Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 1 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling	1. The criteria for GDC 62 are specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.1, ANSI/ANS 57.2, and ANSI/ANS 57.3, as they relate to the prevention of criticality accidents in fuel storage and handling.	Conformance with no exceptions identified.	9.1.1
9.1.2 New and Spent Fuel Storage	1. Acceptance for meeting the relevant aspect of GDC 2 is based on compliance with positions C.1 and C.2 of Regulatory Guide (RG) 1.13 and applicable portions of RG 1.29, and RG 1.117. For the spent fuel storage facility, additional guidance acceptable for meeting this criterion is found in American Nuclear Society (ANS) 57.2, 9.1.2-5 paragraphs 5.1.1, 5.1.3, 5.1.12.9, and 5.3.2. For the new fuel storage facility, additional guidance acceptable for meeting this criterion is found in ANS 57.3, paragraphs 6.2.1.3(2), 6.2.3.1, 6.3.1.1, 6.3.3.4, and 6.3.4.2. 2. Acceptance for meeting the relevant aspect of GDC 4 is based on positions C.2 and C.3 of RG 1.13, and RG 1.115 and 1.117. 3. GDC 5 is met by sharing the SSCs important to safety between the units in a manner that does not degrade the performance of their safety functions. 4. Acceptance for meeting the relevant aspect of GDC 61 for the spent fuel storage facility is based on compliance with positions C.4, C.6, C.10, C.11, and C.12 of RG 1.13 and the appropriate paragraphs of ANS 57.2. Acceptance for meeting this criterion for the new fuel storage facility is based on compliance with the appropriate paragraphs of ANS 57.3. Acceptance is also based on meeting the fuel storage capacity requirements noted in subsection III.1 of this SRP section. The following design considerations are evaluated:	Conformance with exceptions. Criterion 3 is not applicable for US-APWR design certification. (US-APWR is a single unit.)	9.1.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 2 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.2 New and Spent Fuel Storage (continued)	A. Provisions for periodic inspections of components important to safety. B. Suitable shielding for radiation protection, including adequate water levels. C. Appropriate containment and confinement systems. D. Residual heat removal capability by effective coolant flow through the storage racks for spent fuel assemblies. E. Prevention of reduction in fuel storage coolant inventory under accident conditions. 5. Acceptance for meeting the relevant aspect of GDC 63 for spent fuel storage is based on compliance with position C.7 of RG 1.13 and paragraph 5.4 of ANS 57.2. Acceptance for meeting this criterion for the dry storage of new fuel is based on radiation monitoring pursuant to 10 CFR 70.24 or acceptable prevention of an increase in effective multiplication factor (Keff) beyond safe limits as described in 10 CFR 50.68. 6. In meeting the requirements of 10 CFR 20.1101(b), positions C.2.f (2) and C.2.f (6) of RG 8.8 are the bases for acceptance with respect to provisions for decontamination. For spent fuel storage, paragraph 5.1.5 of ANS 57.2 and appropriate positions of RG 1.13 are the bases for acceptance. For new fuel storage, paragraphs 6.3.3.7 and 6.3.4 of ANS 57.3 are the bases for acceptance. 7. 10 CFR 50.68 allows the applicant to follow the guidelines of 10 CFR 70.24 for criticality monitors or the guidelines described therein for significant margins of subcriticality.		
9.1.3 Spent Fuel Pool Cooling and Cleanup System	1. General Design Criterion (GDC) 2 contained in Appendix A to 10 CFR Part 50, 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.	Conformance with no exceptions identified.	9.1.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 3 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.3 Spent Fuel Pool Cooling and Cleanup System (continued)	Acceptance for meeting this criterion is based on conformance to positions C.1, C.2, C.6, and C.8 of RG 1.13 and position C.1 of RG 1.29 for safety-related and position C.2 of RG 1.29 for nonsafety-related portions of the system. This criterion does not apply to the cleanup portion of the system and need not apply to the cooling system if the fuel pool makeup water system and its source meet this criterion, the fuel pool building and its ventilation and filtration system meet this criterion, and the ventilation and filtration system meets the guidelines of RG 1.52. The cooling and makeup system should be designed to Quality Group C requirements in accordance with RG 1.26. However, when the cooling system is not designated Category I it need not meet the requirements of ASME Section XI for inservice inspection of nuclear plant components. 2. GDC 4 with respect to the capability of the system and the structure housing the system to withstand the effects of external missiles. Acceptance is based on meeting position C.2 of RG 1.13. This criterion does not apply to the cleanup system and need not apply to the cooling water system if the makeup system, its source, the building, and its ventilation and filtration system are tornado protected, and the ventilation and filtration system 3. GDC 5 as related to shared systems and components important to safety being capable of performing required safety functions. 4. GDC 61 as related to the system design for fuel storage and handling of radioactive materials, including the following elements:meets the guidelines of RG 1.52.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 4 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.3 Spent Fuel Pool Cooling and Cleanup System (continued)	A. The capability for periodic testing of components important to safety B. Provisions for containment. C. Provisions for decay heat removal that reflects its importance to safety. D. The capability to prevent reduction in fuel storage coolant inventory under accident conditions. E. The capability and capacity to remove corrosion products, radioactive materials, and impurities from the pool water and reduce occupational exposures to radiation. 5. GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal, to detect excessive radiation levels, and to initiate appropriate safety actions. 6. 10 CFR 20.1101(b) as it relates to radiation doses being kept ALARA. In meeting this regulation, RG 8.8, positions C.2.f (2) and C.2.f (3) can be used as a basis for acceptance. 7. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (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 design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 5 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.1.3 Spent Fuel Pool Cooling and Cleanup System (continued)	8. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria 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, and the NRC's regulations. Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations are included in the Requirements subsection, above.		
9.1.4 Light Load Handling System (Related to Refueling)	1. Acceptance for meeting the relevant aspects of GDC 2 is based on RG 1.29, Positions C.1 and C.2. 2. Acceptance for meeting the relevant aspects of GDC 5 is embodied within the other acceptance criteria 3. Acceptance for meeting the relevant aspects of GDC 61 is based in part on the guidelines of American National Standards Institute/ American Nuclear Society (ANSI/ANS) 57.1-1992. 4. Acceptance for meeting the relevant aspects of GDC 62 is based in part on ANSI/ANS 57.1-1992.	Conformance with exceptions. Criterion 2 is not applicable for US-APWR design certification. (US-APWR is a single unit.)	9.1.4
9.1.5 Overhead Heavy Load Handling Systems	1. Acceptance for meeting the relevant aspects of GDC 1 is based in part on NUREG-0554 for overhead handling systems and ANSI N14.6 or ASME B30.9 for lifting devices. 2. Acceptance for meeting the relevant aspects of GDC 2 is based in part on position C.2 of RG 1.29 and Section 2.5 of NUREG-0554. 3. Acceptance for meeting the relevant aspects of GDC 4 is based in part on position C.5 of RG 1.13. 4. Acceptance for meeting the relevant aspects of GDC 5 is embodied within the other acceptance criteria.	Conformance with exceptions. Criterion 4 is not applicable for US-APWR design certification. (US-APWR is a single unit.)	9.1.5

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 6 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.1 Station Service Water System	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the EESWS and the ESWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the ESWS and Position C.2 for nonsafety-related portions of the ESWS are appropriately addressed. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the EESWS, are met: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. In addition, the information will be considered acceptable if the design provisions presented in GL 96-06 and to GL 96-06, Supplement 1 are appropriately addressed. 3. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the ESWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s). In addition, the information will be considered acceptable if the provisions GL 89-13 and GL 91-13 are appropriately addressed.	Conformance with no exceptions identified.	9.2.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 7 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.1 Station Service Water System (continued)	4. Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the ESWS can transfer the combined heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised. 5. Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the ESWS permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components. 6. Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the ESWS is designed for testing to detect degradation in performance or in the system pressure boundary so that the ESWS will function reliably to provide decay heat removal and essential cooling for safety-related equipment.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 8 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.2 Reactor Auxiliary Cooling Water Systems	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the reactor auxiliary CWS and the reactor auxiliary CWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the reactor auxiliary CWS and Position C.2 for nonsafety-related portions of the reactor auxiliary CWS are appropriately addressed. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the reactor auxiliary CWS, are met: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. In addition, the information will be considered acceptable if the design provisions presented in GL 96-06 and GL 96-06, Supplement 1 are appropriately addressed. 3. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the reactor auxiliary CWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).	Conformance with no exceptions identified.	9.2.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 9 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.2 Reactor Auxiliary Cooling Water Systems (continued)	4. Cooling Water System. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if the reactor auxiliary CWS and its components will continue to perform their required safety functions, assuming a single, active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 and to seismic Category I, Quality Group C, and American Society of Mechanical Engineers (ASME) Section III Class 3 requirements concurrent with the loss of offsite power. In addition, the information will be considered acceptable based on appropriate application of IEEE Std 603, as endorsed by RG 1.153, and appropriate application of RG 1.155, Position C.3.3.4. 5. Cooling Water System Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the periodic inspection of important reactor auxiliary CWS components ensures system integrity and capability to perform design safety functions. 6. Cooling Water System Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if periodic system pressure and function testing of the reactor auxiliary CWS will ensure the leak-tight integrity and operability of its components, as well as the operability of the system as a whole, at conditions as close to the design basis as practical		
9.2.3 (Withdrawn) Demineralized Water Makeup System	This SRP has been withdrawn.	Not applicable. SRP has been withdrawn by NRC.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 10 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.4 Potable and Sanitary Water Systems	1. Control of Releases of Radioactive Materials to the PSWS. Information that addresses the requirements of GDC 60 in regards to controlling radioactive effluent releases is considered acceptable if the following are met: A. There are no interconnections between the PSWS and systems having the potential for containing radioactive material. B. The potable water system is protected by an air gap, where necessary. C. An evaluation of potential radiological contamination, including accidental, and safety implications of sharing (for multi-unit facilities) indicates that the system will not result in contamination beyond acceptable limits.	Conformance with exceptions. Safety implications of sharing (for multi-unit facilities) of criteria 1C is N/A.)	9.2.4, 9.2.5
9.2.5 Ultimate Heat Sink	1. GDC 2 as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods. 2. GDC 5 as to capability of shared systems and components important to safety to perform required safety functions. 3. GDC 44 as to: A. The capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions. B. Suitable component redundancy so that safety functions can be performed assuming a single, active component failure coincident with loss of offsite power. C. The capability to isolate components, systems, or piping if required so safety functions are not compromised. 4. GDC 45 as to the design provisions to permit inservice inspection of safety-related components and equipment. 5. GDC 46 as to the design provisions to permit operation functional testing of safety related systems or components.	Conformance with no exceptions identified.	9.2.5, 14.3, also Tier 1 section 2.7 for ITAAC

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 11 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.5 Ultimate Heat Sink (continued)	6. 10 CFR 52.47(b) (1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (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 design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. 7. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria 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, and the NRC's regulations.		
9.2.6 Condensate Storage Facilities	1. Protection Against Natural Phenomena. Acceptance for meeting the relevant aspects of GDC 2 is based in part on meeting the guidance of Position C.1 of Regulatory Guide 1.29 if any portion of the system is deemed to be safety related and the guidance of Position C.2 for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of Regulatory Guide 1.117 with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of Regulatory Guide 1.102 with respect to identifying portions of the system that should be protected from flooding. 2. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CSF in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).	Not applicable. Condensate Storage Facilities has no safety related functions. US-APWR is not multiple – unit.	9.2.6

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SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.2.6 Condensate Storage Facilities (continued)	3. Condensate Storage Facility. Information that addresses the requirements of GDC 44 regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the CSF can transfer the combined heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised. 4. Condensate Storage Facility Inspection. Information that addresses the requirements of GDC 45 regarding the inspection of cooling water systems will be considered acceptable if the design of the CSF permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components. 5. Condensate Storage Facility Testing. Information that addresses the requirements of GDC 46 regarding the testing of cooling water systems will be considered acceptable if the CSF is designed for testing to detect degradation in performance or in the system pressure boundary so that the CSF will function reliably to provide decay heat removal and essential cooling for safety-related equipment. 6. Control of Radioactive Releases to the Environment. Acceptance for meeting the relevant aspects of GDC 60 is based on meeting the guidance of Regulatory Guide 1.143. 7. Loss of All Alternating Current Power. Acceptance for meeting the relevant aspects of 10 CFR 50.63 is based on meeting the guidance of Regulatory Guide 1.155.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 13 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.1 Compressed Air System	1. Acceptance for meeting the relevant aspect of GDC 1 is based on compliance with the criteria specified in American National Standards Institute/Instrument Society of America (ANSI/ISA) S7.3-R1981 related to minimum instrument air quality standards. 2. Acceptance for meeting the relevant requirements of GDC 2 as it relates to seismic classification is based on compliance to guidance provided in RG 1.29, Positions C.1 and C.2. 3. Acceptance for meeting the relevant requirements of GDC 5 as it relates to the sharing of safety-related SSCs is based on the criteria set forth here for CAS SSCs shared among multiple units. 4. Acceptance for meeting the relevant requirements of 10 CFR 50.63 as it relates to the CAS design and the ability of a plant to withstand for a specified duration and recover from a station blackout is based on RG 1.155.	Conformance with exceptions. Criterion 3, the instrument air system of the US-APWR is not shared. Criterion 4, US-APWR can cope with a station blackout [SBO] without air supply from the instrument air system.	9.3.1
9.3.2 Process and Post-Accident Sampling Systems	1. The applicant's design is such that the PSS has the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of Regulatory Guide (RG) 1.21, Position C.2, the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines, and the Electric Power Research Institute (EPRI) PWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC. 2. The plant Technical Specifications include the required analysis and frequencies. 3. The following guidelines should be used to determine the acceptability of the PSS functional design: A. Provisions should be made to ensure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet these criteria.	Conformance with no exception identified. (DCD Chapter16 should include the required analysis and frequencies per criteria 2.)	9.3.2

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SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.2 Process and Post-Accident Sampling Systems (continued)	B. Provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet this criterion. C. Provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of Regulatory Position C.7 in RG 1.21 are followed to meet this criterion. D. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA levels. The guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8 are followed to meet this criterion. E. Isolation valves should fail in the closed position, in accordance with the requirements of GDC 60 to control the release of radioactive materials to the environment. F. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures to ALARA levels and the requirements of GDC 60 to control the release of radioactive materials to the environment. The guidelines of Regulatory Position 2.i. (6) in RG 8.8 should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 15 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.2 Process and Post-Accident Sampling Systems (continued)	4. To meet the requirements of GDCs 1 and 2, the applicant's seismic design and quality group classification of sampling lines, components, and instruments for the PSS should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification), in accordance with Regulatory Positions C.1, C.2, and C.3 in RG 1.26; Regulatory Positions C.1, C.2, C.3, and C.4 in RG 1.29, and the guidelines of RG 1.97. Components and piping downstream of the second isolation valve may be designed to Quality Group D and nonseismic Category I requirements, in accordance with Regulatory Position C.3 in RG 1.26.		
9.3.3 Equipment and Floor Drainage System	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of safety-related system portions of the EFDS to withstand the effects of natural phenomena. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section. If no portion is safety-related, the EFDS need not meet GDC 2. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding the capability to withstand the effects of and to be compatible with the environmental conditions (flooding) of normal operation, maintenance, testing, and postulated accidents (pipe break, tank ruptures) will be considered acceptable if the EFDS is designed to prevent flooding that could affect SSCs important to safety (i.e., necessary for safe shutdown, accident prevention, or accident mitigation) adversely.	Conformance with no exceptions identified.	9.3.3

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SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.3 Equipment and Floor Drainage System (continued)	3. Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of GDC 60 regarding the suitable control of the release of radioactive materials in liquid effluent, including anticipated operational occurrences will be considered acceptable if the EFDS is designed to prevent the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal.		
9.3.4 Chemical and Volume Control System (PWR) (Including Boron Recovery System)	1. The CVCS safety-related functional performance should be maintained in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. For compliance with GDC 29, 33 and 35, the CVCS should provide sufficient pumping capacity to supply borated water to the RCS, maintain RCS water inventory within the allowable pressurizer level range for all normal modes of operation, and function as part of the ECCS, if so designed, to supply reactor coolant makeup in the event of small pipe breaks assuming a single active failure coincident with the loss of offsite power. In addition, Regulatory Guide 1.155 describes a means acceptable to the NRC staff for meeting the requirements of 10 CFR 50.63, “Loss of all ac/AC power.” If the CVCS is necessary to support a plant SBO coping capability as required by 10 CFR 50.63, the positions in Regulatory Guide 1.155 regarding CVCS design provide an acceptable method for showing compliance. 2. SECY-77-439 describes the concept of single failure criteria and the application of the single failure criterion that involves a systematic search for potential single failure points and their effects on prescribed missions. Application of the single failure assumption in system design and analysis provides redundancy and defense-in-depth to ensure functional performance of the CVCS. Also, the requirements of GDC 5 prohibiting the sharing among nuclear units the SSCs important to safety would be met by the use of a separate CVCS for each unit.	Conformance with no exceptions identified.	9.3.4

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SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.4 Chemical and Volume Control System (PWR) (Including Boron Recovery System) (continued)	3. 10 CFR 50.55(a) requires that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III of the ASME Boiler and Pressure Vessel Code or equivalent quality standards. Regulatory Guide 1.26 describes a quality classification system that may be used to determine quality standards acceptable to the NRC staff for satisfying GDC 1 for other safety related components containing water, steam, or radioactive materials in light-water-cooled nuclear power plants. RG 1.29 describes a method acceptable to the NRC staff for identifying and classifying those features of LWRs that should be designed to withstand the effects of the safe shutdown earthquake (SSE). The requirements of GDC 1 regarding the quality standard are met by acceptable application of quality group classifications and application of quality standards as described in RG 1.26. The requirement of GDC 2 regarding the protection against natural phenomena are met by meeting the guidance of RG 1.29, Position C.1, for safety-related portions of the system and Position C.2 for nonsafety-related portion. 4. The CVCS design and arrangement should be that all components and piping that can contain boric acid will either be heat traced or will be located within heated rooms to prevent precipitation of boric acid. As additional specific criteria used to review the CVCS and BRS design, the CVCS should include provisions for monitoring: (a) temperature upstream of the demineralizer to assure that resin temperature limits are not exceeded, and (b) filter demineralizer differential pressure to assure that pressure differential limits are not exceeded. In addition, the CVCS should have provision for automatically diverting or isolating the CVCS flow to the demineralizer in the event the demineralizer influent temperature exceeds the resin temperature limit.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 18 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.3.4 Chemical and Volume Control System (PWR) (Including Boron Recovery System) (continued)	5. 10 CFR 50.34(f)(2)(xxvi), as applicable, specifies the provisions regarding detection of reactor coolant leakage outside containment. These requirements will be met, in part, by providing leakage control and detection systems in the CVCS and implementation of appropriate leakage control program. 6. Implementation of Action 1 specified in Bulletin 80-05 provides an acceptable means for the system to prevent the CVCS holdup tanks, which can contain radioactive release, from the formation of such vacuum conditions that could cause wall inward buckling and failure. The requirements of GDC 60 and 61 can be met, in part, by providing in the CVCS appropriately designed venting and draining closed systems to confine the radioactivity associated with the effluents. 7. 10 CFR 52.47(a)(1)(vi) specifies that the application of a design certification should contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the design certification. 10 CFR 52.97(b)(1) specifies that the COL identifies the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(a)(1)(vi) and 10 CFR 52.97(b)(1) will be met, in part, by identifying inspections, tests, analyses, and acceptance criteria of the top-level design features of the CVCS in the design certification application and the combined license, respectively.		
9.3.5 Standby Liquid Control System (BWR)	This SRP applies to boiling water reactors (BWRs) and is not applicable to the US-APWR.	Not applicable. The SRP applies to BWRs only.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 19 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.1 Control Room Area Ventilation System	1. Protection Against Natural Phenomena. Information that addresses the requirements of GDC 2 regarding the capability of structures housing the CRAVS and the CRAVS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions of the CRAVS and Position C.2 for nonsafety-related portions of the CRAVS are appropriately addressed. 2. Environmental and Dynamic Effects. Information that addresses the requirements of GDC 4 regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the CRAVS, are met: SRP Sections 3.5.1.1, 3.5.2, and 3.6.1. 3. Sharing of SSCs. Information that addresses the requirements of GDC 5 regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CRAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s). 4. Control Room. Information that addresses the requirements of GDC 19 regarding the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents will be considered acceptable if adequate protection against radiation and hazardous chemical releases are provided to permit access to and occupancy of the control room under accident conditions. RG 1.78 provide guidance acceptable to the staff for meeting these control room occupancy protection requirements.	Conformance with exceptions. Criteria 1, 2, 5, 6: Conformance with no exceptions identified. Criterion 3: Not applicable to US-APWR design certification (Not multiple unit plants) Criterion 4: The postulated hazardous chemical release is COLA Specific.	9.4.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 20 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.1 Control Room Area Ventilation System (continued)	5. Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of GDC 60 regarding the suitable control of the release of gaseous radioactive effluents to the environment will be considered acceptable if the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants are appropriately addressed. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3. 6. Loss of All Alternating Current Power. Information that addresses the requirements of 10 CFR 50.63 regarding the necessary support systems providing sufficient capacity and capability for coping with a station blackout event will be considered acceptable if the guidance of RG1.155, including position C.3.2.4 is applied appropriately.		
9.4.2 Spent Fuel Pool Area Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions. 2. For GDC 5, acceptance is based on the determination that the use of the SFPAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).	Conformance with exceptions. Criterion 2 is N/A.(Not multiple unit plants) Criterion 3 is N/A. (Not air .cleanup system) Criterion 4 is N/A.(satisfy the limit offsite dose consequences from fuel handling area without ESF ventilation (filtration) system.)	9.4.2

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 21 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.2 Spent Fuel Pool Area Ventilation System (continued)	3. For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3. 4. For GDC 61, acceptance is based on the guidance of RG 1.13 as to the design of the ventilation system for the spent fuel storage facility, Position C.4.		
9.4.3 Auxiliary and Radwaste Area Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions, and Position C.2 for nonsafety-related portions. 2. For GDC 5, acceptance is based on the determination that the use of the ARAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s). 3. For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.	Conformance with exceptions. Criterion 2:Not multiple unit plants Criterion 3: Air clean up function is provided for TSC HVAC system only.	9.4.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 22 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.4 Turbine Area Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions. 2. For GDC 5, acceptance is based on the determination that the use of the TAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s) 3. For GDC 60, acceptance is based on guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 Revision 2, the applicable regulatory position is C.2. For RG 1.52 Revision 3, the applicable regulatory position is C.3. For RG 1.140 Revision 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 Revision 2, the applicable regulatory positions are C.2 and C.3.	Not applicable. Criterion 1: Turbine Building Area Ventilation System does not need design and manufacture considered SSE because the failure of Turbine Building Area Ventilation System has no influence on safety-related portion and the main control room comfort. Criterion 2: The standard design of US-APWR is single unit and will not be system sharing basically even in case of multiple-unit. Criterion 3: The filter system required is not installed in Turbine Building because Turbine Building has not possibility of contamination by radio-active particle.	9.4.4
9.4.5 Engineered Safety Feature Ventilation System	1. For GDC 2, acceptance is based on the guidance of RG 1.29, Position C.1, for safety-related portions and Position C.2 for nonsafety-related portions. 2. For GDC 4, acceptance is based on meeting the acceptance criteria in the following SRP sections, as they apply to the ESFVS: SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1. 3. For GDC 5, acceptance is based on the determination that the use of the ESFVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s). 4. For GDC 17, acceptance is based on the guidance of item 2 under Subsection A and item 1 under Subsection C of the NUREG-CR/0660 section “Recommendations” for protection of essential electrical components from failure due to the accumulation of dust and particulate materials.	Conform with exceptions Criterion 3: Not multiple unit plants. Criterion 4: Gas turbine has own cooling system. Criterion 5: Air cleanup function is provided for annulus exhaust system only.	9.4.5

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 23 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.4.5 Engineered Safety Feature Ventilation System (continued)	5. For GDC 60, acceptance is based on the guidance of RGs 1.52 and 1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3. 6. For 10 CFR 50.63, acceptance is based on the applicable guidance of RG 1.155, including Position C.3.2.4.		
9.5.1 Fire Protection Program	1. RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," as it applies to the use of PRA in support of changes to the fire protection licensing basis for nuclear power plants. Appropriate techniques for performing a Fire PRA are presented in NUREG/CR-6850 (EPRI TR-1011989), "EPRI/ NRC-RES Fire PRA Methodology for Nuclear Power Facilities." 2. RG 1.188, Revision 1, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," as it applies to FPP considerations for license renewal such as equipment aging issues. This RG endorses the guidance in Nuclear Energy Institute (NEI) document, NEI 95-10, Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule." 3. RG 1.189, Revision 1, "Fire Protection for Nuclear Power Plants," which provides comprehensive staff positions and guidelines on fire protection for nuclear power plants. 4. RG 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," which establishes the fire protection objectives and staff positions for implementing fire protection for those nuclear power plants that have submitted the necessary certifications for license termination under 10 CFR Part 50.82(a).	Conformance with exceptions. Some information of the Fire Protection Program such as the fire protection organization; administrative policies; maintenance, and QA is provided in COLA.	9.5.1

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 24 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.1 Fire Protection Program (continued)	5. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it applies to the FPP of any new reactor COL application submitted in accordance with 10 CFR Part 52. 6. Enhanced fire protection criteria for new reactor designs as documented in SECY 90-016, SECY 93-087, and SECY 94-084. SECY 90-016 established enhanced fire protection criteria for evolutionary light water reactors. SECY 93-087 recommended that the enhanced criteria be extended to include passive reactor designs. SECY 90 016 and SECY 93-087 were approved by the Commission in staff requirements memoranda (SRM). SECY 94-084, in part, establishes criteria defining safe-shutdown conditions for passive light water reactor designs. 7. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the fire protection program are reviewed in accordance with 10 50.48. The operational program for fire protection should be fully implemented prior to fuel receipt at the plant site.		
9.5.2 Communications Systems	1. Information regarding the requirements of Appendix E to 10 CFR Part 50, Part IV.E(9), will be found acceptable if adequate provisions are made and described for emergency facilities and equipment, including: at least one onsite and one offsite communications system; each system shall have a backup power source. 2. For those applicants subject to either 10 CFR 50.34(f) or the TMI Action Plan, information regarding the requirements of 10 CFR 50.34(f) (2)(xxv) and TMI Action Plan Item III A.1.2 will be found acceptable if provisions are made for an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. 3. Information regarding the requirements of 10 CFR 50.47(a) (8) will be found acceptable if adequate emergency facilities and equipment to support the response are provided and maintained.	Conform with exceptions. Criteria #1, #2, #3, #9, #12, #13, and #14 refer to site-specific emergency response and security requirements that will be the responsibility of the COL Applicant. As indicated in section 9.5.2, details of the security communication system design and procedures are the responsibility of the COL Applicant.	9.5.2

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SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	4. Information regarding the requirements of 10 CFR 50.55a will be found acceptable if SSCs are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. 5. Information regarding the requirements of GDC 1 will be found acceptable if SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit. 6. Information regarding the requirements of GDC 2 will be found acceptable if SSCs important to safety are 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. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 26 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	7. Information regarding the requirements of GDC 3 will be found acceptable if SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs. 8. Information regarding the requirements of GDC 4 will be found acceptable if SSCs important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. 9. Information regarding the requirements of GDC 19 will be found acceptable if equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls (I&C) to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 27 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	10. Information regarding the requirements of 10 CFR 73.45(e)(2)(iii) will be found acceptable if communications subsystems and procedures are provided for notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material so that response can be such as to prevent the removal and satisfy the general performance objective and requirements of § 73.20(a). 11. Information regarding the requirements of 10 CFR 73.45(g)(4)(i) will be found acceptable if communications networks are provided to transmit rapid and accurate security information among onsite forces for routine security operation, assessment of a contingency, and response to a contingency. 12. Information regarding the requirements of 10 CFR 73.46(f) will be found acceptable if each guard, watchman, or armed response individual on duty shall be capable of maintaining continuous communication with an individual in each continuously manned alarm station required by 10 CFR 73.46(e)(5), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from law enforcement authorities; each alarm station required by 10 CFR 73.46(e)(5) shall have both conventional telephone service and radio or microwave transmitted two-way voice communication, either directly or through an intermediary, for the capability of communication with the law enforcement authorities; and non-portable communications equipment controlled by the licensee and required by 10 CFR 73.46(f) shall remain operable from independent power sources in the event of the loss of normal power.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 28 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	13. Information regarding the requirements of 10 CFR 73.55(e) will be found acceptable if all alarms required by 10 CFR 73.55 annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station not necessarily onsite, so that a single act cannot remove the capability of calling for assistance or otherwise responding to an alarm. The onsite central alarm station must be considered a vital area and its walls, doors, ceiling, floor, and any windows in the walls and in the doors must be bullet-resisting. The onsite central alarm station must be located within a building in such a manner that the interior of the central alarm station is not visible from the perimeter of the protected area. This station must not contain any operational activities that would interfere with the execution of the alarm response function. Onsite secondary power supply systems for alarm annunciator equipment and non-portable communications equipment as required 10 CFR 73.55(f) of this section must be located within vital areas. All alarm devices including transmission lines to annunciators shall be tamper indicating and self-checking, e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when the system is on standby power. The annunciation of an alarm at the alarm stations shall indicate the type of alarm (e.g., intrusion alarms, emergency exit alarm, etc.) and location. All emergency exits in each protected area and each vital area shall be alarmed.		

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 29 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.2 Communications Systems (continued)	14. Information regarding the requirements of 10 CFR 73.55(f) will be found acceptable if each guard, watchman or armed response individual on duty is capable of maintaining continuous communication with an individual in each continuously manned alarm station required by 10 CFR 73.55(e)(1), who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from local law enforcement authorities. The alarm stations required by 10 CFR 73.55(e)(1) shall have conventional telephone service for communication with the law enforcement authorities as described in 10 CFR 73.55(f)(1). To provide the capability of continuous communication, radio or microwave transmitted two-way voice communication, either directly or through an intermediary, shall be established, in addition to conventional telephone service, between local law enforcement authorities and the facility and shall terminate in each continuously manned alarm station required by 10 CFR 73.55(e)(1). Non-portable communications equipment controlled by the licensee and required by 10 CFR 73.55 shall remain operable from independent power sources in the event of the loss of normal power.		
9.5.3 Lighting Systems	1. Acceptance criteria of the design of the normal and emergency lighting systems, as described in the applicant's safety analysis report (SAR), is based in part on the degree of similarity of the systems design with those for previously reviewed plants with satisfactory operating experience. 2. The normal lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate station lighting in all areas, from power sources described in Section 8.2 of the SRP that are required for control and maintenance of equipment and plant access routes during normal plant operations.	Conformance with no exceptions identified	9.5.3

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 30 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.3 Lighting Systems (continued)	3. The emergency lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate emergency station lighting in all areas, required for fire fighting, control and maintenance of equipment used for implementing safe shutdown of the plant during all plant operating conditions, and the access routes to and from these areas. 4. The lighting systems designs will be acceptable if they conform to the lighting levels recommended in NUREG-0700, which is based on the Illuminating Engineering Society of North America (IESNA) Lighting Handbook (Reference 2) as related to systems design and illumination levels recommended for industrial facilities.		
9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System	1. GDC 2 requirements for which SSCs must be protected from…. 2. GDC 4 requirements for which SSCs must be protected from…Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section. 3. GDC 5 requirements for sharing of SSCs important to safety … 4. GDC 17 requirements for the capability of the cooling water system … 5. GDC 44 requirements are met when the EDECWS has… 6. GDC 45 as to design provisions for periodic inspection… 7. GDC 46 as to design provisions for appropriate functional testing…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	9.5.4
9.5.5 Emergency Diesel Engine Cooling Water System	1. GDC 2 requirements for which SSCs must be protected from…. 2. GDC 4 requirements for which SSCs must be protected from…. 3. GDC 5 requirements for sharing of SSCs important to safety… 4. GDC 17 requirements for the capability of the cooling water system… 5. GDC 44 requirements are met when the EDECWS has… 6. GDC 45 as to design provisions for periodic inspection… 7. GDC 46 as to design provisions for appropriate functional testing…	Not applicable. Emergency power will be provided for US-APWR by gas turbine generators in lieu of diesel generators. The gas turbine generators have no functional equivalent of a cooling water system.	N/A

Table 1.9.2-9 US-APWR Conformance with Standard Review Plan Chapter 9 Auxiliary Systems (Sheet 31 of 31)

SRP Section and Title	SRP Excerpt Indicating Acceptance Criteria for DCD	Status	Appears in DCD Chapter/Section
9.5.6 Emergency Diesel Engine Starting System	1. GDC 2 requirements for SSCs to withstand or be protected from… 2. GDC 4 requirements for SSCs to be protected against… 3. GDC 5 requirements for sharing of SSCs important to safety 4. GDC 17 as to the capability of the diesel engine air starting system…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	9.5.6
9.5.7 Emergency Diesel Engine Lubrication System	1. GDC 2 requirements for SSCs to withstand or be protected… 2. GDC 4 requirements for SSCs to be protected against… 3. GDC 5 requirements for sharing of SSCs important to safety 4. GDC 17 requirements of independence and redundancy criteria…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design..	9.5.7
9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System	1. GDC 2 requirements for SSCs to withstand or be protected from… 2. GDC 4 requirements of SSCs to be protected against 3. GDC 5 requirements for sharing of SSC important to safety… 4. GDC 17 as related to the capabilities of the diesel engine combustion…	Conformance with no exception identified. US-APWR has no diesel generators, but uses gas turbine generators for emergency power in the standard design.	9.5.8

Table 3.2-3 Comparison of Various Requirements to Equipment Class

US-APWR Equipment Class	ASME Code, Section III (Reference 3.2-14), Class	RG1.29 (Reference 3.2-5) Seismic Category	RG1.26 (Reference 3.2-13) NRC Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)
1	1[1]	I	A	YES[2]
2	2[1]	I	B	YES[2]
3	3[1]	I	C	YES[2]
4	N/A[3]	NS or II	D	N/A[4]
5	N/A[5]	NS or II	N/A	N/A[4]
6	N/A[6]	N/A[7]	N/A	N/A[8]
7	N/A[9]	N/A[10]	N/A	N/A[11]
8	N/A	NS	D	N/A
9	N/A	NS	N/A	N/A
10	N/A	NS	N/A	N/A

Notes:

1. Items not covered by the ASME Code are designed to other applicable codes and standards.
2. "Yes" means QA Program is required according to 10 CFR 50, Appendix B (Reference 3.2-8).
3. Refer to Subsection 3.2.2.4.
4. Seismic category II SSCs meet the pertinent QA requirements of 10 CFR 50, Appendix B. (Refer to Subsection 3.2.1.1.2)
5. Code and standard as defined in design bases are applied.
6. Code and standard meeting RG 1.143 (Reference 3.2-10) are applied.
7. Seismic category meeting RG 1.143 (Reference 3.2-10) is applied.
8. A QA program meeting RG 1.143 (Reference 3.2-10) is applied.
9. Code and standard meeting RG 1.189 (Reference 3.2-11) is applied.
10. Seismic category meeting RG 1.189 (Reference 3.2-11) is applied.
11. A QA Program meeting RG 1.189 (Reference 3.2-11) is applied.

Table 3.2-4 Seismic Classification of Buildings and Structures[1]

Structure	Acronym	Seismic Category[2]
Reactor Building[3]	R/B	I
Prestressed Concrete Containment Vessel[3]	PCCV	I
Containment Internal Structure[3]		I
Power Source Building (East and West)[3]	PS/B	I
Power Source Fuel Storage Vault	PSFSV	I
Essential Service Water Pipe Tunnel (ESWPT) (from/to UHS)[5]	ESWPT	I
UHS Related Structures[4]	UHSRS	I
A/B[3]	A/B	II
Turbine Building	T/B	II
AC/B[3]	AC/B	NS
Outside Building (e.g., maintenance facility, operations office)	O/B	NS
Turbine generator pedestal	T/G Pedestal	NS

Notes:

1. Other non-standard plant building structures, such as minor NS buildings and structures in the plant yard, are not listed in the above table and are not considered part of the US-APWR Nuclear Island.

2. Seismic category I (I)
   Seismic category II (II)
   Non-Seismic (NS)

3. US-APWR Nuclear Island

4. UHSRS include but are not limited to (1) dams, (2) ponds, or (3) cooling towers (including cooling tower enclosure, and pump house). The specific features of the UHSRS are site dependent and not part of the US-APWR standard plant. The UHSRS are seismic category I structures selected based on site-specific conditions and site-specific meteorological data.

5. The ESWPT is a site-specific structure, but the existence and functions are required by the plant standard design. The specific features of the ESWPT are site dependent and will depend on the type of UHS.

Table 3.6-1 High and Moderate Energy Fluid Systems

System	High-Energy(1)	Moderate-Energy(1)
Reactor Coolant System (RCS)	X	-
Chemical and Volume Control System (CVCS)	X	-
Safety Injection System (SIS)	X	-
Residual Heat Removal System (RHRS) (2)	-	X
Emergency Feedwater System (EFWS) (2)	-	X
Feedwater System (FWS)	X	-
Main Steam Supply System (MSS)	X	-
Containment Spray System (CSS)	-	X
Component Cooling Water System	-	X
Spent Fuel Pit Cooling and Purification System (SFPCS)	-	X
Essential Service Water System (ESWS)	-	X
Gaseous Waste Management System (GWMS)	-	X
Liquid Waste Management System (LWMS)	-	X
Solid Waste Management System (SWMS)	-	X
Sampling System (SS)	X	-
Steam Generator Blowdown System (SGBDS)	X	-
Refueling Water Storage System (RWS)	-	X
Primary Makeup Water System (PMWS)	-	X
Auxiliary Steam Supply System (ASSS)	X	-
Instrument Air System (IAS)	-	X
Fire Protection Water Supply System (FSS)	-	X
Station Service Air System (SSAS)	-	X
Chilled Water System (VCWS)	-	X

Notes

1. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig.

   Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy are considered as moderate-energy.

   Piping systems that exceed 200°F or 275 psig for two percent or less of the time during which the system is in operation are considered moderate-energy.

2. The RHRS and EFWS lines are classified as moderate-energy based on the 2 percent rule. These lines experience high-energy conditions for less than 2 percent of the system operation time. The portions of the RHR system from the connections to the RCS to the first closed valve in each line are high-energy.

3. If necessary, RCS setpoints are adjusted until optimal response is obtained.

D. Acceptance Criterion

1. The control systems maintain turbine speed, reactor power, RCS temperature, pressurizer level and pressure, and SG levels and pressures without causing a reactor or turbine trip, or lift primary or secondary safety valves during, or following, the transient operation.

14.2.12.2.4.20 Dynamic Response Test

A. Objectives

1. To verify during power range testing that stress analysis of essential NSSS and balance of plant components under transient conditions is in accordance with design.

2. Points are tested to resolve discrepancies from hot functional testing, to test modifications made since hot functional testing, and to test systems not tested during hot functional testing.

B. Prerequisites

1. Temporary instrumentation is installed, as required, to monitor the deflections of components under test.

2. Points are monitored and baseline data are established.

C. Test Method

1. Deflection measurements are recorded during various plant transients.

D. Acceptance Criteria

1. The movements due to flow-induced loads do not exceed the stress analysis of the monitored points.

2. Flow-induced movements and loads do not cause malfunctions of plant equipment or instrumentation.

14.2.12.2.4.21 Ultimate Heat Sink Heat Rejection Capability Test

A. Objectives

1. To determine the heat rejection capabilities of the ESWS to the ultimate heat sink while operating under partial heat load conditions.

2. To evaluate the heat rejection capabilities of the ESWS to the ultimate heat sink determined under test conditions and demonstrate that they meet the design requirements.

B. Prerequisites

1. The ESWS is operable.

2. The plant is operated in a manner to provide the ESWS with a heat load.

C. Test Method

1. Heat rejection capability of ESWS to the ultimate heat sink is measured in accordance with design requirements.

D. Acceptance Criteria

1. The heat rejection capability of the ESWS to the ultimate heat sink meets design requirements.

2. The heat rejection capability of two operating and four operating ESWS trains are verified.

14.2.12.2.4.22 Automatic High Power SG Water Level Control Test

A. Objective

1. To verify the stability of the automatic high power SG water level control system following simulated transients at 30% power conditions and the operation of the variable speed feature of the feedwater pumps.

B. Prerequisites

1. The reactor is critical, and in the 30% power level.

2. The high power SG water level control system is checked and calibrated.

3. Steam generator alarm setpoints are set for each SG.

C. Test Method

1. Induce simulated SG level transients to verify high power SG water level control response.

2. Verify the variable speed features of the main feedwater pumps by manipulation of controllers and test input signals.

D. Acceptance Criteria

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water System (ESWS)

LCO 3.7.8 Three ESWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ESWS train inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ESWS. ---------- Restore three ESWS trains to OPERABLE status.	72 hours
	[OR	
	A.2 ----------NOTE---------- This Required Action is not applicable in MODE 4. ---------- Apply the requirements of Specification 5.5.18.	72 hours]
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	----------NOTE---------- Isolation of ESWS flow to individual components does not render the ESWS inoperable. ---------- Verify each ESWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	[31 days OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.8.2	Verify each ESWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. The motor operated valve provided at the discharge of each pump opens automatically after starting the ESW pump. This interlock prevents the pump from starting if the valve is not closed. The closed discharge valve opens after starting the ESWP.	[24 months OR In accordance with the Surveillance Frequency Control Program]
SR 3.7.8.3	Verify each ESWS pump starts automatically on an actual or simulated actuation signal.	[24 months OR In accordance with the Surveillance Frequency Control Program]

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 [[Three]] UHS [[cooling towers]] shall be OPERABLE [[including their associated fans and three OPERABLE transfer pumps.]]

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. [[One required cooling tower with associated cooling tower fans inoperable.]]	A.1 [[Restore three cooling towers with associated fans to OPERABLE status.]] OR A.2 ----------NOTE---------- This Required Action is not applicable in MODE 4. ---------- Apply the requirements of Specification 5.5.18.	[[72 hours]] [[72 hours]]
B. [[One or more required]] UHS [[basins]] with water temperature not within limits.	B.1 Verify that water temperature of the UHS is [[≤93°F]] averaged over the previous 24 hour period.	Once per hour
C. One or more required UHS [[basins]] with water level not within limits.	C.1 Restore water level(s) to within limits.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. [[One or more required UHS transfer pump(s) inoperable.]]	D.1 [[Restore the transfer pump(s) to OPERABLE status.]] [[OR D.2.1 Implement an alternate method of basin transfer.]] [[AND D.2.2 Restore the transfer pump(s) to OPERABLE status]]	[[7 days]] [[7 days]] [[31 days]]
E. Required Action and associated Completion Time of Condition [[A, B, C, or D]] not met. [[OR UHS inoperable for reasons other than Condition A, B, C, or D.]]	E.1 Be in MODE 3. AND E.1 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify [[each]] required UHS [[basin]] water inventory is [[≥ 2,800,000 gallons]].	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.2	Verify water temperature of UHS is [[≤ 93°F]].	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.9.3	[[Operate each cooling tower fan for ≥ 15 minutes.]]	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.4	[[Verify each cooling tower fan starts automatically on an actual or simulated actuation signal.]]	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.5	[[Verify each UHS transfer pump starts on manual actuation.]]	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.6	Verify each UHS manual, power-operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.9.7	Verify each UHS automatic valve and each control valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Service Water System, and thus to the environment.

A typical CCW System is arranged as four independent, 50% capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes one 50% capacity pump, connection to one of the two surge tanks, a 50% capacity heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. The surge tanks in the system provide pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components are isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in Chapter 9 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Containment Spray/Residual Heat Removal (CS/RHR) System. This may be during a normal or post accident cooldown and shutdown. CCWS cooling to the four RCP seal thermal barriers is used for all operating modes (including accident and safe shutdown) to preclude a RCP seal LOCA in the event that CVCS is unavailable to provide required flow to the RCP seal via seal injection. Manual alignment of RCP thermal barrier cooling is achieved via the CCWS RCP cross-tie valves from the MCR in the event two CCWS trains are unavailable to supply CCWS to a pair of RCP thermal barriers. DCD_09.02.02-84

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for two CCW trains to remove the post loss of coolant accident (LOCA) heat load from the refueling water storage pit and other components, such as Safety Injection Pumps and CS/RHR Pumps. The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCW System, respectively. The normal temperature of the CCW is 100°F, and, during unit cooldown to MODE 5 (T_{cold} < 200°F), a maximum temperature of 110°F is assumed. This prevents the refueling water storage pit fluid from increasing in temperature following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS).

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water System (ESWS)

BASES

BACKGROUND

The ESWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the ESWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The ESWS consists of four separate, safety related, cooling water trains. Each train consists of one 50% capacity pump, one component cooling water (CCW) heat exchanger, one essential chiller unit, piping, valves, instrumentation, and two types of strainers. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

Additional information about the design and operation of the ESWS, along with a list of the components served, is presented in Chapter 9 (Ref. 1). The principal safety related function of the ESWS is the removal of decay heat from the reactor via the CCW System.

APPLICABLE SAFETY ANALYSES

The design basis of the ESWS is for two ESWS trains, in conjunction with the CCW System to remove core decay heat following a design basis LOCA. This prevents the refueling water storage pit fluid from increasing in temperature following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System. The ESWS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The ESWS, in conjunction with the CCW System, also cools the unit from containment spray/residual heat removal (CS/RHR), as discussed in Chapter 5, (Ref. 2) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and CS/RHR System trains that are operating.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two ESWS trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESWS temperature of 95°F occurring simultaneously with maximum heat loads on the system.

The ESWS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Three of the four ESWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ESWS train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. The pump is OPERABLE and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESWS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ESWS are determined by the systems it supports.

ACTIONS

A.1 [and A.2]

If one of the required ESWS trains is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ESWS trains are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ESWS trains could result in loss of ESWS function. Required Action A.1 is modified by two Notes.

BASES

ACTIONS (continued)

The Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ESWS train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. [Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Required Action is not applicable in MODE 4.] The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the ESWS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ESWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the ESWS.

Verifying the correct alignment for manual, power operated, and automatic valves in the ESWS flow path provides assurance that the proper flow paths exist for ESWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

[The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the ESWS valves on an actual or simulated actuation signal. The ESWS is a normally operating system that cannot be fully actuated as part of normal testing. This surveillance is tested to assure the requirements of IST program described in Table 3.9-14. The motor operated valve is provided at the discharge of each pump. The starting logic of the ESWP interlocks the motor operated valve with the pump operation. This interlock prevents the pump from starting if the valve is not closed. The closed discharge valve opens after starting the ESWP. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. [The 24 month Frequency is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length. This equipment is not at risk of imminent damage as it is designed to remain functional and in good condition while in operation, thus significant degradation due to a longer surveillance interval should not be of major concern. The design reliability is, therefore, maintained by taking these considerations based on sound engineering judgment. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

SR 3.7.8.3

This SR verifies proper automatic operation of the ESWS pumps on an actual or simulated actuation signal. The ESWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. [The 24month Frequency is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length. This equipment is not at risk of imminent damage as it is designed to remain functional and in good condition while in operation, thus significant degradation due to a longer surveillance interval should not be of major concern. The design reliability is, therefore, maintained by taking these considerations based on sound engineering judgment. OR The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]

BASES

REFERENCES

1. Subsection 9.2.1.

2. Subsection 5.4.7.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Essential Service Water System (ESWS) and the Component Cooling Water (CCWS) system.

[[The UHS consists of four 50 percent capacity mechanical draft cooling towers, one for each ESWS train. Each cooling tower consists of two cells with one fan per cell. The combined inventory of three of the four UHS basins provides a 30-day storage capacity as discussed in FSAR Chapter 9 (Ref. 1). Each unit is provided with its own independent UHS with no cross connection between the two units.]] The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that an adequate inventory of cooling water be available for [[30]] days without makeup, and that the design basis temperatures of safety-related equipment not be exceeded. [[Each UHS basin provides 33-1/3 percent of the combined inventory for the 30-day storage capacity to satisfy the short-term recommendation of Regulatory Guide 1.27 (Ref. 2). There is one safety-related UHS transfer pump per UHS basin which is used to transfer water between the UHS basins.]]

The [[stored]] water level provides adequate net positive suction head (NPSH) to the ESW pump during a 30-day period of operation following the design basis LOCA or safe shutdown with LOOP scenario without makeup.

Additional information on the design and operation of the system, along with a list of components served, can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation.

[[The operating limits are based on safe shutdown with LOOP. A conservative heat transfer analysis for the worst case LOCA was performed to ensure that the cooling tower capacity and the basin water

BASES

APPLICABLE SAFETY ANALYSES (continued)

inventory adequately remove the heat load for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure. The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2) which requires a 30 day supply of cooling water in the UHS.]]

The UHS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

[[The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the ESWS to operate for at least 30 days following a design basis LOCA or safe shutdown with LOOP, without makeup water, and provide adequate net positive suction head (NPSH) to the ESWS pumps, and without exceeding the maximum design temperature of the equipment served by the ESWS. To meet this condition, three UHS cooling towers with the UHS temperature not exceeding 93°F during MODES 1, 2, 3 and 4 and the level in each of three basins being maintained above 2,800,000 gallons are required—a volume correspondent to the safe shutdown with LOOP conditions that bounds the LOCA condition. Additionally, three of the UHS transfer pumps shall be OPERABLE, with each pump capable of transferring flow from a UHS basin meeting water inventory and temperature limits, and powered from an independent Class 1E electrical division.]]

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

[[A.1 and A.2

If one of the required cooling towers and associated fans is inoperable (i.e., one or more fans per cooling tower inoperable), action must be taken to restore the inoperable cooling tower and associated fan(s) to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE cooling towers with associated fans are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE UHS cooling towers could result in a loss of UHS function. Required Action A.2 allows the option to apply the requirements of Specification 5.5.18 to determine a Risk Informed Completion Time (RICT). This Required Action is not applicable in MODE 4. The 72-hour

BASES

ACTIONS (continued)

Completion Time is based on the capability of the OPERABLE cooling towers to provide the UHS cooling capability and the low probability of an accident occurring during the 72 hours that one required cooling tower and associated fans are inoperable.]]

B.1

With water temperature of the UHS [[> 93°F]], the design basis assumption associated with initial UHS temperature is bounded provided the temperature of the UHS averaged over the previous 24-hour period is [[≤ 93°F]]. With the water temperature of the UHS [[> 93°F]], long-term cooling capability of the ECCS loads may be affected. Therefore, to ensure longterm cooling capability is provided to the ECCS loads when water temperature of the UHS is [[> 93°F]], Required Action B.1 is provided to monitor the water temperature of the UHS more frequently and verify the temperature is [[≤ 93°F]] when averaged over the previous 24 hour period. The once per hour Completion Time takes into consideration UHS temperature variations and the increased monitoring frequency needed to ensure design basis assumptions and equipment limitations are not exceeded in this condition. If the water temperature of the UHS exceeds [[93°F]] when averaged over the previous 24 hour period, Condition E must be entered immediately.

[[C.1

If one or more required UHS basins have a water level not within the limits, action must be taken to restore the water level to within limits within 72 hours.

The 72 hour Completion Time is reasonable based on the low probability of an accident occurring during the 72 hours, the considerable cooling capacity still available in the basin(s), and the time required to reasonably complete the Required Action. Furthermore, there would be no significant loss in the UHS cooling capacity when the water level drops below the normal level during a 72-hour period because of sufficient cooling tower basin inventory. The UHS has a combined design heat removal capacity of approximately 20 days from two operable cooling tower basins and 30 days from three operable cooling tower basins.]]

BASES

ACTIONS (continued)

[[D.1, D.2.1, and D.2.2

If one or more required UHS transfer pump(s) are inoperable, action must be taken to restore the pump(s) to OPERABLE status or implement an alternate method of transferring the affected basin within 7 days. If an alternate method is utilized, action still must be taken to restore the transfer pump(s) to OPERABLE status within 31 days.

The Completion Times are reasonable based on the low probability of an accident occurring during the time allowed to restore the pump(s) or implement an alternate method, the availability of alternate methods, and the amount of time available to transfer the water from one basin to the other under the worst case accident assumptions. Furthermore, the inoperability of all required transfer pumps leaves only two cooling tower basins with a combined design heat removal capacity of approximately 20 days. This cooling period bounds and justifies the 7-day completion time to restore the transfer pumps to operable status.]]

E.1 and E.2

If the Required Actions and Completion Times of Condition [[A, B, C, or D]] are not met, or the UHS is inoperable for reasons other than Condition [[A, B, C, or D]] the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ESWS pumps. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. This SR verifies that [[each]] required UHS [[basin]] water inventory is [[≥ 2,800,000 gallons]]. Plant procedures provide the corresponding water level to be verified to assure a usable volume of [[2,800,000 gallons]], accounting for unusable volume and measurement uncertainty.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies that the ESWS is available to cool the CCW System and essential chiller unit to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a design basis LOCA or safe shutdown with LOOP. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. This SR verifies that the water temperature of the UHS is [[≤ 93°F]].

[[SR 3.7.9.3

Operating each cooling tower fan for ≥15 minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]]

[[SR 3.7.9.4

This SR verifies that each UHS fan starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]]

[[SR 3.7.9.5

This SR verifies that each UHS transfer pump starts and operates on a manual actuation signal. Verification of the UHS transfer pump operation includes testing to verify the pump's developed head at the flow test point is greater than or equal to the required developed head. Testing also includes verification of required valve position.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.]]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.6

This SR verifies the correct alignment for manual, power-operated, and automatic valves in the UHS flow path to assure that the proper flow paths exist for UHS operation. This SR does not apply to valves that are locked, sealed or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk, and is controlled under the Surveillance Frequency Control Program.

SR 3.7.9.7

This SR verifies proper manual and automatic operation of the UHS valves on remote manual or on an actual or simulated actuation signal. The UHS is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk, and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR Subsection 9.2.5.

2. Regulatory Guide 1.27.

3.0 INTERFACE REQUIREMENTS

3.1 DESIGN DESCRIPTION

This section identifies the safety significant interface requirements between the US-APWR standard plant design and the site-specific design.

The US-APWR standard plant design consists of several buildings (reactor building including the prestressed concrete containment vessel and containment internal structure, power source buildings, auxiliary building, turbine building and access building); and the systems and equipment located in those buildings. For some systems included in the standard plant design, the associated structure (e.g., the power source fuel storage vaults and essential service water pipe tunnel) is a site-specific structure. As allowed by the regulations, conceptual designs for systems that are not part of the US-APWR standard plant design are included in the DCD for the purpose of allowing the NRC to evaluate the overall acceptability of the standard plant design. However, the final details of these conceptual designs are subject to change due to site-specific conditions.

Although descriptions of the power source fuel storage vaults (PSFSVs) and the essential service water pipe tunnel (ESWPT) are provided in this DCD, the structural design of the PSFSVs and ESWPT, including seismic and dynamic qualification, as applicable, are finalized based on the site-specific arrangement.

An interface requirement as specified in this section applies to a system, a portion of a system, or a structure that must be added or connected to the standard plant design to complete the design of the US-APWR at a specific site.

A COL applicant referencing the US-APWR certified design is responsible for site-specific designs that meet the interface requirements and for verifying that the as-built structures, systems, and components conform to the site-specific designs using an ITAAC process that is similar to that provided for the certified design.

3.2 INTERFACE REQUIREMENTS

3.2.1 Ultimate Heat Sink

Ultimate heat sink (UHS) is a safety-related system and is site-specific. The following are site-specific interface requirements:

a. The UHS system design meets the divisional separation requirements of the essential service water system (ESWS) and the UHS is capable of performing its safety functions under design basis event conditions and coincident single failure with or without offsite power available.

b. The safety related, pressure retaining components, and their supports, are designed, constructed and inspected in accordance with ASME Code Section III, if applicable to the site-specific design.

c. The maximum supply water temperature is 95 °F under the peak heat loads condition to provide sufficient cooling capacity to ESWS.

d. The UHS water level is maintained such that available net positive suction head (NPSH) is greater than the ESW pump's required NPSH during all plant operating conditions including normal plant operations, abnormal and accident conditions. The ESW pump operation does not cause vortex formation at minimum allowed UHS water level.

e. The UHS system has main control room (MCR) and remote shutdown console (RSC) alarms and displays for UHS water level and water temperature.

f. The UHS system has MCR and RSC controls for UHS components' active safety functions if applicable to the site-specific design.

g. UHS components that have protection and safety monitoring system (PSMS) control (if applicable to the site-specific design) perform an active safety function after receiving a signal from PSMS.

h. The UHS can provide the required cooling for a minimum of 30 days without make-up during accident conditions.

i. The UHS system is designed to prevent water hammer.

3.2.2 Fire Protection System

Portions of the fire protection system are site specific. The following are the site-specific interface requirements:

a. The seismic standpipe system can be supplied from a seismic Category I water source with a capacity of at least 18,000 gallons.

b. The fire protection system water supply is from two separate, reliable freshwater sources; or from one freshwater lake or pond of sufficient size with two separate and independent suctions in one or more intake structure(s).

3.2.3 Essential Service Water System

Portions of the ESWS are site specific due to its dependence on the site-specific UHS system. The following are the site-specific interface requirements:

a. The ESWS piping in the ESWPT that connects to the UHS system is designed, constructed and inspected in accordance with ASME Code Section III.

b. System layout of the ESWS and UHS system is verified to assure that the pressures in the ESWS and UHS system are above saturation conditions during all plant operating conditions including normal plant operations, abnormal and accident conditions.

c. The sum of the ESW pump shutoff head and static head is such that the ESW system design pressure is not exceeded.

d. The ESWS is designed to prevent water hammer.

e. The ESWS can provide cooling water required for the component cooling water (CCW) heat exchangers and the essential chiller units of the essential chilled water system

(ECWS) during all plant operating conditions, including normal plant operations, abnormal and accident conditions.

3.2.4 Electrical System

The offsite power system and components are site-specific. The following features are site-specific interface requirements:

a. The electrical system has a minimum of two independent offsite transmission circuits from the transmission network (TN) to the safety buses with no intervening non-safety buses (direct connection).

b. The offsite TN voltage variations during steady state operation do not cause voltage variations beyond an acceptable tolerance of the loads' nominal ratings.

c. The offsite TN normal steady state frequency is within an acceptable tolerance of 60Hz during recoverable periods of instability.

d. The offsite transmission circuits have the capacity and capability to power the required loads during steady state, transient, and postulated events and accident conditions.

e. There is physical separation and electrical independence between the offsite circuits and onsite class 1E electrical system and components.

f. Lightning protection and grounding features exist for the systems and components of the offsite circuits from the TN to the safety buses.

g. The electrical system has alarms and displays for monitoring the switchyard status.

h. The electrical system has the capability to automatically fast transfer from the preferred power supply to the non-preferred power supply.

i. The switchyard agreement and protocols between the nuclear power plant and the TN owner/operator assess the risk and probability of a loss of offsite power due to performing maintenance activities on the electrical system.

j. The electrical system is designed to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear unit, the loss of power from the TN, or the loss of power from the onsite electric power supplies.