revisions will be made to DCD Tier 1 to include check valves with an active safety function. The following portions of the DCD Tier 1 will be revised to include these check valves: Table 2.7.1.11-1, "Emergency Feedwater System Location of Equipment and Piping;" Table 2.7.1.11-2, "Emergency Feedwater System Equipment Characteristics;" and Figure 2.7.1.11-1, "Emergency Feedwater System." In addition, DCD Tier 1, Table 2.7.1.11-5, "Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria," will be updated to include testing of these check valves through revisions to ITAAC entry no. 9.a.

The staff has confirmed that Revision 2 DCD Tier 1, Tables 2.7.1.11-1, 2.7.1.11-2, 2.7.1.11-5, and Figure 2.7.1.11-1 were revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 160-1848 Question 10.4.9-19 resolved.

The staff noted an apparent typographical error in the applicant's proposed revision to DCD Tier 1, Figure 2.7.1.11-1 with regard to the numbering of the check valves. In particular, what would appear to be check valve "VLV-109A" seems to be incorrectly labeled as check valve "VLV-109D." Accordingly, the staff requested in RAI 408-3170 Question 10.4.9-29 that the applicant correct the proposed revision to DCD Tier 1, Figure 2.7.1.11-1 with regard to the labeling of check valve "VLV-109A."

In its response to RAI 408-3170 Question 10.4.9-29 in a letter dated July 28, 2009. In that response, the applicant agreed that the check valve was incorrectly labeled in the proposed revision to DCD Tier 1, Figure 2.7.1.11-1. The applicant proposed to further revise this figure to correct the check valve labeling. The staff has confirmed that Revision 2 DCD Tier 1, Figure 2.7.1.11-1 now has the check valve correctly labeled. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 408-3170 Question 10.4.9-29 resolved

As previously noted, there is one Technical Specification surveillance requirement (SR) for the EFW pits, namely that the pit level be maintained at or above 7.75 x 10^5 L (204,850 gallons) (SR 3.7.6.1). Also, DCD Tier 2 Section 10.4.9.3 states that the usable volume per pit is 204,850 gallons. However, the Acceptance Criteria for ITAAC 13 as shown in DCD Tier 1 Table 2.7.1.11-5 states that the water volume of each pit must be greater than or equal to 7.05 x 10^5 L (186,200 gallons). Thus, it appears that ITAAC 13 is not consistent with the minimum pit capacity cited in SR 3.7.6.1 and DCD Tier 2 Section 10.4.9.3. Accordingly, staff requested in RAI 160-1848 Question 10.4.9-21 that the applicant reconcile the discrepancy between the minimum pit capacity cited in the ITACC and the minimum pit capacity cited in the Technical Specifications and DCD Tier 2 Section 10.4.9.3.

In its response to RAI 160-1848 Question 10.4.9-21 in a letter dated February 20, 2009. In that response, the applicant stated that 7.05 x 10^5 L (186,200 gallons) represents the quantity of water per pit that is required for removing decay heat for 14 hours, based on 8 hours of hot standby and 6 hours of hot shutdown cooling. The applicant stated that while ITAAC 13 in DCD Tier 1, Table 2.7.1.11-5, “Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria,” specifies a minimum pit capacity of 7.05 x 10^5 L (186,200 gallons), the actual available volume of water will be at least 204,850 gallons once design margins and instrumentation error allowances are accounted for. The applicant also provided a figure that displays a cross-section schematic of a pit, where various water levels are labeled, including the 100% pit capacity (9.12 x 10^5 L (241,000 gallons)). However, the applicant proposed to revise only DCD Tier 2, Section 10.4.9.3, “Safety Evaluation,” with regard to stating that the total required EFW pit volume with a 10% margin is 7.75 x 10^5 L (204,850 gallons) per pit, or 1.55 x 10^6 L 409,700 gallons combined for both pits.

The staff felt that the applicant’s response had not adequately reconciled the discrepancy between the Technical Specifications and ITAAC 13, given that the applicant had not proposed to revise ITAAC 13. Accordingly, the staff requested in RAI 408-3170 Question 10.4.9-30 that the applicant revise the wording in ITAAC 13 so that it clearly specifies a minimum available pit capacity of 7.75 x 10^5 L (204,850 gallons).

In its response to RAI 408-3170 Question 10.4.9-30 in a letter dated July 28, 2009. In that response, the applicant proposed to revise ITAAC Item 13 in DCD Tier 1 Table 2.7.1.11-5, “Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria” so that it specifies a minimum available pit capacity of 7.75 x 10^5 L (204,850 gallons). The staff reviewed the proposed revisions to DCD Tier 1 Table 2.7.1.11 and found that with the revisions of ITAAC Item 13, the concerns raised by the staff in RAI 408-3170 Question 10.4.9-30 were resolved. The staff has confirmed that Revision 2 DCD Table 2.7.1.11-5 was revised as committed in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue since now the DCD is clear regarding the minimum pit capacity. Therefore, the staff considers RAI 408-3170 Question 10.4.9-30 resolved.

ITAAC requirements in DCD Tier 1, Table 2.7.1.11-5 and Tier 2 Section 14.3 were also reviewed. The ITAAC acceptance criteria contained in Table 2.7.1.11-5 were found to be appropriate.

Based on the above review, the staff finds that the ITAAC will adequately confirm EFWS and EFW pit design capabilities, design features, and systems interfaces.

10.4.9.4.4 Initial Test Program

Applicants for standard plant design approval must provide plans for preoperational testing and initial operations in accordance with 10 CFR 50.34(b)(6)(iii) requirements. Preoperational test requirements for EFWS are listed in DCD Tier 2 Section 14.2.12.1.24, "Motor-Driven Emergency Feedwater System Preoperational Test:" Section 14.2.12.1.25, "Turbine-Driven Emergency Feedwater System Preoperational Test:" and Section 14.2.12.1.97, "Emergency Feedwater Pump Area HVAC System Preoperational Test." The initial test program for US-APWR is evaluated in Section 14.2 of this SER, and evaluation of the EFWS initial test program in this section is an extension of the evaluation provided in Section 14.2 of this SER.

Testing of EFWS includes checks on the operability of system pumps, valves, controls, instrumentation, indications, and alarms. Also included are checks to ensure that flow instabilities, such as water hammer, do not occur. Testing of the EFW pits did not include chemistry or water quality testing. Accordingly, the staff requested in **RAI 160-1848, Question 10.4.9-22** that the applicant address testing of pit water chemistry and water quality as part of the preoperational testing.

In its response to RAI 160-1848, Question 10.4.9-22 in a letter dated February 20, 2009. In that response, the applicant provided rationale for excluding testing of the pit water chemistry, though the staff found portions of the applicant's discussion to be unclear. Specifically, the applicant stated that the maximum design temperature of the EFW pit is 40.6°C (105°F), and "at the low temperature like that, SCC does not occur." The applicant further stated that water sampling will be limited to ensuring that the turbidity does not exceed 1 ppm, a statement also made by the applicant in response to RAI 160-1848 Question 10.4.9-6.

The staff requested in RAI 408-3170 Question 10.4.9-31 that the applicant clarify the statement quoted above.

In its response to RAI 408-3170, Question 10.4.9-31 in a letter dated July 28, 2009. In that response, the applicant referred to their response to RAI 408-3170, Question 10.4.9-27. As the applicant stated in the response to RAI 408-3170, Question 10.4.9-31, the EFWS heat removal function does not depend on the water quality, but only on the amount of water available in the EFW pits. The applicant further stated in the response to RAI 408-3170, Question 10.4.9-27 that water for the EFW pits is supplied by the CST. Because water transferred into the pits is not deaerated, and because the pits have direct communication with the atmosphere, the pit inventory will contain dissolved oxygen. However, the dissolved oxygen would not cause stress corrosion cracking of the pit surfaces, given that the pit design temperature is 40.6°C (105°F), which is well below temperatures needed to initiate stress corrosion cracking. The applicant proposed to revise DCD Tier 2, Section 10.4.9.2.1, "Description of Major Components," to explain why the dissolved oxygen does not lead to stress corrosion cracking. The staff has confirmed that Revision 2 DCD Table 2.7.1.11-5, "Emergency Feedwater System Inspections, Tests, Analyses, and Acceptance Criteria," was revised as committed. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 408-3170, Question 10.4.9-31 to be resolved.

Based on the above review, the staff finds that the initial test program is adequate to demonstrate proper EFWS and EFW pit operations.

10.4.9.5 Combined License (COL) Information Items

The following is a list of item numbers, with the table-provided descriptions, from Table 1.8-2, “Compilation of All Combined License Applicant Items for Chapters 1-19,” of the Tier 2 application that are directly applicable to the EFWS.

Table 10.4.9-1
U.S. APWR Combined License Information Items

COL Item No.	Description	DCD Tier 2 Section	Action Required by COL Applicant	Action Required by COL Holder
COL 10.4(6)	Operating and maintenance procedures for water hammer prevention; The Combined License Applicant is to provide operating and maintenance procedures in accordance with NUREG-0927 and a milestone schedule for implementation of the procedure. *implementation of the procedure*	10.4.9	Y	
COL 13.5(5)	The COL Applicant is to describe the program for developing operating procedures	13.5.3	Y	

10.4.9.6 Conclusions

The staff finds that the review of the DCD application supported that the EFWS functional design is acceptable for the reasons set forth above and meets the requirements of GDCs 2, 4, 5, 19, 34, 44, 45, 46, and 60, ATWS requirements of 10 CFR 50.62, SBO requirements of 10 CFR 50.63(a)(2), and the requirements of 10 CFR 52.47 (b)(1) regarding ITTAC.

As set forth above, the staff finds that the ITAAC, TSs and COL applicant information items specified, ensure that site-specific information not provided in the DCD is identified and addressed with respect to the EFWS, and that the EFWS can be properly inspected, tested and operated in accordance with DCD requirements.

10.4.10 Secondary Side Chemical Injection System

10.4.10.1 Introduction

The secondary side chemical injection system (SCIS) feeds required chemicals to control pH and dissolved oxygen content of the feedwater, condensate and SG secondary side water. The deaerator (Section 10.4.7), the CPS (Section 10.4.6), the SGBS (Section 10.4.8), and the secondary side sampling system (Section 9.3.2); also contribute to the secondary side water chemistry control. The SCIS does not have a safety-related function.

10.4.10.2 Summary of Application

Tier 1: The Tier 1 information associated with this Section 10.4.10 is found in Tier 1, Section 2.7.1.12, "Secondary Side Chemical Injection System," of the US-APWR DCD.

Tier 2: The applicant has provided a Tier 2 system description in Section 10.4.10, "Secondary Side Chemical Injection System," of the US-APWR DCD, summarized here in part, as follows:

The SCIS consists of a bulk chemical sub-system and two chemical injection sub-systems: one for pH controller injection for pH control, and the other for oxygen scavenger injection for oxygen removal. Each chemical injection sub-system consists of an agitated chemical addition tank and injection pump. The bulk chemical system includes one bulk chemical storage tank, two bulk chemical storage pumps, and three portable drum pumps. The chemical injection system is shown in Figure 10.4.10-1. Major component parameters are provided in Table 10.4.10-1.

ITAAC: The ITAAC associated with Tier 2, Section 10.4.10 is given in Tier 1, Section 2.7.1.12.2, "Secondary Side Chemical Injection System," of the US-APWR DCD.

TSs: There is no TSs for this area of review.

10.4.10.3 Regulatory Basis

NUREG-0800 does not include a section specifically addressing the secondary SCIS. The acceptability of the system is based on meeting the requirements of GDC 4. In other words, failure of the secondary SCIS, as a result of a pipe break or malfunction of the system should not adversely affect safety-related systems or components.

Also applicable is 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

10.4.10.4 Technical Evaluation

The secondary SCIS feeds required chemicals to control pH and dissolved oxygen content of the feedwater, condensate and SG secondary side water. Alkaline pH is maintained in the secondary side with pH controller injection and dissolved oxygen is removed (scavenged) by oxygen scavenger injection.

For the US-APWR, none of the users depend on the SCIS system for safety-related functions or backup. The SCIS does not have a safety-related function and has no safety design basis. Tier 2, DCD, Revision 3, Table 3.2-2, "Classification of Mechanical and Fluid Systems, Components, and Equipment," describes the SCIS (Item 62) as being located in the TB with a seismic category as non-seismic. Tier 2, DCD Section 3.7.2.4.1, "Seismic System Analysis," describes that the TB does not contain or support safety-related SSCs.

The SCIS utilizes morpholine and dimethylamine for pH control and ammonia for system layup. These specific chemicals related to control room habitability is address under COL Information Item 6.4(1).

The staff's evaluation related to verifying that SCIS nonsafety-related SSCs do not reduce the function of safety related SSCs during and after a safe shutdown earthquake is described in Section 3.7 of this SE.

10.4.10.5 Combined License (COL) Information Items

There are no COL Information Items associated with Section 10.4.10 identified in the US-APWR DCD.

10.4.10.6 Conclusions

The staff's reviewed of the SCIS is found in Section 3.7 as it relates to the SCIS non safety-related SSCs interactions against the function of safety related SSCs.

10.4.11 Auxiliary Steam Supply System (ASSS)

10.4.11.1 Introduction

The auxiliary steam supply system (ASSS) is a nonsafety-related system that supplies auxiliary steam required for plant use during startup, shutdown, and normal operation. The system includes a control valve to reduce the main steam pressure, auxiliary boiler package, auxiliary steam drain tank, auxiliary steam drain pump, auxiliary steam drain monitor, auxiliary steam drain monitor heat exchanger, condensed water piping and other components.

10.4.11.2 Summary of Application

DCD Tier 1: There are no Tier 1 requirements specific for the ASSS system.

DCD Tier 2: The applicant has provided the design description and operational details for the ASSS in DCD Tier 2, Section 10.4.11, "Auxiliary Steam Supply Steam." The ASSS is supplied by the main steam system through a pressure reducing valve, when the unit is in operation and by the auxiliary boiler at all other times. A schematic of the system is depicted in DCD Tier 2 Figure 10.4.11-1

It is indicated in DCD Tier 2 Section 10.4.11.2, "General Description," that for operation as required during startup, shutdown, plant regular inspection and normal operation, auxiliary steam from the auxiliary boiler or main steam is supplied to the components continuously or intermittently. During plant normal operation, the ASSS supplies steam to the primary system components (e.g., boric acid (B/A) evaporator and B/A batching tank) and nonsafety-related HVAC equipment, by taking part of the main steam. During plant startup, shutdown, and plant regular inspections, as the main steam is not available, the auxiliary boiler provides to the primary system components, HVAC, and also to the secondary system components such as; turbine gland seal, deaerator seal, and deaerator heating.

DCD Section 10.11.5, "Instrumentation Applications," indicates that the ASSS is provided with the necessary controls and indications for local or remote monitoring of system operation, and also states that radiation monitoring is provided to monitor the leakage of radioactive materials in the condensed water from the B/A evaporator. Additionally it is stated that when the concentration of radioactive material exceeds the set point, the auxiliary steam drain tank pump discharge isolation valve is closed and auxiliary steam drain tank pump is stopped. High radiation is alarmed in the MCR.

The ASSS has incorporated design and system features to address RG 4.21 “Minimization of Contamination and Radioactive Waste Generation; Life-Cycle Planning,” and the design and system features addressing RG 4.21 are included in Section 2.3.1.3, “Onsite Meterology Measurements Program,” of the DCD.

A discussion of the radiological aspects of the system leakage is contained in DCD Section 11.1, “Source Terms,” and a summary of the design features for minimizing contamination and generation of radioactive waste is provided in DCD Table 12.3-8.

10.4.11.3 Regulatory Basis

The relevant requirements of the NRC regulations for this area of review, and the associated acceptance criteria, are summarized below:

1. GDC 4, “Environmental and dynamic effects design bases,” in that failure of the ASSS due to pipe break or malfunction of the ASSS should not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).

2. GDC 60, “Control of releases of radioactive materials to the environment,” as it relates to the ability of the ASSS system design to control releases of radioactive materials to the environment.

3. 10 CFR 20.1406, “Minimization of contamination,” as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste

10.4.11.4 Technical Evaluation

The NRC staff reviewed the ASSS described in Tier 2 of the DCD to determine if the failure or malfunction of the system could adversely affect SSCs, or result in release of radioactive materials to the environment. The staff’s acceptance of the ASSS is based on the compliance of the system design with the requirements of GDC, 4, and 60, and adherence to 10 CFR 20.1406.

The results of the staff’s review are discussed below.

GDC 4, “Environmental and dynamic effects design bases”

The staff reviewed the ASSS for compliance with the requirements of GDC 4. Compliance with the requirements of GDC 4 is based on the determination, by the staff through its review, that failure of the ASSS due to pipe break or malfunction of the ASSS does not adversely affect any of the plant’s essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).

The ASSS is a nonsafety-related system. The staff reviewed DCD Tier 2, Section 10.4.11, and DCD Tier 2 Figure 10.4.11-1, and found all the ASSS equipment and piping to be located in areas of the plant that do not contain safety-related SSC's. The ASSS components and associated piping are located in either the TB, the auxiliary building, or outside in the yard. Neither of these areas contains safety-related SSCs, therefore safety-related SSCs will not be affected due to the ASSS operation or its proximity to safety-related SSCs. Additionally, the ASSS does not interface directly with nuclear process systems, and failure of the system as a result of a pipe break or malfunction of the system does not adversely affect safety-related systems or components.

GDC 60, "Control of releases of radioactive materials to the environment,"

The staff reviewed the design of the ASSS for compliance with the requirements of GDC 60 with respect to control of releases of radioactive materials. GDC 60 requires that a means be provided to control the release of radioactive materials in liquid effluents. The requirements of GDC 60 are met if the ASSS design includes provisions to prevent excessive release of radioactivity to the environment in the event of an ASSS system, or component failure.

The ASSS provides steam to the boric acid evaporator, and thus the ASSS may potentially contain radioactive effluents. DCD Section 10.4.11.5 states, a radiation monitor is provided to monitor the leakage of radioactive material in the condensed water from the boric acid evaporators, and that when the concentration of radioactive material exceeds the setpoint, the auxiliary steam drain dischare isolation valve is closed. The staff evaluation of the radiological aspects of the system leakage is provided in Section 11.1, "Liquid Waste Management System," of this report and information pertaining to the auxiliary steam condensate water radiation monitor is provided in DCD Tier 2 Section 11.5.2.3.2. Based on the provisions described above, the staff finds that the design includes sufficient provisions to prevent excessive release of radioactivity to the environment in the event of failure of the SSC's in the ASSS.

10 CFR 20.1406, "Minimization of contamination"

10 CFR 20.1406 requires in part that each DC applicant shall describe how the facility design and procedures for operation will minimize, to extent practicable, contamination of the facility and environment, as well as the generation of radioactive waste.

System design features to provide for minimization of contamination are discussed in Tier 2, DCD Section 10.4.11.2.1. In this section, the applicant describes the condensate piping from the ASSS drain tank as a single-walled carbon steel pipe run above ground in pipe chases from the auxiliary building to the TB, and is then connected to double-walled welded carbon steel piping through the TB wall penetration to the auxiliary boiler. It is further stated in the application that since this is not a high traffic area, this segment of pipe is run above ground and is slightly sloped so that any leakage is collected in the outer pipe and drained to the auxiliary boiler building. Also in the description it is indicated that at the auxiliary boiler building end, a leak detection instrument is provided to monitor leakage, and that a drain pipe is provided to direct any drains to the building sump. In addition to the above design features, it is also indicated in the DCD that the design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.

Based on the above, the staff concludes that the ASSS design, as described in the DCD, is in compliance with 10 CFR 20.1406, since it provides for monitoring and controls of ASSS leakage.

10.4.11.4.1 Technical Specifications

There are no US-APWR TS sections for the ASSS. The system is not safety-related and is not required for safe shutdown; therefore the staff finds this acceptable.

10.4.11.4.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

There are no ITAAC required for this system. The system is not safety-related and is not required for safe shutdown; therefore the staff finds this acceptable.

10.4.11.5 Combined License (COL) Information Items

There are no COL Information items associated with this system.

10.4.11.6 Conclusions

The staff reviewed the ASSS and found the design acceptable, because as set forth above, it meets the appropriate regulatory requirements, including GDC 4, protection against missiles and effects of pipe break; and GDC 60, control of releases of radioactive materials into the environment. The staff also found that the design incorporates design features to Minimize contamination and radioactive waste as recommended by RG 4.21 and therefore meets 10 CFR 20.1406.