COMBINED LICENSE

VIRGIL C. SUMMER NUCLEAR STATION UNIT 2

SOUTH CAROLINA ELECTRIC AND GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

Docket No. 52-027

License No. NPF-93

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for a combined license (COL) for Virgil C. Summer Nuclear Station (VCSNS) Unit 2 filed by South Carolina Electric & Gas Company (SCE&G), acting on behalf of itself and South Carolina Public Service Authority (Santee Cooper), herein referred to as "the VCSNS owners," which incorporates by reference Appendix D to 10 CFR Part 52, complies with the applicable standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. There is reasonable assurance that the facility will be constructed and will operate in conformity with the application, as amended, the provisions of the Act, and the Commission regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.F below;

   C. There is reasonable assurance (i) that the activities authorized by this COL can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.F below;

   D. SCE&G[1] is technically qualified to engage in the activities authorized by this license in accordance with the Commission regulations set forth in 10 CFR Chapter I. The VCSNS owners are financially qualified to engage in the activities authorized by this COL in accordance with the Commission regulations set forth in 10 CFR Chapter I;

[1] SCE&G is authorized by Santee Cooper to exercise responsibility and control over the physical construction, operation, and maintenance of the facility.

E. The VCSNS owners have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements;"

F. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

G. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of this license subject to the conditions for protection of the environment set forth herein is in accordance with Subpart A of 10 CFR Part 51 and all applicable requirements have been satisfied; and

H. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the applicable regulations in 10 CFR Parts 30, 40, and 70.

2. On the basis of the foregoing findings regarding this facility, COL No. NPF-93 is hereby issued to SCE&G and Santee Cooper (the licensees), to read as follows:

A. This license applies to the VCSNS Unit 2, a light-water nuclear reactor and associated equipment (the facility), owned by the VCSNS owners. The facility would be located approximately 1 mile from the center of VCSNS Unit 1 in western Fairfield County, approximately 15 miles west of Winnsboro, and 26 miles northwest of Columbia, SC and is described in the licensee's final safety analysis report (FSAR), as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) (a) SCE&G, pursuant to Sections 103 and 185b. of the Act and 10 CFR Part 52, to construct, possess, use, and operate the facility at the designated location in accordance with the procedures and limitations set forth in this license;

(b) Santee Cooper pursuant to the Act and 10 CFR Part 52, to possess but not operate the facility at the designated location in Fairfield County, South Carolina, in accordance with the procedures and limitations set forth in this license;

(2) (a) SCE&G, pursuant to the Act and 10 CFR Part 70, to receive and possess at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

(b) SCE&G, pursuant to the Act and 10 CFR Part 70, to use special nuclear material as reactor fuel, after a Commission finding under 10 CFR 52.103(g) has been made, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

(3) (a) SCE&G, pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, at any time before a Commission finding under 10 CFR 52.103(g), such byproduct and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts, as necessary;

(b) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as necessary;

(4) (a) SCE&G, pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, before a Commission finding under 10 CFR 52.103(g), in amounts not exceeding those specified in 10 CFR 30.72, any byproduct or special nuclear material that is (1) in unsealed form; (2) on foils or plated surfaces, or (3) sealed in glass, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components;

(b) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), in amounts as necessary, any byproduct, source, or special nuclear material without restriction as to chemical or physical form, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components but not uranium hexafluoride; and

(5) SCE&G, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license is subject to, and the licensees shall comply with, all applicable provisions of the Act and the rules, regulations, and orders of the Commission, including the conditions set forth in 10 CFR Chapter I, now or hereafter in effect.

D. The license is subject to, and SCE&G shall comply with, the conditions specified and incorporated below:

(1) <u>Changes during Construction</u>

(a) SCE&G may request use of a preliminary amendment request (PAR) process, for license amendments, at any time before a Commission finding under 10 CFR 52.103(g). To use the PAR process, SCE&G shall submit a written request to the Office of New Reactors (NRO) in accordance with COL-ISG-025, "Changes during Construction under Part 52."

(b) Before NRO's issuance of a written PAR notification, SCE&G shall submit the license amendment request (LAR). Thereafter, NRO will issue a written PAR notification, setting forth whether SCE&G may proceed in accordance with the PAR, LAR, and COL-ISG-025. If SCE&G elects to proceed and the LAR is subsequently denied, SCE&G shall return the facility to its current licensing basis.

(2) Pre-operational Testing

(a) SCE&G shall perform the design-specific pre-operational tests identified below:

1. In-Containment Refueling Water Storage Tank (IRWST) Heatup Test (first plant test as identified in AP1000 Design Control Document (DCD), Rev. 19, Section 14.2.9.1.3 Item (h));

2. Pressurizer Surge Line Stratification Evaluation (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.7 Item (d));

3. Reactor Vessel Internals Vibration Testing (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.9);

4. Core Makeup Tank Heated Recirculation Tests (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Items (k) and (w)); and

5. Automatic Depressurization System Blowdown Test (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Item (s)).

(b) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(2)(a) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.9.

(c) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the design-specific pre-operational tests identified in Section 2.D.(2)(a) of this license; and

(d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon the successful completion of all the ITAAC included in Appendix C to this license.

(3) Nuclear Fuel Loading and Pre-critical Testing

(a) Until the submission of the notification required by Section 2.D.(2)(c) of this license, SCE&G shall not load fuel into the reactor vessel;

(b) Upon submission of the notification required by Section 2.D.(2)(c) of this license and upon a Commission finding in accordance with 10 CFR 52.103(g) that all the acceptance criteria in the ITAAC in Appendix C to this license are met, SCE&G is authorized to perform pre-critical tests in accordance with the conditions specified herein;

(c) SCE&G shall perform the pre-critical tests identified in AP1000 DCD Rev. 19, Section 14.2.10.1;

(d) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(3)(c) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10; and

(e) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the pre-critical tests identified in Section 2.D.(3)(c) of this license.

(4) Initial Criticality and Low-Power Testing

(a) Upon submission of the notification required by Section 2.D.(3)(e) of this license, SCE&G is authorized to operate the facility at reactor steady-state core power levels not to exceed 5-percent thermal power in accordance with the conditions specified herein;

(b) SCE&G shall perform the initial criticality and low-power tests identified in AP1000 DCD Rev. 19, Sections 14.2.10.2 and 14.2.10.3, respectively, the Natural Circulation (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.6, and the Passive Residual Heat Removal Heat Exchanger (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.7;

(c) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(4)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10.2 and 14.2.10.3; and

(d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of initial criticality and low-power tests identified in Section 2.D.(4)(b) of this license, including the design-specific tests identified therein.

(5) Power Ascension Testing

(a) Upon submission of the notification required by Section 2.D.(4)(d) of this license, SCE&G is authorized to operate the facility at reactor steady-state core power levels not to exceed 100-percent thermal power in accordance with the conditions specified herein, but only for the purpose of performing power ascension testing;

(b) SCE&G shall perform the power ascension tests identified in the AP1000 DCD Rev. 19, Section 14.2.10.4, the Rod Cluster Control Assembly Out of Bank Measurements (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.6, and the Load Follow Demonstration (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.22;

(c) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(5)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev.19, Section 14.2.10.4; and

(d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of power ascension tests identified in Section 2.D.(5)(b) of this license, including the design-specific tests identified therein.

(6) Maximum Power Level

Upon submission of the notification required by Section 2.D.(5)(d) of this license, SCE&G is authorized to operate the facility at steady state reactor core power levels not to exceed 3400 MW thermal (100-percent thermal power), as described in the FSAR, in accordance with the conditions specified herein.

(7) Reporting Requirements

(a) Within 30 days of a change to the initial test program described in FSAR Section 14, Initial Test Program, made in accordance with 10 CFR 50.59 or in accordance with 10 CFR Part 52, Appendix D, Section VIII, "Processes for Changes and Departures," SCE&G shall report the change to the Director of NRO, or the Director's designee, in accordance with 10 CFR 50.59(d).

(b) SCE&G shall report any violation of a requirement in Section 2.D.(3), Section 2.D.(4), Section 2.D.(5), and Section 2.D.(6) of this license within 24 hours. Initial notification shall be made to the NRC Operations Center in accordance with 10 CFR 50.72, with written follow up in accordance with 10 CFR 50.73.

(8) Incorporation

The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. 75, are hereby incorporated into this license.

(9) Technical Specifications

The technical specifications in Appendix A to this license become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g).

(10) Operational Program Implementation

SCE&G shall implement the programs or portions of programs identified below, on or before the date SCE&G achieves the following milestones.

(a) Environmental Qualification Program implemented before initial fuel load;

(b) Reactor Vessel Material Surveillance Program implemented before initial criticality;

(c) Preservice Testing Program implemented before initial fuel load;

(d) Containment Leakage Rate Testing Program implemented before initial fuel load;

(e) Fire Protection Program

1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18);

2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite;

3. All fire protection program features implemented before initial fuel load;

(f) Standard Radiological Effluent Controls implemented before initial fuel load;

(g) Offsite Dose Calculation Manual implemented before initial fuel load;

(h) Radiological Environmental Monitoring Program implemented before initial fuel load;

(i) Process Control Program implemented before initial fuel load;

(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof:

1. RPP features applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials;

2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site;

3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive waste shipment implemented before initial fuel load;

4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before first shipment of radioactive waste;

(k) Reactor Operator Training Program implemented 18 months before the scheduled date of initial fuel load;

(l) Motor-Operated Valve Testing Program implemented before initial fuel load;

(m) Initial Test Program (ITP)

1. Component Test Program implemented before the first component test;

2. Preoperational Test Program implemented before the first preoperational test; and

3. Startup Test Program implemented before initial fuel load;

(n) Special Nuclear Material Control and Accounting Program implemented before initial receipt of special nuclear material; and

(o) Special Nuclear Material Physical Protection Program implemented before initial receipt of special nuclear material on site.

(11) Operational Program Implementation Schedule

No later than 12 months after issuance of the COL, SCE&G shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the operational programs listed in FSAR Table 13.4-201, including the associated estimated date for initial loading of fuel. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until all the operational programs listed in FSAR Table 13.4-201 have been fully implemented.

(12) Site-and Unit-specific Conditions

(a) Before commencing installation of individual piping segments and connected components in their final locations, SCE&G shall complete the as-designed pipe rupture hazards analysis for compartments (rooms) containing those segments in accordance with the criteria outlined in the AP1000 DCD, Rev. 19, Sections 3.6.1.3.2 (as revised by Amendment No. 48) and 3.6.2.5, and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of this analysis and the availability of the as-designed pipe rupture hazards analysis reports.

(b) Before commencing installation of individual piping segments identified in AP1000 DCD, Rev. 19, Section 3.9.8.7, and connected components in their final locations in the facility, SCE&G shall complete the analysis of the as-designed individual piping segments and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of these analyses and the availability of the design reports for the selected piping packages.

(c) No later than 180 days before initial fuel load, SCE&G shall submit to the Director of NRO, or the Director's designee, in writing:

1. A fully developed set of plant-specific emergency action levels (EALs) for VCSNS Unit 2 in accordance with the criteria defined in Amendment 68. The EALs shall have been discussed and agreed upon with State and local officials.

2. An assessment of emergency response staffing performed in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

(d) SCE&G shall not revise or modify the provisions of Sections 5.3, 5.4, 5.6, 5.9, and 5.10 of the Special Nuclear Material (SNM) Physical Protection Program until the requirements of 10 CFR 73.55 are implemented.

(e) No later than 12 months after issuance of the COL, SCE&G shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the following license conditions. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until each license condition has been fully implemented. The schedule shall identify the completion of or implementation of the following:

1. The construction and inspection procedures for steel concrete composite (SC) construction activities for seismic Category I nuclear island modules (including shield building SC modules) described in AP1000 DCD Rev. 19, Section 3.8.4.8;

2. The spent fuel rack Metamic Coupon monitoring program (before initial fuel load);

3. Implementation of the flow accelerated corrosion (FAC) program including construction phase activities (before initial fuel load);

4. A turbine maintenance and inspection program, which must be consistent with the maintenance and inspection program plan activities and inspection intervals identified in FSAR Section 10.2.3.6 (before initial fuel load);

5. The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (before initial fuel load);

6. The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (before initial fuel load);

7. The site-specific severe accident management guidelines (before startup testing);

8. The operational and programmatic elements of the mitigative strategies for responding to circumstances associated with loss of large areas of the plant due to explosions or fire developed in accordance with 10 CFR 50.54(hh)(2) (before initial fuel load); and

9. The ITP procedures identified in FSAR Section 14.2.3

    a. administrative manual (before the first component test)

    b. preoperational testing (before scheduled performance)

    c. startup testing (before initial fuel load)

(f) Before initial fuel load, SCE&G shall:

1. Update the seismic interaction analysis in AP1000 DCD, Rev. 19, Section 3.7.5.3 to reflect as-built information, which must be based on as-procured data, as well as the as-constructed condition;

2. Reconcile the seismic analyses described in Section 3.7.2 of the AP1000 DCD, Rev. 19, to account for detailed design changes, including, but not limited to, those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information;

3. Calculate the instrumentation uncertainties of the actual plant operating instrumentation to confirm that either the design limit departure from nucleate boiling ratio (DNBR) values remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties;

4. Update the pressure-temperature (P-T) limits using the pressure temperature limits report (PTLR) methodologies approved in AP1000 DCD, Rev. 19, using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications of and use the Westinghouse generic PTLR curves;

5. Verify that plant-specific belt line material properties are consistent with the properties given in AP1000 DCD Rev. 19, Section 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification must include a pressurized thermal shock (PTS) evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective. Submit this PTS evaluation report to the Director of NRO, or the Director's designee, in writing, at least 18 months before initial fuel load;

6. Review differences between the as-built plant and the design used as the basis for the AP1000 seismic margin analysis. SCE&G shall perform a verification walkdown to identify differences between the as-built plant and the design. SCE&G shall evaluate any differences and must modify the seismic margin analysis as necessary to account for the plant-specific design and any design

changes or departures from the certified design. SCE&G shall compare the as-built structures, systems, and components (SSC) high confidence, low probability of failures (HCLPFs) with those assumed in the AP1000 seismic margin evaluation, before initial fuel load. SCE&G shall evaluate deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis to determine if vulnerabilities have been introduced;

7. Review differences between the as-built plant and the design used as the basis for the AP1000 probabilistic risk assessment (PRA) and the AP1000 DCD, Rev. 19, Table 19.59-18. SCE&G shall evaluate the plant-specific PRA-based insight differences and shall modify the plant-specific PRA model as necessary to account for the plant-specific design and any design changes or departure from the PRA certified in Rev. 19 of the AP1000 DCD;

8. Review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. SCE&G shall evaluate the plant-specific internal fire and internal flood analyses and shall modify the analyses as necessary to account for the plant-specific design and any design changes or departures from the design certified in Rev. 19 of the AP1000 DCD; and

9. Perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069, “Equipment Survivability Assessment,” to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. SCE&G shall perform this assessment for equipment used for severe accident mitigation that has not been tested at severe accident conditions. SCE&G shall assess the ability of the as-built equipment to perform during accident hydrogen burns using the environment enveloping method or the test based thermal analysis method described in Electric Power Research Institute (EPRI) NP-4354, “Large Scale Hydrogen Burn Equipment Experiments.”

10. Implement a surveillance program for explosively actuated valves (squib valves) that includes the following provisions in addition to the requirements specified in the edition of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) as incorporated by reference in 10 CFR 50.55a.

a. Preservice Testing

All explosively actuated valves shall be preservice tested by verifying the operational readiness of the actuation logic and associated electrical circuits for each explosively actuated valve with its pyrotechnic charge removed from the valve. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available at the explosively actuated valve from each circuit that is relied upon to actuate the valve. In addition, a sample of at least 20% of the pyrotechnic charges in all explosively actuated valves shall be tested in the valve or a qualified test fixture to confirm the capability of each sampled pyrotechnic charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. The sampling must select at least one explosively actuated valve from each redundant safety train. Corrective action shall be taken to resolve any deficiencies identified in the operational readiness of the actuation logic or associated electrical circuits, or the capability of a pyrotechnic charge. If a charge fails to fire or its capability is not confirmed, all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch number that has demonstrated successful 20% sampling of the charges.

b. Operational Surveillance

Explosively actuated valves shall be subject to the following surveillance activities after commencing plant operation:

i. At least once every 2 years, each explosively actuated valve shall undergo visual external examination and remote internal examination (including evaluation and removal of fluids or contaminants that may interfere with operation of the valve) to verify the operational readiness of the valve and its actuator. This examination shall also verify the appropriate position of the internal actuating mechanism and proper operation of remote position indicators. Corrective action shall be taken to resolve any deficiencies identified during the

examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

ii. At least once every 10 years, each explosively actuated valve shall be disassembled for internal examination of the valve and actuator to verify the operational readiness of the valve assembly and the integrity of individual components and to remove any foreign material, fluid, or corrosion. The examination schedule shall provide for both of the two valve designs used for explosively actuated valves at the facility to be included among the explosively actuated valves to be disassembled and examined every 2 years. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

iii. For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the operational readiness of the actuation logic and associated electrical circuits shall be verified for each sampled explosively actuated valve following removal of its charge. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available for each valve actuation circuit. Corrective action shall be taken to resolve any deficiencies identified in the actuation logic or associated electrical circuits.

iv. For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the sampling must select at least one explosively actuated valve from each redundant safety train. Each sampled pyrotechnic charge shall be tested in the valve or a qualified test fixture to confirm the capability of the charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. Corrective action shall be taken to resolve any

deficiencies identified in the capability of a pyrotechnic charge in accordance with the preservice testing requirements.

This license condition shall expire upon (1) incorporation of the above surveillance provisions for explosively actuated valves into the facility's inservice testing program, or (2) incorporation of inservice testing requirements for explosively actuated valves in new reactors (i.e., plants receiving a construction permit, or combined license for construction and operation, after January 1, 2000) to be specified in a future edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a, including any conditions imposed by the NRC, into the facility's inservice testing program.

(13) <u>Mitigation Strategies for Beyond-Design Basis External Events</u>

SCE&G shall address the following requirements:

(a) SCE&G shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment and spent fuel pool cooling capabilities following a beyond-design-basis external event.

(b) These strategies must be capable of mitigating a simultaneous loss of all AC power and loss of normal access to the normal heat sink and have adequate capacity to address challenges to core cooling, containment, and spent fuel pool cooling capabilities at all units on the VCSNS site.

(c) SCE&G must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and spent fuel pool cooling capabilities at all units on the VCSNS site.

(d) SCE&G must be capable of implementing the strategies in all modes.

(e) Full compliance shall include procedures, guidance, training, and acquisition, staging, or installing of equipment needed for the strategies.

(f) SCE&G shall promptly start implementation of the requirements stated in this condition and shall complete full implementation prior to initial fuel load.

1. SCE&G shall, within twenty (20) days of issuance of this license, notify the Commission (1) if they are unable to comply with any of these requirements, (2) if compliance with any of the requirements is unnecessary in their

specific circumstances, or (3) if implementation of any of the requirements would cause SCE&G to be in violation of provisions of any Commission regulation or license. The notification shall provide SCE&G's justification for seeking relief from or variation of any specific requirement.

2. If SCE&G considers that implementation of any of these requirements would adversely impact safe and secure operation of the facility, SCE&G must notify the Commission, within twenty (20) days of issuance of the license, of the adverse safety impact, the basis for their determination that the requirement has an adverse safety impact, and either a proposal for achieving the same objectives specified in this license condition, or a schedule for modifying the facility to address the adverse safety condition. If neither approach is appropriate, then SCE&G must supplement their response to paragraph 2.D.(13)(f)1. of this license to identify the condition as a requirement with which they cannot comply, with attendant justifications as required in paragraph 2.D.(13)(f)1. of this license.

3. SCE&G shall, within one (1) year after issuance of the NRC's final Interim Staff Guidance detailing an acceptable approach for complying with these requirements, submit to the Commission for review an overall integrated plan, including a description of how compliance with the requirements described in section 2.D.(13) of this license will be achieved.

4. SCE&G shall provide an initial status report sixty (60) days following issuance of the final Interim Staff Guidance and at six (6)-month intervals following submittal of the overall integrated plan, as required in paragraph 2.D.(13)(f)3. of this license, which delineates progress made in implementing the requirements of this license condition

5. SCE&G shall report to the Commission when full compliance with the requirements described in section 2.D.(13) of this license is achieved.

6. SCE&G responses to conditions 2.D.(13)(f)1., 2.D.(13)(f)2., 2.D.(13)(f)3., 2.D.(13)(f)4., and 2.D.(13)(f)5. of this license, shall be submitted in accordance with 10 CFR 52.3.

E. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

F. Exemptions

(1) The following exemptions from any part of the referenced design certification rule meet the requirements of 10 CFR 52.7 and Section VIII.A.4, VIII.B.4, or VIII.C.4 of Appendix D to 10 CFR Part 52, are authorized by law, will not present an undue risk to the public health or safety, and are consistent with the common defense and security. Special circumstances are present in that the application of the regulation in these particular circumstance are not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the application and the staff SER dated August 17, 2011. In addition, for exemption 2.F.(1)(b) the exemption will not result in a significant decrease in the level of safety otherwise provided by the design, and the special circumstances outweigh any decrease in the safety that may result from the reduction in standardization caused by the exemption.

(a) The licensees are exempt from the requirement of 10 CFR Part 52, Appendix D, Section IV.A.2.a to include a plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 certified design. This exemption is specific to the organization and numbering scheme in the FSAR and is related to departure number STD DEP 1.1-1 and VCS DEP 2.0-1.

(b) The licensees are exempt from the requirement of 10 CFR Part 52, Appendix D, Section IV.A.2d to include information demonstrating compliance with the site parameters and interface requirements. This exemption is specific to the maximum safety wet bulb (noncoincident) air temperature in the FSAR and is related to departure number VCS DEP 2.0-2.

(2) The following exemptions from regulations were granted in the rulemaking for the design certification rule that is referenced in the application. In accordance with 10 CFR Part 52, Appendix D, Section V, Applicable Regulations, Subsection B, and pursuant to 10 CFR 52.63(a)(5), the licensees are exempt from portions of the following regulations:

(a) Paragraph (f)(2)(iv) of 10 CFR 50.34—Plant Safety Parameter Display Console;

(b) Paragraph (c)(1) of 10 CFR 50.62—Auxiliary (or emergency) feedwater system; and

(c) Appendix A to 10 CFR Part 50, GDC 17—Second offsite power supply circuit.

(3) For the reasons set forth below, the following specific exemptions which are outside the scope of the design certification rule referenced in the application are granted:

(a) The licensees are exempt from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c), 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51 because the licensees meet the requirements of 10 CFR 70.17 and 74.7 as follows. The exemption is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Additionally, special circumstances are present in that the application of the regulations in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the FSAR and the staff SER dated August 17, 2011.

(b) The licensees are exempt from the requirements of 10 CFR 52.93(a)(1) as it relates to the exemption granted in Section 2.F.(1)(a) of this license because the exemption meets the requirements of 10 CFR 52.7, because the exemption is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Additionally, special circumstances are present in that the application of the regulation in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the staff SER dated August 17, 2011.

G. Following SCE&G's ITAAC closure notifications under paragraph (c)(1) of 10 CFR 52.99 until the Commission makes the finding under 10 CFR 52.103(g), SCE&G shall notify the NRC, in a timely manner, of new information that materially alters the bases for determining that either inspections, tests, or analyses were performed as required, or that acceptance criteria are met. The notification must contain sufficient information to demonstrate that, notwithstanding the new information, the prescribed inspections, tests, or analyses have been performed as required, and the prescribed acceptance criteria are met.

H. SCE&G shall maintain the guidance and strategies developed in accordance with 10 CFR 50.54(hh)(2).

I. This license is effective as of March 30, 2012 and shall expire at midnight on the date 40 years from the date that the Commission finds that the acceptance criteria in the combined license are met in accordance with 10 CFR 52.103(g).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael R. Johnson, Director
Office of New Reactors

Appendices:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan
Appendix C – Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

APPENDIX A

VIRGIL C. SUMMER NUCLEAR STATION UNIT 2

TECHNICAL SPECIFICATIONS

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1.0 USE AND APPLICATION

1.1 Definitions

- NOTE -

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state required for OPERABILITY of a logic circuit and the verification of the required logic output. The ACTUATION LOGIC TEST shall be conducted such that it provides component overlap with the actuated device.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two-section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for OPERABILITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

1.1 Definitions

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.3. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same committed effective dose equivalent as the quantity and isotopic mixture of I-130, I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Table 2.1 of EPA Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” EPA-520/1-88-020, September 1988.
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same effective dose equivalent as the quantity and isotopic mixture of noble gases (Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138) actually present. The dose conversion factors used for this calculation shall be those listed in Table III.1 of EPA Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” EPA 402-R-93-081, September 1993.

1.1 Definitions

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions). The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from seals or valve packing, that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE); or

4. RCS LEAKAGE through the passive residual heat removal heat exchanger (PRHR HX) to the In-containment Refueling Water Storage Tank (IRWST).

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE and PRHR HX tube LEAKAGE) through a nonisolatable fault in a RCS component body, pipe wall, or vessel wall.

1.1 Definitions

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE-OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3400 MWt.

1.1 Definitions

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single assembly of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCAs not capable of being fully inserted, the reactivity worth of these assemblies must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level. c. In MODE 2 with k_{eff} < 1.0, and MODES 3, 4, and 5, the worth of fully inserted Gray Rod Cluster Assemblies (GRCAs) will be included in the SDM calculation.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

1.1 Definitions

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps.

Table 1.1-1 (page 1 of 1)
MODES

MODES	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[(a)]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	> 420
4	Safe Shutdown[(b)]	< 0.99	NA	$420 \geq T_{avg} > 200$
5	Cold Shutdown[(b)]	< 0.99	NA	≤ 200
6	Refueling[(c)]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in Technical Specifications are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meaning.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example, the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip ... OR A.2.1 Verify ... AND A.2.2.1 Reduce ... OR A.2.2.2 Perform ... OR A.3 Align ...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

However, when a <u>subsequent</u> train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within

1.3 Completion Times

DESCRIPTION (continued)

limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the <u>first</u> inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery ...."

1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 in 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One valve inoperable.	A.1 Restore valve to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a valve is declared inoperable, Condition A is entered. If the valve is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion time clocks for Required Actions B.1 and B.2 start. If the inoperable valve is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second valve is declared inoperable while the first valve is still inoperable, Condition A is not re-entered for the second valve. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable valve. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable valves is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable valves is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

1.3 Completion Times

EXAMPLES (continued)

On restoring one of the valves to OPERABLE status the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. This Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second valve being inoperable for > 7 days.

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

1.3 Completion Times

EXAMPLES (continued)

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A.

It is possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, there shall be administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls shall ensure that the Completion Times for those Conditions are not inappropriately extended.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours. If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

- NOTE -
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was only applicable to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve which caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve. Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Reduce THERMAL POWER to ≤ 50% RTP.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hours interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each “Once per 8 hours thereafter” interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour, or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

1.3 Completion Times

IMMEDIATE COMPLETION TIME	When “Immediately” is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The “specified Frequency” is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The “specified Frequency” consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are “otherwise stated” conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillances, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only “required” when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of “met” or “performed” in these instances conveys specific meanings. A Surveillance is “met” only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not “met.” “Performance” refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be

1.4 Frequency

DESCRIPTION (continued)

performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discusses these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated surveillance must be performed at least one time. Performance of the surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside the specified limits, or the Unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the surveillance must be performed within 12 hours.

The use of "Once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-- **- NOTE -** Not required to be performed until 12 hours after ≥ 25% RTP. -- Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-- **- NOTE -** Only required to be met in MODE 1. -- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an “otherwise stated” exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-- **- NOTE -** Only required to be performed in MODE 1. -- Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required <u>performance</u> of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
--- **- NOTE -** Not required to be met in MODE 3. --- Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an “otherwise stated” exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop cold leg temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5 the RCS pressure shall be maintained ≤ 2733.5 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and 3.0.6.

If the LCO is met, or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours; and

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

3.0 LCO Applicability

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the test required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.7, “Safety Function Determination Program (SFDP).” If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system’s Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability of individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the Surveillance, shall be a failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once”, the above interval extension does not apply.

If a Completion Time requires periodic performance on a “once per...” basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, which ever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period, and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SR Applicability

SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of a LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$.
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM to be within limits.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ±1% Δk/k of the predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	7 days
	AND	
	A.2 Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	-- **- NOTE -** The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. -- Verify measured core reactivity is within ±1% Δk/k of predicted values.	 Once prior to entering MODE 1 after each refueling <u>AND</u> ------------------------- **- NOTE -** Only required to be performed after 60 EFPD ------------------------- 31 EFPD thereafter

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 for the upper MTC limit.
MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit.
MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
C. MTC not within lower limit.	C.1 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC within upper limit.	Once prior to entering MODE 1 after each refueling

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	-- **- NOTE -** Not required to be performed if the MTC measured at the equivalent of equilibrium RTP all rods out (ARO) boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. --	
	Verify MTC is within lower limit.	Once within 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm <u>AND</u> 14 EFPD thereafter when MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

- NOTE -

Not applicable to Gray Rod Cluster Assemblies (GRCAs) during GRCA bank sequence exchange with the On-Line Power Distribution Monitoring System (OPDMS) monitoring parameters.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM to be within the limits specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One rod not within alignment limits.	B.1	Restore rod to within alignment limits.	1 hour with the OPDMS not monitoring parameters <u>AND</u> 8 hours
	<u>OR</u>		
	B.2.1.1	Verify SDM to be within the limits specified in the COLR.	1 hour
		<u>OR</u>	
	B.2.1.2	Initiate boration to restore SDM within limit.	1 hour
	<u>AND</u>		
	B.2.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	<u>AND</u>		
	B.2.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>		
	B.2.4	------------------------------------ **- NOTE -** Only required to be performed when OPDMS is not monitoring parameters. ------------------------------------ Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	<u>AND</u>		

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2.5	------------------------------------ **- NOTE -** Only required to be performed when OPDMS is not monitoring parameters. ------------------------------------	
		Perform SR 3.2.2.1.	72 hours
	<u>AND</u>		
	B.2.6	Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 3.	6 hours
D. More than one rod not within alignment limit.	D.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>		
	D.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>		
	D.2	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	-- **- NOTE -** Not applicable to GRCAs. -- Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	-- **- NOTE -** Not applicable to GRCAs. -- Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 500°F, and b. All reactor coolant pumps operating.	Once prior to reactor criticality after each removal of the reactor head, and after each earthquake requiring plant shutdown

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each Shutdown Bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

- NOTE -

This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is within the limits specified in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Restore shutdown banks to within limits.	1 hour 1 hour 2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the insertion limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1.
MODE 2 with $k_{eff} \geq 1.0$.

- NOTES -

1. This LCO is not applicable while performing SR 3.1.4.2.

2. This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks during GRCA bank sequence exchange with On-Line Power Distribution Monitoring System monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control Bank insertion limits not met.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limits.	1 hour
	<u>AND</u>	
	A.2 Restore control bank(s) to within insertion limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM is within the limits specified in the COLR. <u>OR</u> B.1.2 Initiate boration to restore SDM to within limits. <u>AND</u> B.2 Restore control bank sequence and overlap to within limits.	1 hour 1 hour 2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify the estimated critical control bank position is within limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours
SR 3.1.6.3	Verify sequence and overlap limits, specified in the COLR, are met for control banks not fully withdrawn from the core.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Bank Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using the On-Line Power Distribution Monitoring System (OPDMS).	Once per 8 hours
	<u>OR</u>	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI per group inoperable.	B.1	Place the control rods under manual control.	Immediately
	<u>AND</u>		
	B.2	Monitor and record Reactor Coolant System (RCS) T_{avg}.	Once per 1 hour
	<u>AND</u>		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using the incore detectors.	Once per 8 hours
	<u>AND</u>		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable position indicators by using the OPDMS.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1 Verify by administrative means all DRPIs for the affected banks are OPERABLE.	Once per 8 hours
	<u>AND</u>	
	D.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours
	<u>OR</u>	
	D.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Once prior to criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of:

LCO 3.1.3 “Moderator Temperature Coefficient (MTC),”
LCO 3.1.4 “Rod Group Alignment Limits,”
LCO 3.1.5 “Shutdown Bank Insertion Limits,”
LCO 3.1.6 “Control Bank Insertion Limits,” and
LCO 3.4.2 “RCS Minimum Temperature for Criticality”

may be suspended, and the number of required channels for LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation,” Functions 1, 2, and 3 may be reduced to 3 provided:

a. Reactor Coolant System (RCS) lowest loop average temperature is ≥ 541°F,

b. SDM is within the limits specified in the COLR, and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limits.	A.1 Initiate boration to restore SDM to within limits. <u>AND</u> A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a COT on power range neutron flux and intermediate range neutron flux channels per SR 3.3.1.6, SR 3.3.1.7, and SR 3.3.3.2.	Prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 541°F.	30 minutes
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.8.4	Verify SDM is within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves

LCO 3.1.9 Two CVS Demineralized Water Isolation Valves and two CVS Makeup Line Isolation Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

- NOTE -

Flow path(s) may be unisolated intermittently under administrative controls.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CVS demineralized water isolation valve inoperable. <u>OR</u> One CVS makeup line isolation valve inoperable. <u>OR</u> One CVS demineralized water isolation valve and one CVS makeup line isolation valve inoperable.	A.1 Restore two CVS demineralized water isolation valves and two CVS makeup line isolation valves to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two CVS demineralized water isolation valves inoperable. <u>OR</u> Two CVS makeup line isolation valves inoperable.	B.1 Isolate the affected flow path(s) from the demineralized water storage tank to the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify two CVS demineralized water isolation valves and two CVS makeup line isolation valves stroke closed.	In accordance with the Inservice Testing Program
SR 3.1.9.2	Verify closure time of each CVS makeup isolation valve is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.1.9.3	Verify each CVS demineralized water isolation valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Action A.4 shall be completed whenever this Condition is entered. -------------------------------- $F_Q^C(Z)$ not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^C(Z)$ exceeds limit.	15 minutes after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours after each $F_Q^C(Z)$ determination
	<u>AND</u>	
	A.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------------------------- **- NOTE -** Required Action B.4 shall be completed whenever this Condition is entered. -------------------------------- $F_Q^W(Z)$ not within limits.	B.1	Reduce THERMAL POWER ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	<u>AND</u>		
	B.2	Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	<u>AND</u>		
	B.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	72 hours
	<u>AND</u>		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.1
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	--- **- NOTE -** Not required to be performed if OPDMS was monitoring parameters upon exceeding 75% RTP. --- Verify $F_Q^C(Z)$ within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP
SR 3.2.1.2	--- **- NOTE -** Not required to be performed if OPDMS was monitoring parameters upon exceeding 75% RTP. --- Verify $F_Q^W(Z)$ within limits.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	--- **- NOTE -** Not required to be performed until 31 days after the last verification of OPDMS parameters. ---	
	Verify $F_Q^C(Z)$ within limit.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified <u>AND</u> 31 effective full power days (EFPD) thereafter

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.4	-- **- NOTES -** 1. Not required to be performed until 31 days after the last verification of OPDMS parameters. 2. If $F_Q^W(Z)$ measurements indicate maximum over z $F_Q^C(Z)$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.4 once per 7 EFPD until two successive flux maps indicate maximum over z $F_Q^C(Z)$ has not increased. --	
	Verify $F_Q^W(Z)$ within limits.	Once within 12 hours after achieving equilib-rium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE –** Required Actions A.2 and A.3 must be completed whenever Condition A is entered. -------------------------------- $F^N_{\Delta H}$ not within limits.	A.1.1 Restore $F^N_{\Delta H}$ to within limit. <u>OR</u> A.1.2.1 Reduce THERMAL POWER to < 50% RTP. <u>AND</u> A.1.2.2 Reduce Power Range Neutron Flux – High trip setpoints to ≤ 55% RTP. <u>AND</u> A.2 Perform SR 3.2.2.1. <u>AND</u>	4 hours 4 hours 72 hours 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.3	------------------------------------ **- NOTE -** THERMAL POWER does not have to be reduced to comply with this Required Action. ------------------------------------ Perform SR 3.2.2.1.	 Prior to THERMAL POWER exceeding 50% RTP <u>AND</u> Prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	-- **- NOTE -** Not required to be performed if OPDMS was monitoring parameters upon exceeding 75% RTP. -- Verify $F^N_{\Delta H}$ within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP
SR 3.2.2.2	-- **- NOTE -** Not required to be performed until 31 days after the last verification of OPDMS parameters. -- Verify $F^N_{\Delta H}$ within limits specified in the COLR.	31 effective full power days (EFPD)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band specified in the COLR about the target flux difference.

b. May deviate outside the target band with THERMAL POWER < 90% RTP, but ≥ 50% RTP, provided AFD is within the acceptable operation limits specified in the COLR and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours.

c. May deviate outside the target band with THERMAL POWER < 50% RTP

- NOTES -

1. The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
2. With THERMAL POWER ≥ 50% RTP, penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.
3. With THERMAL POWER < 50% RTP and > 15% RTP, penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.
4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.5, provided AFD is maintained within acceptable operation limits.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP. <u>AND</u> AFD not within the target band.	A.1 Restore AFD to within the target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP	15 minutes
C. ----------NOTE------------ Required Action C.1 must be completed whenever Condition C is entered. ------------------------------- THERMAL POWER < 90% and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. <u>OR</u> THERMAL POWER < 90% and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes
D. Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to ≤ 15% RTP.	9 hours

SURVEILLANCE REQUIREMENTS

- NOTE -
Not required to be performed until 7 days after the last verification of OPDMS parameters.

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	Update the target flux difference.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter
SR 3.2.3.3	- NOTE - The initial target flux difference after each refueling may be determined from design predictions. Determine, by measurement, the target flux difference.	Once within 31 EFPD after each refueling AND 92 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP and with the On-Line Power Distribution Monitoring System (OPDMS) not monitoring parameters.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	AND	
	A.2 Perform SR 3.2.4.1.	Once per 12 hours
	AND	
	A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1 AND Once per 7 days thereafter
	AND	
	A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND	

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.5	------------------------------------ **- NOTES -** 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. ------------------------------------	
		Normalize excore detectors to restore QPTR to within limit.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		
	A.6	------------------------------------ **- NOTE -** Perform Required Action A.6 only after Required Action A.5 is completed. ------------------------------------	
		Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

- NOTE -

Not required to be performed until 12 hours after the last verification of OPDMS parameters.

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	**- NOTES -** 1. With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. Verify QPTR within limit by calculation.	7 days
SR 3.2.4.2	**- NOTE -** Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. Verify QPTR is within limit using a minimum of 4 symmetric pairs of fixed incore detectors.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.5 On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters

LCO 3.2.5 The following parameters shall not exceed their operating limits as specified in the COLR:

a. Peak Linear Power Density;
b. $F^{N}_{\Delta H}$;
c. DNBR; and
d. SDM.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP with OPDMS monitoring parameters a, b, and c.
MODE 1 with OPDMS monitoring parameter d.
MODE 2 with k_{eff} ≥ 1.0 and OPDMS monitoring parameter d.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the parameters a. through c. above not within limits.	A.1 Restore all parameters to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours
C. Parameter d above not within limits.	C.1 Initiate boration to restore SDM to within limits.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	Verify the parameters a. through d. to be within their limits.	24 hours

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place inoperable channel in bypass or trip.	6 hours
B. One or more Functions with two channels inoperable.	B.1 Place one inoperable channel in bypass. <u>AND</u> B.2 Place one inoperable channel in trip.	6 hours 6 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> One or more Functions with three or more channels inoperable.	C.1 Enter the Condition Referenced in Table 3.3.1-1 for the channel(s).	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1 Reduce THERMAL POWER to < P-10.	6 hours

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	----------------------------------- **- NOTES -** 1. Adjust nuclear instrument channel in the Protection and Safety Monitoring System (PMS) if absolute difference is > 1% RTP. 2. Required to be met within 12 hours after reaching 15% RTP. 3. If the calorimetric heat balance is < 70% RTP, and if the nuclear instrumentation channel indicated power is: a. lower than the calorimetric measurement by > 1%, then adjust the nuclear instrumentation channel upward to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. -----------------------------------	
	Compare results of calorimetric heat balance to nuclear instrument channel output.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-- **- NOTES -** 1. Adjust the conversion factor, ΔT°, in the ΔT power calculation ($q_{\Delta T}$) if absolute difference between $q_{\Delta T}$ and the calorimetric measurement is > 1% RTP. 2. Required to be met within 12 hours after reaching 50% RTP. 3. If the calorimetric heat balance is < 70% RTP, and if $q_{\Delta T}$ is: a. lower than the calorimetric measurement by > 5%, then adjust ΔT° to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. -- Compare results of calorimetric heat balance to the ΔT power calculation ($q_{\Delta T}$) output.	24 hours
SR 3.3.1.4	-- **- NOTES -** 1. Adjust nuclear instrument channel in PMS if absolute difference is ≥ 3% AFD. 2. Required to be met within 24 hours after reaching 20% RTP. -- Compare results of the incore detector measurements to nuclear instrument channel AXIAL FLUX DIFFERENCE.	31 effective full power days (EFPD)
SR 3.3.1.5	-- **- NOTE -** Required to be met within 24 hours after reaching 50% RTP. -- Calibrate excore channels to agree with incore detector measurements.	92 EFPD

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	Perform COT in accordance with Setpoint Program.	92 days
SR 3.3.1.7	-- **- NOTE -** Only required to be performed when not performed within previous 92 days. -- Perform COT in accordance with Setpoint Program.	Prior to reactor startup <u>AND</u> 4 hours after reducing power below P-10 <u>AND</u> 92 days thereafter
SR 3.3.1.8	-- **- NOTE -** This Surveillance shall include verification that the time constants are adjusted to within limits. -- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.1.9	-- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.10	--- **- NOTE -** Verification of setpoint is not required. --- Perform TADOT.	24 months
SR 3.3.1.11	--- **- NOTE -** Neutron detectors are excluded from response time testing. --- Verify RTS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 1 of 2)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
1. Power Range Neutron Flux				
a. High Setpoint	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.6 SR 3.3.1.9 SR 3.3.1.11
b. Low Setpoint	1[(a)],2	4	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.11
2. Power Range Neutron Flux High Positive Rate	1,2	4	D	SR 3.3.1.6 SR 3.3.1.9 SR 3.3.1.11
3. Overtemperature ΔT	1,2	4 (2/loop)	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
4. Overpower ΔT	1,2	4 (2/loop)	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
5. Pressurizer Pressure				
a. Low Setpoint	1[(b)]	4	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
b. High Setpoint	1,2	4	D	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
6. Pressurizer Water Level – High 3	1[(b)]	4	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11

(a) Below the P-10 (Power Range Neutron Flux) interlocks.

(b) Above the P-10 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 2)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
7. Reactor Coolant Flow – Low	1[(b)]	4 per hot leg	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
8. Reactor Coolant Pump (RCP) Bearing Water Temperature – High	1,2	4 per RCP	D	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
9. RCP Speed – Low	1[(b)]	4 (1/pump)	E	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
10. Steam Generator (SG) Narrow Range Water Level – Low	1,2	4 per SG	D	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
11. Steam Generator (SG) Narrow Range Water Level – High 2	1,2[(c)]	4 per SG	D	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.8 SR 3.3.1.11
12. Passive Residual Heat Removal Actuation	1,2	4 per valve	D	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11

(b) Above the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-11 (Pressurizer Pressure) interlock.

3.3 INSTRUMENTATION

3.3.2 Reactor Trip System (RTS) Source Range Instrumentation

LCO 3.3.2 Four channels of RTS Source Range Neutron Flux - High Setpoint instrumentation shall be OPERABLE.

APPLICABILITY: MODE 2 with intermediate range neutron flux below the P-6 interlock,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable in MODE 2.	A.1 Place inoperable channel in bypass or trip.	2 hours
B. Two channels inoperable in MODE 2.	B.1 Place one inoperable channel in bypass. AND B.2 Place one inoperable channel in trip.	2 hours 2 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Suspend operations involving positive reactivity additions.	Immediately
D. One or two channels inoperable in MODE 3, 4, or 5.	D.1 Restore three of four channels to OPERABLE status.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition D not met.	E.1 Intiate action to fully insert all rods.	1 hour
	AND	
	E.2 Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour
F. Three or more channels inoperable.	F.1 Open reactor trip breakers (RTBs).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.2	-- **- NOTES -** 1. Only required to be performed when not performed within previous 92 days. 2. Not required to be performed prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. -- Perform COT in accordance with Setpoint Program.	 Prior to reactor startup <u>AND</u> 4 hours after reducing power below P-6 <u>AND</u> 92 days thereafter
SR 3.3.2.3	-- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	 24 months
SR 3.3.2.4	-- **- NOTE -** Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	 24 months on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.3 Reactor Trip System (RTS) Intermediate Range Instrumentation

LCO 3.3.3 Four channels of RTS Intermediate Range Neutron Flux – High instrumentation shall be OPERABLE.

APPLICABILITY: MODE 1 with Power Range Neutron Flux below the P-10 interlock, MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable with THERMAL POWER ≥ P-6.	A.1 Place one inoperable channel in bypass or trip.	2 hours
	OR	
	A.2 Reduce THERMAL POWER to < P-6.	2 hours
	OR	
	A.3 Increase THERMAL POWER to > P-10.	2 hours
B. Two channels inoperable with THERMAL POWER ≥ P-6.	B.1.1 Place one inoperable channel in bypass.	2 hours
	AND	
	B.1.2 Place one inoperable channel in trip.	2 hours
	OR	
	B.2 Reduce THERMAL POWER to < P-6.	2 hours
	OR	
	B.3 Increase THERMAL POWER to > P-10.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or two channels inoperable with THERMAL POWER < P-6.	C.1	Restore three of four channels to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6
D. Three or more channels inoperable.	D.1	Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>		
	D.2	Reduce THERMAL POWER to < P-6.	2 hours
	<u>AND</u>		
	D.3	Be in MODE 3.	7 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.3.2	- NOTE - Only required to be performed when not performed within previous 92 days. Perform COT in accordance with Setpoint Program.	Prior to reactor startup <u>AND</u> 4 hours after reducing power below P-10 <u>AND</u> 92 days thereafter
SR 3.3.3.3	- NOTE - Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.3.4	- NOTE - Neutron detectors are excluded from response time testing. Verify RTS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.4 Reactor Trip System (RTS) Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.4 The RTS ESFAS instrumentation for each Function in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.4-1.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more Functions with one or two channels inoperable in MODE 1 or 2.	A.1	Restore three of four channels to OPERABLE status.	6 hours
B.	Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more Functions with three or more channels inoperable in MODE 1 or 2.	B.1	Be in MODE 3.	6 hours
C.	One or more Functions with one or two channels inoperable in MODE 3, 4, or 5.	C.1	Restore three of four channels to OPERABLE status.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met. <u>OR</u> One or more Functions with three or more channels inoperable in MODE 3, 4, or 5.	D.1	Initiate action to fully insert all rods.	1 hour
	<u>AND</u>		
	D.2	Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform ACTUATION LOGIC TEST.	92 days

Table 3.3.4-1 (page 1 of 1)
Reactor Trip System Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS
1. Safeguards Actuation Input from Engineered Safety Feature Actuation System – Automatic	1,2	4
2. ADS Stages 1, 2, and 3 Actuation Input from Engineered Safety Feature Actuation System – Automatic	1,2,3[(a)],4[(a)],5[(a)]	4
3. Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Automatic	1,2,3[(a)],4[(a)],5[(a)]	4

(a) With Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

3.3 INSTRUMENTATION

3.3.5 Reactor Trip System (RTS) Manual Actuation

LCO 3.3.5 The RTS manual actuation channels for each Function in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one manual actuation channel inoperable.	A.1 Restore manual actuation channel to OPERABLE status.	48 hours
B. Required Action and associated Completion Time of Condition A not met in MODE 1 or 2. <u>OR</u> One or more Functions with two manual actuation channels inoperable in MODE 1 or 2.	B.1 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 3, 4, or 5. OR One or more Functions with two manual actuation channels inoperable in MODE 3, 4, or 5.	C.1	Initiate action to fully insert all rods.	1 hour
	AND		
	C.2	Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	-- **- NOTE -** Verification of setpoint is not required. -- Perform TADOT.	24 months

Table 3.3.5-1 (page 1 of 1)
Reactor Trip System Manual Actuation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS
1. Manual Reactor Trip	1,2,3[(a)],4[(a)],5[(a)]	2
2. Safeguards Actuation Input from Engineered Safety Feature Actuation System – Manual	1,2	2
3. ADS Stages 1, 2, and 3 Actuation Input from Engineered Safety Feature Actuation System – Manual	1,2,3[(a)],4[(a)],5[(a)]	2 switch sets
4. Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Manual	1,2,3[(a)],4[(a)],5[(a)]	2 switch sets

(a) With Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

3.3 INSTRUMENTATION

3.3.6 Reactor Trip System (RTS) Automatic Trip Logic

LCO 3.3.6 Four divisions of RTS Automatic Trip Logic shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two divisions inoperable in MODE 1 or 2.	A.1 Restore three of four divisions to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Three or more divisions inoperable in MODE 1 or 2.	B.1 Be in MODE 3.	6 hours
C. One or two divisions inoperable in MODE 3, 4, or 5.	C.1 Restore three of four divisions to OPERABLE status.	48 hours
D. Required Action and associated Completion Time of Condition C not met. <u>OR</u> Three or more divisions inoperable in MODE 3, 4, or 5.	D.1 Initiate action to fully insert all rods. <u>AND</u> D.2 Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform ACTUATION LOGIC TEST.	92 days

3.3 INSTRUMENTATION

3.3.7 Reactor Trip System (RTS) Trip Actuation Devices

LCO 3.3.7 Four divisions of RTS trip actuation devices for the following Functions shall be OPERABLE:

a. Reactor Trip Breakers (RTBs); and

b. RTB Undervoltage and Shunt Trip Mechanisms.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both Functions within one division inoperable.	A.1 Open RTBs in inoperable division.	8 hours
B. One or both Functions within two divisions inoperable.	B.1 Restore one division to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1 or 2. <u>OR</u> One or both Functions within three or more divisions inoperable in MODE 1 or 2.	C.1 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 3, 4, or 5. OR One or both Functions within three or more divisions inoperable in MODE 3, 4, or 5.	D.1	Initiate action to fully insert all rods.	6 hours
	AND		
	D.2	Place the Plant Control System in a condition incapable of rod withdrawal.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform TADOT on both reactor trip breakers in one division.	92 days on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.8 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.8 The ESFAS instrumentation channels for each Function in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.8-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 Place inoperable channel in bypass or trip.	6 hours
B. One or more Functions with two channels inoperable.	B.1 Place one inoperable channel in bypass. <u>AND</u> B.2 Place one inoperable channel in trip.	6 hours 6 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> One or more Functions with three or more channels inoperable.	C.1 Enter the Condition referenced in Table 3.3.8-1 for the channel(s).	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.8-1.	D.1 Be in MODE 3.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.8-1.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 4.	12 hours
F. As required by Required Action C.1 and referenced in Table 3.3.8-1.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours
G. As required by Required Action C.1 and referenced in Table 3.3.8-1.	G.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	G.2	Be in MODE 4.	12 hours
	<u>AND</u>		
	G.3	Establish RCS cooling provided by RNS.	24 hours
H. As required by Required Action C.1 and referenced in Table 3.3.8-1.	H.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	H.2	Be in MODE 5	36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.8-1.	I.1	Declare affected isolation valve(s) inoperable.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
J. As required by Required Action C.1 and referenced in Table 3.3.8-1.	J.1	Be in MODE 5.	37 hours with three or more inoperable channels AND 180 hours
	AND		
	J.2	Initiate action to open the RCS pressure boundary and establish a pressurizer level ≥ 20%.	180 hours
K. As required by Required Action C.1 and referenced in Table 3.3.8-1.	K.1	Suspend positive reactivity additions.	Immediately
	AND		
	K.2	Initiate action to open RCS pressure boundary and establish ≥ 20% pressurizer level.	Immediately
L. As required by Required Action C.1 and referenced in Table 3.3.8-1.	L.1	Suspend positive reactivity additions.	Immediately
	AND		
	L.2	Initiate action to remove the upper internals.	Immediately
M. As required by Required Action C.1 and referenced in Table 3.3.8-1.	M.1	Suspend positive reactivity additions.	Immediately
	AND		
	M.2	Be in MODE 5.	12 hours
	AND		
	M.3	Initiate action to establish a pressurizer level ≥ 20% with the RCS pressure boundary intact.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
N. As required by Required Action C.1 and referenced in Table 3.3.8-1.	N.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	N.2	Initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
O. As required by Required Action C.1 and referenced in Table 3.3.8-1.	O.1	Declare affected isolation valve(s) inoperable.	Immediately
	<u>AND</u>		
	O.2	Be in MODE 3.	6 hours
P. As required by Required Action C.1 and referenced in Table 3.3.8-1.	P.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	P.2	Be in MODE 5.	36 hours
	<u>AND</u>		
	P.3	Open a containment air flow path ≥ 6 inches in diameter.	44 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	Perform CHANNEL OPERATIONAL TEST (COT) in accordance with Setpoint Program.	92 days
SR 3.3.8.3	-- **- NOTE –** This surveillance shall include verification that the time constants are adjusted to within limits. -- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.8.4	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

Table 3.3.8-1 (page 1 of 2)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1. Containment Pressure –Low 2	1,2,3,4,5[(a)],6[(a)]	4	P
2. Containment Pressure – High 2	1,2,3,4	4	H
3. Containment Radioactivity – High 1	1,2,3,4[(b)]	4	I
4. Containment Radioactivity – High 2	1,2,3	4	I
5. Pressurizer Pressure – Low	1,2,3[(c)]	4	E
6. Pressurizer Water Level – Low 1	1,2	4	D
7. Pressurizer Water Level – Low 2	1,2,3,4[(b)]	4	F
	4[(d)],5[(e)(f)]	4	J
8. Pressurizer Water Level – High 1	1,2,3	4	I
9. Pressurizer Water Level – High 2	1,2,3,4[(g)]	4	I
10. Pressurizer Water Level, High 3	1,2,3,4[(g)]	4	F
11. RCS Cold Leg Temperature (T_{cold}) – Low	1,2,3[(c)]	4 per loop	E
12. Reactor Coolant Average Temperature (T_{avg}) – Low 1	1,2	4	D
13. Reactor Coolant Average Temperature (T_{avg}) – Low 2	1,2	4	D
14. RCS Wide Range Pressure – Low	1,2,3,4	4	H
	5	4	K
	6[(h)]	4	L

(a) Without an open containment air flow path ≥ 6 inches in diameter.

(b) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(c) Above the P-11 (Pressurizer Pressure) interlock, when the RCS boron concentration is below that necessary to meet the SDM requirements at an RCS temperature of 200°F.

(d) With the RCS being cooled by the RNS.

(e) With the RCS pressure boundary intact.

(f) With RCS not being cooled by the RNS and with pressurizer level ≥ 20%.

(g) Above the P-19 (RCS Pressure) interlock with the RCS not being cooled by RNS.

(h) With upper internals in place.

Table 3.3.8-1 (page 2 of 2)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
15. Core Makeup Tank (CMT) Level – Low 1	1,2,3,4	4 per tank	H
	5(i)	4 per OPERABLE tank	J
16. CMT Level – Low 2	1,2,3,4	4 per tank	H
	5	4 per OPERABLE tank	J
17. Source Range Neutron Flux Doubling	2(j),3(j),4(l)	4	I
	5(l)	4	I
18. IRWST Level – Low 3	1,2,3,4(b)	4	F
	4(d),5	4	M
	6(h)	4	N
19. Reactor Coolant Pump Bearing Water Temperature – High	1,2,3,4	4 per RCP	O
20. SG Narrow Range Water Level – Low	1,2,3,4(b)	4 per SG	F
21. SG Wide Range Water Level – Low	1,2,3,4(b)	4 per SG	F
22. SG Narrow Range Water Level High	1,2,3,4	4 per SG	I
23. SG Narrow Range Water Level – High 2	1,2	4 per SG	D
	3,4	4 per SG	I
24. Steam Line Pressure – Low	1,2,3,4(b)	4 per steam line	G
25. Steam Line Pressure – Negative Rate – High	3(k)	4 per steam line	I

(b) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(d) With the RCS being cooled by the RNS.

(g) Above the P-19 (RCS Pressure) interlock with the RCS not being cooled by RNS.

(h) With upper internals in place.

(i) With RCS pressure boundary intact and with pressurizer level ≥ 20%.

(j) With unborated water source flow paths not isolated except when critical or except during intentional approach to criticality.

(k) Below the P-11 (Pressurizer Pressure) interlock.

(l) With unborated water source flow paths not isolated.

3.3 INSTRUMENTATION

3.3.9 Engineered Safety Feature Actuation System (ESFAS) Manual Initiation

LCO 3.3.9 The ESFAS manual initiation channels for each Function in Table 3.3.9-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.9-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Not applicable to Functions 1, 6, 7, 8, 12, and 13 in MODE 5 or 6. -------------------------------- One or more Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours
B. -------------------------------- **- NOTE -** Only applicable to Functions 1, 6, 7, 8, 12, and 13 in MODE 5 or 6. -------------------------------- One or more Functions with one channel inoperable.	B.1 Restore channel to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Requried Action and associated Completion Time of Condition A or B not met. <u>OR</u> One or more Functions with two channels inoperable.	C.1 Enter the Condition referenced in Table 3.3.9-1 for the channel(s).	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.9-1.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	6 hours 24 hours
E. As required by Required Action C.1 and referenced in Table 3.3.9-1.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.	6 hours 36 hours
F. As required by Required Action C.1 and referenced in Table 3.3.9-1.	F.1 Declare affected isolation valve(s) inoperable.	Immediately
G. As required by Required Action C.1 and referenced in Table 3.3.9-1.	G.1 Be in MODE 5. <u>AND</u> G.2 Initiate action to open the RCS pressure boundary.	12 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action C.1 and referenced in Table 3.3.9-1.	H.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	H.2	Initiate action to open RCS pressure boundary and establish ≥ 20% pressurizer level.	Immediately
I. As required by Required Action C.1 and referenced in Table 3.3.9-1.	I.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	I.2	Initiate action to remove the upper internals.	Immediately
J. As required by Required Action C.1 and referenced in Table 3.3.9-1.	J.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	J.2	Be in MODE 5.	12 hours
	<u>AND</u>		
	J.3	Initiate action to establish a pressurizer level ≥ 20% with the RCS pressure boundary intact.	12 hours
K. As required by Required Action C.1 and referenced in Table 3.3.9-1.	K.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	K.2	Initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. As required by Required Action C.1 and referenced in Table 3.3.9-1.	L.1 Be in MODE 3. <u>AND</u>	6 hours
	L.2 Be in MODE 5. <u>AND</u>	36 hours
	L.3 Open a containment air flow path ≥ 6 inches in diameter.	44 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	----------------------------- **- NOTE -** Verification of setpoint not required. -----------------------------	
	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT).	24 months

Table 3.3.9-1 (page 1 of 2)
Engineered Safeguards Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1.	Safeguards Actuation - Manual Initiation	1,2,3,4	2 switches	E
		5	2 switches	J
2.	Core Makeup Tank (CMT) Actuation - Manual Initiation	1,2,3,4[(a)]	2 switches	D
		4[(b)], 5[(c)]	2 switches	G
3.	Containment Isolation - Manual Initiation	1,2,3,4	2 switches	E
4.	Steam Line Isolation - Manual Initiation	1,2,3,4	2 switches	F
5.	Feedwater Isolation - Manual Initiation	1,2,3,4	2 switches	F
6.	ADS Stages 1, 2 & 3 Actuation - Manual Initiation	1,2,3,4	2 switch sets	E
		5[(d)]	2 switch sets	H
7.	ADS Stage 4 Actuation - Manual Initiation	1,2,3,4	2 switch sets	E
		5	2 switch sets	H
		6[(e)]	2 switch sets	I
8.	Passive Containment Cooling Actuation - Manual Initiation	1,2,3,4	2 switches	E
		5[(f)]	2 switches	J
		6[(f)]	2 switches	K
9.	Passive Residual Heat Removal Heat Exchanger Actuation - Manual Initiation	1,2,3,4	2 Switches	E
		5[(c)]	2 switches	G
10.	Chemical Volume and Control System Makeup Isolation - Manual Initiation	1,2,3,4[(a)]	2 switches	F
11.	Normal Residual Heat Removal System Isolation - Manual Initiation	1,2,3	2 switch sets	F

(a) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(b) With the RCS being cooled by the RNS.

(c) With the RCS pressure boundary intact.

(d) With RCS pressure boundary intact and with pressurizer level ≥ 20%.

(e) With upper internals in place.

(f) With decay heat > 6.0 MWt.

Table 3.3.9-1 (page 2 of 2)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
12. In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation - Manual Initiation	1,2,3,4[(a)]	2 switch sets	D
	4[(b)],5	2 switch sets	J
	6	2 switch sets	K
13. IRWST Containment Recirculation Valve Actuation - Manual Initiation	1,2,3,4[(a)]	2 switch sets	D
	4[(b)],5	2 switch sets	J
	6	2 switch sets	K
14. SG Power Operated Relief Valve and Block Valve Isolation - Manual Initiation	1,2,3,4[(a)]	2 switches	D
15. Containment Vacuum Relief Valve Actuation – Manual Initiation	1,2,3,4,5[(g)],6[(g)]	2 switches	L

(a) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(b) With the RCS being cooled by the RNS.

(g) Without an open containment air flow path ≥ 6 inches in diameter.

3.3 INSTRUMENTATION

3.3.10 Engineered Safety Feature Actuation System (ESFAS) Reactor Coolant System (RCS) Hot Leg Level Instrumentation

LCO 3.3.10 The ESFAS RCS Hot Leg Level instrumentation channels for each function in Table 3.3.10-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.10-1.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place inoperable channel in bypass. <u>AND</u> A.2 ------------------------------------ **- NOTE -** Only applicable to Function 1. ------------------------------------ Continuously monitor hot leg level.	6 hours 6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Enter the Condition referenced in Table 3.3.10-1 for the channel.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action B.1 and referenced in Table 3.3.10-1.	C.1	Suspend positive reactivity additions.	Immediately
	AND		
	C.2	Be in MODE 5.	12 hours
	AND		
	C.3	Initiate action to establish a pressurizer level ≥ 20% with the RCS pressure boundary intact.	12 hours
D. As required by Required Action B.1 and referenced in Table 3.3.10-1.	D.1	Suspend positive reactivity additions.	Immediately
	AND		
	D.2	Initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action B.1 and referenced in Table 3.3.10-1.	-- **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. --	
	E.1.1 Isolate the affected flow path(s).	24 hours
	<u>AND</u>	
	E.1.2.1 Isolate the affected flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	7 days
	<u>OR</u>	
	E.1.2.2 Verify the affected flow path is isolated	Once per 7 days
	<u>OR</u>	
	E.2.1 Be in MODE 5.	12 hours
	<u>AND</u>	
	E.2.2 Initiate action to establish a pressurizer level ≥ 20%	12 hours
F. As required by Required Action B.1 and referenced in Table 3.3.10-1.	F.1 Initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.10.2	Perform CHANNEL OPERATIONAL TEST (COT) in accordance with Setpoint Program.	92 days
SR 3.3.10.3	-- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to within limits. -- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.10.4	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

Table 3.3.10-1 (page 1 of 1)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1. Hot Leg Level – Low 2	4(a),5	1 per loop	C
	6(b)	1 per loop	D
2. Hot Leg Level – Low 1	4(a)(c),5(c)	1 per loop	E
	6(c)(d)	1 per loop	F

(a) With the RCS being cooled by the RNS.

(b) With upper internals in place.

(c) Below the P-12 (Pressurizer Level) interlock.

(d) With the water level < 23 feet above the top of the reactor vessel flange.

3.3 INSTRUMENTATION

3.3.11 Engineered Safety Feature Actuation System (ESFAS) Startup Feedwater Flow Instrumentation

LCO 3.3.11 Two channels of ESFAS Startup Feedwater Flow instrumentation for each startup feedwater line shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTE -

Separate condition entry is allowed for each startup feedwater line.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more startup feedwater lines with one channel inoperable.	A.1 Place channel in trip.	6 hours
B. Required Action and associated Completion Time of Condition A not met. OR One or more startup feedwater lines with two channels inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.11.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.11.2	Perform CHANNEL OPERATIONAL TEST (COT) in accordance with Setpoint Program.	92 days
SR 3.3.11.3	- NOTE - This surveillance shall include verification that the time constants are adjusted to within limits. Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.11.4	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.12 Engineered Safety Feature Actuation System (ESFAS) Reactor Trip Initiation

LCO 3.3.12 Three ESFAS Reactor Trip (P-4) divisions shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required division inoperable.	A.1	Restore required division to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two or three required divisions inoperable.	B.1	Declare affected isolation valve(s) inoperable.	Immediately
	<u>AND</u>		
	B.2	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.3	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.12.1	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT).	24 months

3.3 INSTRUMENTATION

3.3.13 Engineered Safety Feature Actuation System (ESFAS) Control Room Air Supply Radiation Instrumentation

LCO 3.3.13 Two channels of ESFAS Control Room Air Supply Radiation - High 2 instrumentation shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable in MODE 1, 2, 3, or 4.	A.1 Verify alternate radiation monitors are OPERABLE. <u>AND</u> A.2 Verify control room isolation and air supply initiation manual controls are OPERABLE.	72 hours 72 hours
B. One channel inoperable during movement of irradiated fuel assemblies.	B.1 Restore channel to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two channels inoperable in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B not met. OR Two channels inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.13.2	Perform CHANNEL OPERATIONAL TEST (COT) in accordance with Setpoint Program.	92 days
SR 3.3.13.3	- NOTE - This surveillance shall include verification that the time constants are adjusted to within limits. Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.13.4	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.14 Engineered Safety Feature Actuation System (ESFAS) Spent Fuel Pool Level Instrumentation

LCO 3.3.14 Three channels of ESFAS Spent Fuel Pool Level – Low instrumentation shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place channel in trip.	6 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two or more channels inoperable.	-- **- NOTE -** Flow path(s) may be unisolated intermittently under administrative controls. --	
	B.1 Isolate the affected flow path(s).	24 hours
	<u>AND</u>	
	B.2.1 Isolate the affected flow path(s) by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	7 days
	<u>OR</u>	
	B.2.2 Verify the affected flow path is isolated.	Once per 7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.14.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.14.2	Perform CHANNEL OPERATIONAL TEST (COT) in accordance with Setpoint Program.	92 days
SR 3.3.14.3	--- **- NOTE -** This surveillance shall include verification that the time constants are adjusted to within limits. --- Perform CHANNEL CALIBRATION in accordance with Setpoint Program.	24 months
SR 3.3.14.4	Verify ESF RESPONSE TIME is within limit.	24 months on a STAGGERED TEST BASIS

3.3 INSTRUMENTATION

3.3.15 Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Operating

LCO 3.3.15 Four divisions with one subsystem for each of the following Functions shall be OPERABLE:

a. ESF Coincidence Logic; and

b. ESF Actuation.

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

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one division inoperable.	A.1 Restore division to OPERABLE status.	6 hours
B. Required Action and associated Completion Time of Condition A not met. OR One or more Functions within two or more divisions inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.15.2	-------------------- **- NOTE -** Only required to be met in MODE 4 above the P-19 (RCS Pressure) interlock with the RCS not being cooled by RNS. -------------------- Verify pressurizer heater circuit breakers trip open on an actual or simulated actuation signal.	24 months
SR 3.3.15.3	Verify reactor coolant pump breakers trip open on an actual or simulated actuation signal.	24 months
SR 3.3.15.4	-------------------- **- NOTE -** Only required to be met in MODE 4 with the RCS being cooled by the RNS or below the P-12 (Pressurizer Level) interlock. -------------------- Verify CVS letdown isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.3.15.5	Verify main feedwater and startup feedwater pump breakers trip open on an actual or simulated actuation signal.	24 months
SR 3.3.15.6	-------------------- **- NOTE -** Only required to be met in MODES 1 and 2. -------------------- Verify auxiliary spray and purification line isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months

3.3 INSTRUMENTATION

3.3.16 Engineered Safety Feature Actuation System (ESFAS) Actuation Logic – Shutdown

LCO 3.3.16 Four divisions with one subsystem for each of the following Functions shall be OPERABLE:

a. ESF Coincidence Logic; and

b. ESF Actuation.

- NOTE -

Only the divisions necessary to support Main Control Room Isolation and Air Supply Initiation are required to be OPERABLE during movement of irradiated fuel assemblies when not in MODE 1, 2, 3, 4, 5, or 6.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

Separate condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions within one required division inoperable.	A.1 Restore required division to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met in MODE 5. <u>OR</u> One or more Functions within two or more divisions inoperable in MODE 5.	B.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	B.2	Initiate action to open RCS pressure boundary and establish ≥ 20% pressurizer level.	Immediately
	<u>AND</u>		
	B.3	Initiate action to isolate the flow path from the demineralized water storage tank to the RCS by use of at least one closed and de-activated automatic valve or closed manual valve.	Immediately
C. Required Action and associated Completion Time of Condition A not met in MODE 6. <u>OR</u> One or more Functions within two or more divisions inoperable in MODE 6.	C.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	C.2	Initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies. <u>OR</u> One or more Functions within two or more required divisions inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.16.1	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.16.2	-------------------- **- NOTE -** -------------------- Only required to be met in MODE 5. -- Verify reactor coolant pump breakers trip open on an actual or simulated actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.16.3	-- **- NOTES -** 1. Not required to be met in MODE 5 above the P-12 (Pressurizer Level) interlock. 2. Not required to be met in MODE 6 above the P-12 (Pressurizer Level) interlock and water level $\geq$ 23 feet above the top of the reactor vessel flange. -- Verify CVS letdown isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.3.16.4	-- **- NOTE -** Only required to be met in MODE 6. -- Verify Spent Fuel Pool Cooling System containment isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months

3.3 INSTRUMENTATION

3.3.17 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.17 The PAM instrumentation for each Function in Table 3.3.17-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTES -

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1	Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action in accordance with Specification 5.6.5.	Immediately
C. One or more Functions with two required channels inoperable.	C.1	Restore one channel to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1	Enter the Condition referenced in Table 3.3.17-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.17-1.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

- NOTE -
SR 3.3.17.1 and SR 3.3.17.2 apply to each PAM instrumentation Function in Table 3.3.17-1.

	SURVEILLANCE	FREQUENCY
SR 3.3.17.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.17.2	---------- **- NOTE -** Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	24 months

Table 3.3.17-1 (page 1 of 1)
Post-Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITION REFERENCED FROM REQUIRED ACTION D.1
1.	Neutron Flux (Intermediate Range)	2	E
2.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2	E
3.	RCS Cold Leg Temperature (Wide Range)	2	E
4.	RCS Pressure (Wide Range)	2	E
5.	RCS Subcooling Monitor	2	E
6.	Containment Water Level	2	E
7.	Containment Pressure	2	E
8.	Containment Pressure (Extended Range)	2	E
9.	Containment Area Radiation (High Range)	2	E
10.	Pressurizer Level and Associated Reference Leg Temperature	2	E
11.	In-Containment Refueling Water Storage Tank (IRWST) Water Level	2	E
12.	Passive Residual Heat Removal (PRHR) Heat Removal	2	E
13.	Core Exit Temperature -- Quadrant 1	2[(a)]	E
14.	Core Exit Temperature -- Quadrant 2	2[(a)]	E
15.	Core Exit Temperature -- Quadrant 3	2[(a)]	E
16.	Core Exit Temperature -- Quadrant 4	2[(a)]	E
17.	Passive Containment Cooling System (PCS) Heat Removal	2	E
18.	Penetration Flow Path Remotely Operated Containment Isolation Valve Position	2 per penetration flow path[(b)(c)]	E
19.	IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status	2	E

(a) A channel consists of two thermocouples within a single division. Each quadrant contains two divisions. The minimum requirement is two OPERABLE thermocouples in each of the two divisions.

(b) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(c) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

3.3 INSTRUMENTATION

3.3.18 Remote Shutdown Workstation (RSW)

LCO 3.3.18 The RSW shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with Reactor Coolant System (RCS) average temperature (T_{avg}) ≥ 350°F.

ACTIONS

- NOTE -

LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RSW inoperable.	A.1	Restore to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 4 with T_{avg} < 350°F.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.18.1	Verify each required transfer switch is capable of performing the required function.	24 months
SR 3.3.18.2	Verify the RSW communicates indication and controls with Division A, B, C and D of the Protection and Safety Monitoring System (PMS).	24 months
SR 3.3.18.3	Verify the OPERABILITY of the RSW hardware and software.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.18.4	Perform TADOT of the reactor trip breaker open/closed indication.	24 months

3.3 INSTRUMENTATION

3.3.19 Diverse Actuation System (DAS) Manual Controls

LCO 3.3.19 The DAS manual controls for each function in Table 3.3.19-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.19-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more manual DAS controls inoperable.	A.1	Restore DAS manual controls to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met for inoperable DAS manual reactor trip control.	B.1	Perform SR 3.3.7.1.	Once per 31 days on a STAGGERED TEST BASIS
	<u>AND</u>		
	B.2	Restore all controls to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
C. Required Action and associated Completion Time of Condition A not met for inoperable DAS manual actuation control other than reactor trip.	C.1	Perform SRs 3.3.15.1 and 3.3.16.1, as applicable.	Once per 31 days on a STAGGERED TEST BASIS
	<u>AND</u>		
	C.2	Restore all controls to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition B not met. OR Required Action and associated Completion Time of Condition C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.19.1	-------------------- - NOTE - -------------------- Verification of setpoint not required. -- Perform TRIP ACTUATION DEVICE OPERATIONAL TEST (TADOT).	24 months

Table 3.3.19-1 (page 1 of 1)
DAS Manual Controls

FUNCTION		APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CONTROLS
1.	Reactor trip manual controls	1,2	2 switches
2.	Passive Residual Heat Removal Heat Exchanger (PRHR HX) control and In-Containment Refueling Water Storage Tank (IRWST) gutter control valves	1,2,3,4,5[(a)]	2 switches
3.	Core Makeup Tank (CMT) isolation valves	1,2,3,4,5[(a)]	2 switches
4.	Automatic Depressurization System (ADS) stage 1 valves	1,2,3,4,5[(a)]	2 switches
5.	ADS stage 2 valves	1,2,3,4,5[(a)]	2 switches
6.	ADS stage 3 valves	1,2,3,4,5[(a)]	2 switches
7.	ADS stage 4 valves	1,2,3,4,5,6[(c)]	2 switches
8.	IRWST injection squib valves	1,2,3,4,5,6	2 switches
9.	Containment recirculation valves	1,2,3,4,5,6	2 switches
10.	Passive containment cooling drain valves	1,2,3,4,5[(b)],6[(b)]	2 switches
11.	Selected containment isolation valves	1,2,3,4,5,6	2 switches

(a) With Reactor Coolant System (RCS) pressure boundary intact.

(b) With the reactor decay heat > 6.0 MWt.

(c) With reactor internals in place.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer Pressure is greater than or equal to the limit specified in the COLR

b. RCS Average Temperature is less than or equal to the limit specified in the COLR, and

c. RCS total flow rate ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

- NOTE -

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute, or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	Perform a CHANNEL CALIBRATION of differential pressure RCS total flow rate indication channels.	24 months
SR 3.4.1.5	-- **- NOTE -** Not required to be performed until 24 hours after ≥ 90% RTP. -- Verify RCS total flow rate is ≥ 301,670 gpm and greater than or equal to the limit specified in the COLR as determined by precision heat balance or differential pressure RCS total flow rate indication measurements.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ 551°F.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Action A.2 shall be completed whenever this Condition is entered. -------------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameters to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -------------------------------- **- NOTE -** Required Action C.2 shall be completed whenever this Condition is entered. -------------------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-- **- NOTE -** Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in the PTLR.	 30 minutes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops

LCO 3.4.4 Two RCS loops shall be OPERABLE with four Reactor Coolant Pumps (RCPs) in operation with variable speed control bypassed.

- NOTES -

1. No RCP shall be started when the RCS temperature is ≥ 350°F unless pressurizer level is < 92%.

2. No RCP shall be started with any RCS cold leg temperature ≤ 350°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures and the RCP is started at ≤ 25% of RCP speed.

3. All RCPs may be removed from operation in MODE 3, 4, or 5 for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with Plant Control System capable of rod withdrawal or one or more rods not fully inserted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Actions must be completed whenever Condition A is entered. -------------------------------- Requirements of LCO not met in MODE 1 or 2.	A.1 Suspend start of any RCP. <u>AND</u> A.2 Be in MODE 3. <u>AND</u>	Immediately 6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Initiate action to fully insert all rods.	6 hours
	AND		
	A.4	Place the Plant Control System in a condition incapable of rod withdrawal.	6 hours
B. --------------------------------- **- NOTE -** Required Actions must be completed whenever Condition B is entered. -------------------------------- Requirements of LCO not met in MODE 3, 4, or 5.	B.1	Suspend start of any RCP.	Immediately
	AND		
	B.2	Initiate action to fully insert all rods.	1 hour
	AND		
	B.3	Place the Plant Control System in a condition incapable of rod withdrawal.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation with variable speed control bypassed.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 Pressurizer

LCO 3.4.5 The pressurizer water level shall be ≤ 92% of span.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Pressurizer water level not within limit.	A.1	Be in MODE 3.	6 hours
	AND		
	A.2	Initiate action to fully insert all rods.	6 hours
	AND		
	A.3	Place the Plant Control System in a condition incapable of rod withdrawal.	6 hours
	AND		
	A.4	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify pressurizer water level ≤ 92% of span.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 Pressurizer Safety Valves

LCO 3.4.6 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with Normal Residual Heat Removal System (RNS) isolated,
MODE 4 with RCS temperature ≥ 275°F.

- NOTE -

The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. One pressurizer safety valve at a time may be inoperable for hot lift setting adjustment.

This exception is allowed for 36 hours following entry into MODE 3, provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with RNS aligned to the RCS and RCS temperature < 275°F.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify each pressurizer safety valve OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Operational LEAKAGE

LCO 3.4.7 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 0.5 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE from the RCS,

d. 150 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG), and

e. 500 gallons per day primary to In-Containment Refueling Water Storage Tank (IRWST) LEAKAGE through the passive residual heat removal heat exchanger (PRHR HX).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	--- **- NOTES -** 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.7.2	--- **- NOTE -** Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Minimum RCS Flow

LCO 3.4.8 At least one Reactor Coolant Pump (RCP) shall be in operation with a total flow through the core of ≥ 3,000 gpm.

- NOTES -

1. All RCPs may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started when the RCS temperature is ≥ 350°F unless pressurizer level is < 92%.

3. No RCP shall be started with any RCS cold leg temperature ≤ 350°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures and the RCP is started at ≤ 25% of RCP speed.

APPLICABILITY: MODES 3, 4, and 5 with Plant Control System incapable of rod withdrawal, all rods fully inserted, and unborated water sources not isolated from the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Action A.2 shall be completed prior to starting any RCP whenever this Condition is entered. -------------------------------- No RCP in operation.	A.1 Isolate all sources of unborated water. AND A.2 Perform SR 3.1.1.1.	1 hour 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify at least one RCP is in operation with total flow through the core ≥ 3,000 gpm.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Leakage Detection Instrumentation

LCO 3.4.9 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Two containment sump level channels; and

b. One containment atmosphere F18 particulate monitor.

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

- NOTES -

1. The containment atmosphere F18 particulate monitor is only required to be OPERABLE in MODE 1 with RTP > 20%.

2. Containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump such as by redirection to the In-Containment Refueling Water Storage Tank (IRWST) by the containment shell gutter drains.

ACTIONS

- NOTE -

LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required containment sump channel inoperable.	A.1	Verify that the volume input per day to the containment sump does not change (+ or -) more than 10 gallons or 33% of the volume input (whichever is greater). The volume used for comparison will be the value taken during the first day following the entrance into this CONDITION.	Once per 24 hours
	AND		

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	Restore two containment sump channels to OPERABLE status.	14 days
B. Two required containment sump channels inoperable.	B.1	----------------------------------- **- NOTE -** Not required until 12 hours after establishment of steady state operation. -----------------------------------	
		Perform SR 3.4.7.1.	Once per 24 hours
	<u>AND</u>		
	B.2	Restore one containment sump channel to OPERABLE status.	72 hours
C. Containment atmosphere F18 particulate monitor inoperable.	C.1.1	Analyze grab samples of containment atmosphere.	Once per 24 hours
	<u>OR</u>		
	C.1.2	----------------------------------- **- NOTE -** Not required until 12 hours after establishment of steady state operation. -----------------------------------	
		Perform SR 3.4.7.1.	Once per 24 hours
	<u>AND</u>		
	C.2	Restore containment atmosphere F18 particulate monitor to OPERABLE status.	30 days
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 5.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. All required monitors inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Perform a CHANNEL CHECK of containment atmosphere F18 particulate monitor.	12 hours
SR 3.4.9.2	Perform a COT of containment atmosphere F18 particulate monitor.	92 days
SR 3.4.9.3	Perform a CHANNEL CALIBRATION of required containment sump monitor.	24 months
SR 3.4.9.4	Perform a CHANNEL CALIBRATION of containment atmosphere F18 particulate monitor.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Specific Activity

LCO 3.4.10 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2.
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 µCi/gm.	-- **- NOTE -** LCO 3.0.4 is not applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 > 280 µCi/gm.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 > 60 µCi/gm.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 μCi/gm.	7 days
SR 3.4.10.2	-- **- NOTE -** Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 to 6 hours after a THERMAL POWER change of ≥ 15% of RTP within a 1 hour period

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Automatic Depressurization System (ADS) – Operating

LCO 3.4.11 Ten ADS flow paths shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One stage 1, 2, or 3 ADS flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. One stage 4 ADS flow path inoperable.	B.1 Restore flow path to OPERABLE status.	72 hours
C. Two or three ADS flow paths inoperable with a combined inoperable flow capacity less than or equal to that of a division with the largest ADS flow capacity.	C.1 Restore flow paths to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR LCO not met for reasons other than Condition A, B, or C.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify the motor operated valve in series with each 4th stage ADS valve is fully open.	12 hours
SR 3.4.11.2	Verify each stage 1, 2, and 3 ADS valve strokes open.	In accordance with the Inservice Testing Program
SR 3.4.11.3	Verify each stage 4 ADS valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.4.11.4	Verify each stage 1, 2, and 3 ADS valve actuates to the open position on an actual or simulated actuation signal.	24 months
SR 3.4.11.5	--- **- NOTE -** Squib actuation may be excluded. --- Verify continuity of the circuit from the Protection Logic Cabinets to each stage 4 ADS valve.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Automatic Depressurization System (ADS) – Shutdown, RCS Intact

LCO 3.4.12 Nine ADS flow paths shall be OPERABLE.

APPLICABILITY: MODE 5 with RCS pressure boundary intact.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required stage 1, 2, or 3 ADS flow path inoperable.	A.1 Restore required flow path to OPERABLE status.	7 days
B. One required stage 4 ADS flow path inoperable.	B.1 Restore required flow path to OPERABLE status.	72 hours
C. Two or three required ADS flow paths inoperable with a combined inoperable flow capacity less than or equal to that of a division with the largest ADS flow capacity.	C.1 Restore required flow paths to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. <u>OR</u> LCO not met for reasons other than Condition A, B, or C.	D.1 Initiate action to open the RCS pressure boundary.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	For flow paths required to be OPERABLE, the SRs of LCO 3.4.11, "Automatic Depressurization System (ADS) – Operating" are applicable.	In accordance with applicable SRs

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 Automatic Depressurization System (ADS) – Shutdown, RCS Open

LCO 3.4.13 ADS stage 1, 2, and 3 flow paths shall be open.
Two ADS stage 4 flow paths shall be OPERABLE.

- NOTE -

In MODE 5, the ADS valves may be closed to facilitate RCS vacuum fill operations to establish a pressurizer level ≥ 20%, provided ADS valve OPERABILITY meets LCO 3.4.12, ADS – Shutdown, RCS Intact.

APPLICABILITY: MODE 5 with pressurizer level < 20%,
MODE 5 with RCS pressure boundary open,
MODE 6 with upper internals in place.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ADS stage 1, 2, or 3 flow path not open.	A.1 Open the affected flow path.	72 hours
	<u>OR</u>	
	A.2 Open an alternative flow path with an equivalent area.	72 hours
B. One required ADS stage 4 flow path inoperable.	B.1 Open an alternative flow path with an equivalent area.	36 hours
	<u>OR</u>	
	B.2 Restore required ADS stage 4 flow paths to OPERABLE status.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met while in MODE 5. <u>OR</u> LCO not met for reasons other than Condition A or B while in MODE 5.	C.1 Initiate action to fill the RCS to establish ≥ 20% pressurizer level. <u>AND</u> C.2 Suspend positive reactivity additions.	Immediately Immediately
D. Required Action and associated Completion Time of Condition A or B not met while in MODE 6. <u>OR</u> LCO not met for reasons other than Condition A or B while in MODE 6.	D.1 Initiate action to remove the upper internals. <u>AND</u> D.2 Suspend positive reactivity additions.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	Verify each ADS stage 1, 2, and 3 valve is in the open position.	12 hours
SR 3.4.13.2	For each ADS stage 4 flow path required to be OPERABLE, the following SRs are applicable: SR 3.4.11.1 SR 3.4.11.3 SR 3.4.11.5	In accordance with applicable SRs

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 Low Temperature Overpressure Protection (LTOP)

LCO 3.4.14 At least one of the following overpressure protection methods shall be OPERABLE, with the accumulators isolated:

a. The Normal Residual Heat Removal System (RNS) suction relief valve with lift setting within the limit specified in the PTLR; or

b. The RCS depressurized and an RCS vent of ≥ 4.15 square inches.

- NOTES -

1. No reactor coolant pump (RCP) shall be started when the RCS temperature is ≥ 350°F unless pressurizer level is < 92%.

2. No RCP shall be started with any RCS cold leg temperature ≤ 350°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures and the RCP is started at ≤ 25% of RCP speed.

3. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

APPLICABILITY: MODE 4 when any cold leg temperature is ≤ 275°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	A.1 Isolate affected accumulator.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1	Increase RCS cold leg temperature to a level acceptable for the existing accumulator pressure allowed in the PTLR.	12 hours
	<u>OR</u>		
	B.2	Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
C. Required LTOP method inoperable for reasons other than Condition A or B.	C.1	Restore the RNS suction relief valve to OPERABLE status.	12 hours
	<u>OR</u>		
	C.2	Depressurize RCS and establish RCS vent of ≥ 4.15 square inches.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-- **- NOTE -** Only required to be met when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. -- Verify each accumulator is isolated.	12 hours

SURVEILANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.14.2	-- **- NOTE -** Only required to be met when complying with LCO 3.4.14.a. -- Verify both RNS suction isolation valves in one RNS suction flow path are open.	12 hours
SR 3.4.14.3	-- **- NOTE -** Only required to be met when complying with LCO 3.4.14.b. -- Verify RCS vent ≥ 4.15 square inches is open.	12 hours for unlocked-open vent <u>AND</u> 31 days for locked-open vent
SR 3.4.14.4	-- **- NOTE -** Only required to be met when complying with LCO 3.4.14.a. -- Verify the lift setting of the RNS suction relief valve.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Pressure Isolation Valve (PIV) Integrity

LCO 3.4.15 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4, with the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTES -

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Leakage from one or more RCS PIVs not within limit.	------------------------------------ **- NOTE -** Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.15.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. ------------------------------------	
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	8 hours
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Verify a second OPERABLE PIV can meet the leakage limits. This valve is required to be a check valve, or a closed valve, if it isolates a line that penetrates containment.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Verify leakage of each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 and $\leq$ 2255 psig.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Reactor Vessel Head Vent (RVHV)

LCO 3.4.16 The Reactor Vessel Head Vent shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path to OPERABLE status.	72 hours
B. Two flow paths inoperable.	B.1 Restore at least one flow path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 4, with the RCS cooling provided by the RNS.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.16.1 Verify each RVHV valve strokes open.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

- NOTE -

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Once prior to entering MODE 4 following a SG tube inspection

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.1 Accumulators

LCO 3.5.1 Both accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.
MODES 3 and 4 with Reactor Coolant System (RCS) pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration outside limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour from discovery of LCO 3.5.1 Condition B entry concurrent with LCO 3.5.2 Condition C or E entry <u>AND</u> 8 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify the borated water volume in each accumulator is ≥ 1667 cu. ft., and ≤ 1732 cu. ft.	12 hours
SR 3.5.1.3	Verify the nitrogen cover gas pressure in each accumulator is ≥ 637 psig and ≤ 769 psig.	12 hours
SR 3.5.1.4	Verify the boron concentration in each accumulator is ≥ 2600 ppm and ≤ 2900 ppm.	31 days <u>AND</u> ------------------------- **- NOTE -** Only required for affected accumulators. ------------------------- Once within 6 hours after each solution volume increase of ≥ 51 cu. ft. that is not the result of addition from the in-containment refueling water storage tank
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when pressurizer pressure is ≥ 2000 psig.	31 days
SR 3.5.1.6	Verify system flow performance of each accumulator in accordance with the System Level OPERABILITY Testing Program.	10 years

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.2 Core Makeup Tanks (CMTs) – Operating

LCO 3.5.2 Both CMTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CMT inoperable due to one CMT outlet isolation valve inoperable.	A.1 Restore outlet isolation valve to OPERABLE status.	72 hours
B. One CMT inoperable due to water temperature or boron concentration not within limits.	B.1 Restore water temperature and boron concentration to within limits.	72 hours
C. Two CMTs inoperable due to water temperature or boron concentration not within limits.	C.1 Restore water temperature and boron concentration to within limits for one CMT.	1 hour from discovery of LCO 3.5.2 Condition C entry concurrent with LCO 3.5.1 Condition B entry <u>AND</u> 8 hours
D. One CMT inlet line with noncondensible gas volume not within limit.	D.1 Restore CMT inlet line noncondensible gas volume to within limit.	24 hours
E. One CMT inoperable for reasons other than Condition A, B, or D.	E.1 Restore CMT to OPERABLE status.	1 hour from discovery of LCO 3.5.2 Condition E entry concurrent with LCO 3.5.1 Condition B entry <u>AND</u> 8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> Two CMTs inoperable for reasons other than Condition C.	F.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	F.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the temperature of the borated water in each CMT is < 120°F.	24 hours
SR 3.5.2.2	Verify the borated water volume in each CMT is ≥ 2487 cu. ft.	7 days
SR 3.5.2.3	Verify each CMT inlet isolation valve is fully open.	12 hours
SR 3.5.2.4	Verify the volume of noncondensible gases in each CMT inlet line has not caused the high-point water level to drop below the sensor.	24 hours
SR 3.5.2.5	Verify the boron concentration in each CMT is ≥ 3400 ppm, and ≤ 3700 ppm.	7 days
SR 3.5.2.6	Verify each CMT outlet isolation valve strokes open.	In accordance with the Inservice Testing Program
SR 3.5.2.7	Verify each CMT outlet isolation valve actuates to the open position on an actual or simulated actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.8	Verify system flow performance of each CMT in accordance with the System Level OPERABILITY Testing Program.	10 years

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.3 Core Makeup Tanks (CMTs) – Shutdown, Reactor Coolant System (RCS) Intact

LCO 3.5.3 One CMT shall be OPERABLE.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS).
MODE 5 with the RCS pressure boundary intact.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required CMT inoperable due to one outlet isolation valve inoperable.	A.1 Restore required isolation valve to OPERABLE status.	72 hours
B. Required CMT inoperable due to water temperature or boron concentration not within limits.	B.1 Restore water temperature and boron concentration to within limits.	72 hours
C. Required CMT inoperable for reasons other than Condition A or B.	C.1 Restore required CMT to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Initiate action to be in MODE 5 with RCS pressure boundary open.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	For the CMT required to be OPERABLE, the SRs of Specification 3.5.2, “Core Makeup Tanks (CMTs) – Operating” are applicable.	In accordance with applicable SRs

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.4 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating

LCO 3.5.4 The PRHR HX shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with the Reactor Coolant System (RCS) not being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One air operated PRHR HX outlet isolation valve inoperable.	A.1 Restore air operated PRHR HX outlet isolation valve to OPERABLE status.	72 hours
B. One air operated In-Containment Refueling Water Storage Tank (IRWST) gutter isolation valve inoperable.	B.1 Restore air operated IRWST gutter isolation valve to OPERABLE status.	72 hours
C. PRHR HX inlet line noncondensible gas volume not within limit.	C.1 Restore PRHR HX inlet line noncondensible gas volume to within limit.	24 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours
E. LCO not met for reasons other than Condition A, B, or C.	E.1 Restore PRHR HX to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time of Condition E not met.	F.1	------------------------------------ **- NOTE -** If redundant means of providing steam generator (SG) feedwater are not available, suspend LCO 3.0.3 and all other LCO Required Actions requiring MODE changes until redundant means are available. ------------------------------------	
		Be in MODE 3.	6 hours from discovery of redundant means of providing SG feedwater
	<u>AND</u>		
	F.2	------------------------------------ **- NOTE -** If redundant means of cooling the RCS to MODE 5 are not available, suspend LCO 3.0.3 and all other LCO Required Actions requiring MODE changes until redundant means are available. ------------------------------------	
		Be in MODE 5.	36 hours from discovery of redundant means of cooling the RCS to MODE 5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify the PRHR HX outlet manual isolation valve is fully open.	12 hours
SR 3.5.4.2	Verify the PRHR HX inlet motor operated isolation valve is open.	12 hours
SR 3.5.4.3	Verify the volume of noncondensible gases in the PRHR HX inlet line has not caused the high-point water level to drop below the sensor.	24 hours
SR 3.5.4.4	-- **- NOTE -** Only required to be met when one or more reactor coolant pumps (RCPs) are in operation. -- Verify one Loop 1 RCP is in operation.	12 hours
SR 3.5.4.5	Verify power is removed from the PRHR HX inlet motor operated isolation valve.	31 days
SR 3.5.4.6	Verify both PRHR HX air operated outlet isolation valves and both IRWST gutter isolation valves stroke open.	In accordance with the Inservice Testing Program
SR 3.5.4.7	Verify by visual inspection that the IRWST gutter and downspout screens are not restricted by debris.	24 months
SR 3.5.4.8	Verify both PRHR HX air operated outlet isolation valves actuate to the open position and both IRWST gutter isolation valves actuate to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.5.4.9	Verify PRHR HX heat transfer performance in accordance with the System Level OPERABILITY Testing Program.	10 years

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.5 Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, Reactor Coolant System (RCS) Intact

LCO 3.5.5 The PRHR HX shall be OPERABLE.

APPLICABILITY: MODE 4 with the RCS cooling provided by the Normal Residual Heat Removal System (RNS).
MODE 5 with the RCS pressure boundary intact and pressurizer level ≥ 20%.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One air operated PRHR HX outlet isolation valve inoperable.	A.1 Restore air operated PRHR HX outlet valve to OPERABLE status.	72 hours
B. One air operated In-Containment Refueling Water Storage Tank (IRWST) gutter isolation valve inoperable.	B.1 Restore air operated IRWST gutter isolation valve to OPERABLE status.	72 hours
C. PRHR HX inlet line noncondensible gas volume not within limit.	C.1 Restore PRHR HX inlet line noncondensible gas volume to within limit.	24 hours
D. PRHR HX inoperable for reasons other than Condition A, B, or C.	D.1 Restore PRHR HX to OPERABLE status.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Initiate action to be in MODE 5 with the RCS pressure boundary open.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	The SRs of Specification 3.5.4, “Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Operating” are applicable.	In accordance with applicable SRs

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.6 In-containment Refueling Water Storage Tank (IRWST) – Operating

LCO 3.5.6 The IRWST, with two injection flow paths and two containment recirculation flow paths, shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One IRWST injection line actuation valve flow path inoperable. OR One containment recirculation line actuation valve flow path inoperable.	A.1	Restore the inoperable actuation valve flow path to OPERABLE status.	72 hours
B. One IRWST injection flow path with noncondensible gas volume in one squib valve outlet line pipe stub not within limit.	B.1	Restore noncondensible gas volume in squib valve outlet line pipe stub to within limit.	72 hours
C. One IRWST injection flow path with noncondensible gas volume in both squib valve outlet line pipe stubs not within limit.	C.1	Restore noncondensible gas volume in one squib valve outlet line pipe stub to within limit.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. IRWST boron concentration not within limits. <u>OR</u> IRWST borated water temperature not within limits. <u>OR</u> IRWST borated water volume < 73,100 cu. ft. and $\geq$ 70,907 cu. ft.	D.1	Restore IRWST to OPERABLE status.	8 hours
E. One motor operated IRWST isolation valve not fully open. <u>OR</u> Power is not removed from one or more motor operated IRWST isolation valves.	E.1	Restore motor operated IRWST isolation valve to fully open condition with power removed from both valves.	1 hour
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> LCO not met for reasons other than Condition A, B, C, D, or E.	F.1 <u>AND</u> F.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.6.1	Verify the IRWST water temperature is < 120°F.	24 hours
SR 3.5.6.2	Verify the IRWST borated water volume is $\geq$ 73,100 cu. ft.	24 hours
SR 3.5.6.3	Verify the volume of noncondensible gases in each of the four IRWST injection squib valve outlet line pipe stubs has not caused the high-point water level to drop below the sensor.	24 hours
SR 3.5.6.4	Verify the IRWST boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days <u>AND</u> Once within 6 hours after each solution volume increase of ≥ 15,000 gal
SR 3.5.6.5	Verify each motor operated IRWST isolation valve is fully open.	12 hours
SR 3.5.6.6	Verify power is removed from each motor operated IRWST isolation valve.	31 days
SR 3.5.6.7	Verify each motor operated containment recirculation isolation valve is fully open.	31 days
SR 3.5.6.8	Verify each IRWST injection and containment recirculation squib valve is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.5.6.9	--- - NOTE - Squib actuation may be excluded. --- Verify continuity of the circuit from the Protection Logic Cabinets to each IRWST injection and containment recirculation squib valve on an actual or simulated actuation signal.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.6.10	Verify by visual inspection that the IRWST screens and the containment recirculation screens are not restricted by debris.	24 months
SR 3.5.6.11	Verify IRWST injection and recirculation system flow performance in accordance with the System Level OPERABILITY Testing Program.	10 years

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.7 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5

LCO 3.5.7 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. Required IRWST injection flow path with noncondensible gas volume in one squib valve outlet line pipe stub not within limit.	B.1 Restore noncondensible gas volume in squib valve outlet line pipe stub to within limit.	72 hours
C. Required IRWST injection flow path with noncondensible gas volume in both squib valve outlet line pipe stubs not within limit.	C.1 Restore noncondensible gas volume in one squib valve outlet line pipe stub to within limit.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. IRWST boron concentration not within limits. <u>OR</u> IRWST borated water temperature not within limits. <u>OR</u> IRWST borated water volume < 73,100 cu. ft. and $\geq$ 70,907 cu. ft.	D.1 Restore IRWST to OPERABLE status.	8 hours
E. Required motor operated IRWST isolation valve not fully open. <u>OR</u> Power is not removed from required motor operated IRWST isolation valve.	E.1 Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> LCO not met for reasons other than Condition A, B, C, D, or E.	F.1 Initiate action to establish ≥ 20% pressurizer level with the Reactor Coolant System (RCS) pressure boundary intact. <u>AND</u> F.2 Suspend positive reactivity additions.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.7.1	For the IRWST and flow paths required to be OPERABLE, the SRs of Specification 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating,” are applicable.	In accordance with applicable SRs

3.5 PASSIVE CORE COOLING SYSTEM (PXS)

3.5.8 In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6

LCO 3.5.8 The IRWST, with one injection flow path and one containment recirculation flow path, shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required motor operated containment recirculation isolation valve not fully open.	A.1 Open required motor operated containment recirculation isolation valve.	72 hours
B. Required IRWST injection flow path with noncondensible gas volume in one squib valve outlet line pipe stub not within limit.	B.1 Restore noncondensible gas volume in squib valve outlet line pipe stub to within limit.	72 hours
C. Required IRWST injection flow path with noncondensible gas volume in both squib valve outlet line pipe stubs not within limit.	C.1 Restore noncondensible gas volume in one squib valve outlet line pipe stub to within limit.	8 hours
D. IRWST and refueling cavity boron concentration not within limits. <u>OR</u> IRWST and refueling cavity borated water temperature not within limits. <u>OR</u> IRWST and refueling cavity borated water volume < 73,100 cu. ft. and $\geq$ 70,907 cu. ft.	D.1 Restore IRWST to OPERABLE status.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required motor operated IRWST isolation valve not fully open. <u>OR</u> Power is not removed from required motor operated IRWST isolation valve.	E.1	Restore required motor operated IRWST isolation valve to fully open condition with power removed.	1 hour
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> LCO not met for reasons other than Condition A, B, C, D, or E.	F.1 <u>AND</u> F.2	Initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange. Suspend positive reactivity additions.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.8.1	Verify the IRWST and refueling cavity water temperature is < 120°F.	24 hours
SR 3.5.8.2	Verify the IRWST and refueling cavity water total borated water volume is ≥ 73,100 cu. ft.	24 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.8.3	Verify the IRWST and refueling cavity boron concentration is ≥ 2600 ppm and ≤ 2900 ppm.	31 days <u>AND</u> Once within 6 hours after each solution volume increase of ≥ 15,000 gal
SR 3.5.8.4	For the IRWST and flow paths required to be OPERABLE, the following SRs are applicable: SR 3.5.6.3 SR 3.5.6.6 SR 3.5.6.8 SR 3.5.6.10 SR 3.5.6.5 SR 3.5.6.7 SR 3.5.6.9 SR 3.5.6.11	In accordance with applicable SRs

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage-rate testing except for containment air-lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

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

ACTIONS

- NOTES -

1. Entry and exit is permissible to perform repairs on the affected air lock components.

2. Separate Condition entry is allowed for each air lock.

3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	---------------------------------- **- NOTES -** 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ---------------------------------- A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	 1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Lock the OPERABLE door closed in the affected air lock. <u>AND</u> A.3 ------------------------------------ **- NOTE -** Air lock doors in high radiation areas may be verified locked closed by administrative means. ------------------------------------ Verify the OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	-- **- NOTES -** 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. -- B.1 Verify an OPERABLE door is closed in the affected air lock. <u>AND</u>	 1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	B.3	------------------------------------ **- NOTE -** Air lock doors in high radiation areas may be verified locked closed by administrative means. ------------------------------------	
		Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-- **- NOTES -** 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. -- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE, except for the containment isolation valves associated with closed systems.

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

ACTIONS

- NOTES -

1. Penetration flow path(s) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, “Containment,” when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one containment isolation valve inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours
	AND		

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	------------------------------------ **- NOTES -** 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ------------------------------------ Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
B. One or more penetration flow paths with two containment isolation valves inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 16 inch containment purge valve is closed, except when the 16 inch containment purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances which require the valves to be open.	31 days
SR 3.6.3.2	-- **- NOTE -** Valves and blind flanges in high radiation areas may be verified by use of administrative controls. -- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	-- **- NOTE -** Valves and blind flanges in high radiation areas may be verified by use of administrative controls. -- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.2 psig and ≤ +1.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter.

- NOTE -

The high pressure LCO limit is not applicable in MODES 5 or 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment pressure not within limits.	A.1	Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. Required Action and associated Completion Time of Condtion A not met in MODE 5 or 6.	C.1	Open a containment air flow path ≥ 6 inches in diameter.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4,
MODES 5 and 6 with both containment equipment hatches and both containment airlocks closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6.	C.1 Open containment equipment hatch or containment airlock.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Passive Containment Cooling System (PCS)

LCO 3.6.6 The passive containment cooling system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
MODES 5 and 6 with the reactor decay heat > 6.0 MWt.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One passive containment cooling water flow path inoperable.	A.1 Restore flow path to OPERABLE status.	7 days
B. Two passive containment cooling water flow paths inoperable.	B.1 Restore one flow path to OPERABLE status.	72 hours
C. One or more water storage tank parameters not within limits.	C.1 Restore water storage tank to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, 3, or 4. <u>OR</u> LCO not met for reasons other than Condition A, B, or C in MODE 1, 2, 3, or 4.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 84 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required Action and associated Completion time of Condition A, B, or C not met in MODE 5. <u>OR</u> LCO not met for reasons other than Condition A, B, or C in MODE 5.	E.1	Initiate action to establish pressurizer level ≥ 20% with the Reactor Coolant System (RCS) pressure boundary intact.	Immediately
	<u>AND</u>		
	E.2	Suspend positive reactivity additions.	Immediately
F. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 6. <u>OR</u> LCO not met for reasons other than Condition A, B, or C in MODE 6.	F.1	Initiate action to establish water level ≥ 23 ft above the top of the reactor vessel flange.	Immediately
	<u>AND</u>		
	F.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify the water storage tank temperature ≥ 40°F and ≤ 120°F.	24 hours
SR 3.6.6.2	Verify the water storage tank volume ≥ 756,700 gallons.	7 days
SR 3.6.6.3	Verify each passive containment cooling system manual, power operated, and automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.4	Verify each passive containment cooling system automatic valve in each flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.5	Verify the air flow path from the shield building annulus inlet to the exit is unobstructed and, that all air baffle sections are in place.	24 months
SR 3.6.6.6	Verify passive containment cooling system flow and water coverage performance in accordance with the System Level OPERABILITY Testing Program.	At first refueling <u>AND</u> 10 years

3.6 CONTAINMENT SYSTEMS

3.6.7 Containment Penetrations

LCO 3.6.7 The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, can be closed prior to steaming into the containment.

b. One door in each air lock closed or, if open, can be closed prior to steaming into the containment.

c. The containment spare penetrations, if open, can be closed prior to steaming into the containment.

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere, if open, can be closed by a manual or automatic isolation valve, blind flange, or equivalent prior to steaming into the containment.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Restore containment penetrations to required status.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR LCO not met for reasons other than Condition A.	B.1.1	If in MODE 5, initiate action to establish ≥ 20% pressurizer level with the Reactor Coolant System (RCS) pressure boundary intact.	Immediately
	OR		
	B.1.2	If in MODE 6, initiate action to establish water level ≥ 23 feet above the top of the reactor vessel flange.	Immediately
	AND		
	B.2	Suspend positive reactivity additions.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify each required containment penetration is in the required status.	7 days
SR 3.6.7.2	--- **- NOTE -** Only required to be met for an open equipment hatch. --- Verify the hardware, tools, equipment and power source necessary to close the equipment hatch are available.	Prior to hatch removal AND 7 days

3.6 CONTAINMENT SYSTEMS

3.6.8 pH Adjustment

LCO 3.6.8 The pH adjustment baskets shall contain ≥ 26,460 lbs of trisodium phosphate (TSP).

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The weight of TSP in the pH adjustment baskets not within limit.	A.1	Restore weight of TSP in the pH adjustment baskets to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify the pH adjustment baskets contain ≥ 26,460 lbs of TSP.	24 months
SR 3.6.8.2	Verify a sample from the pH adjustment baskets provides adequate pH adjustment of the post-accident water.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.9 Vacuum Relief Valves

LCO 3.6.9 Two vacuum relief flow paths shall be OPERABLE.

AND

Containment inside to outside differential air temperature shall be ≤ 90°F.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 without an open containment air flow path ≥ 6 inches in diameter.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vacuum relief flow path inoperable.	A.1 Restore vacuum relief flow path to OPERABLE status.	72 hours
B. Containment inside to outside differential air temperature > 90°F.	B.1 Restore containment inside to outside differential air temperature to within limit. OR B.2 Reduce containment average temperature ≤ 80°F.	8 hours 8 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Both vacuum relief flow paths inoperable in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6. <u>OR</u> Both vacuum relief flow paths inoperable in MODE 5 or 6.	D.1 Open a containment air flow path ≥ 6 inches in diameter.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Verify containment inside to outside differential air temperature is ≤ 90°F.	12 hours
SR 3.6.9.2	Verify each vacuum relief flow path is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program
SR 3.6.9.3	Verify each vacuum relief valve actuates to relieve vacuum on an actual or simulated signal.	24 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Six MSSVs per steam generator shall be OPERABLE.

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

- NOTE -

The MSSVs are not required to be OPERABLE for opening in MODE 4 when the Reactor Coolant System (RCS) is being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTE -

Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both steam generators with one or more MSSVs inoperable for opening.	A.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. <u>AND</u>	4 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ------------------------------------ **- NOTE -** Only required in MODE 1. ------------------------------------ Reduce the Power Range Neutron Flux – High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
B. One or both steam generators with one or more MSSVs inoperable for closing.	B.1 Restore MSSV to OERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or both steam generators with ≥ 5 MSSVs inoperable for opening.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4 with the RCS cooling provided by the RNS.	6 hours 24 hours
D. Required Action and associated Completion Time of Condition B not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-- **- NOTE -** Only required to be performed in MODES 1 and 2. -- Verify each MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
OPERABLE MSSVs versus Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
5	60
4	46
3	32
2	18

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 1%)
STEAM GENERATOR		
#1	#2	
V030A	V030B	1185
V031A	V031B	1197
V032A	V032B	1209
V033A	V033B	1221
V034A	V034B	1232
V035A	V035B	1232

3.7 PLANT SYSTEMS

3.7.2 Main Steam Line Flow Path Isolation Valves

LCO 3.7.2 Each of the following main steam line flow path isolation valves shall be OPERABLE:

a. Main steam isolation valves (MSIVs);

b. MSIV bypass valves;

c. Main steam line drain valves;

d. Turbine stop valves or turbine control valves;

e. Turbine bypass valves; and

f. Moisture separator reheater 2nd stage steam isolation valves.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1	Restore valve to OPERABLE status.	8 hours
B. One or more of the turbine stop valves and associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1.	B.1	Restore valve(s) to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two MSIVs inoperable in MODE 1. OR One MSIV inoperable and one or more of the turbine stop valves and associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 1. OR Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -------------------------------- **- NOTE -** Separate Condition entry is allowed for each main steam line flow path. -------------------------------- One or two MSIVs inoperable in MODE 2, 3, or 4. <u>OR</u> One or more of the turbine stop valves and associated turbine control valves, turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 2, 3, or 4.	D.1 Isolate affected main steam line flow path. <u>AND</u> D.2 Verify affected main steam line flow path is isolated.	8 hours Once per 7 days
E. -------------------------------- **- NOTE -** Separate Condition entry is allowed for each penetration flow path. -------------------------------- One or more MSIV bypass or main steam line drain valves inoperable.	--- **- NOTE –** Penetration flow path(s) may be unisolated intermittently under administrative controls. --- E.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. <u>AND</u>	72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.2	------------------------------------ **- NOTES -** 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ----------------------------------- Verify the affected penetration flow path is isolated.	 Once per 31 days
F. Required Action and associated Completion Time of Condition D or E not met.	F.1	Be in MODE 3.	6 hours
	AND		
	F.2	Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours
	AND		
	F.3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	-- **- NOTE -** Only required to be performed prior to entry into MODE 2. -- Verify MSIV closure time is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.2.2	-- **- NOTE -** Only required to be performed prior to entry into MODE 2. -- Verify required turbine stop, turbine control, turbine bypass, and moisture separator reheater 2nd stage steam isolation valves' closure time is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.2.3	Verify the isolation time of each MSIV bypass valve and main steam line drain isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.7.2.4	Verify each MSIV bypass valve and main steam line drain isolation valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Control Valves (MFCVs)

LCO 3.7.3 The MFIV and the MFCV for each Steam Generator shall be OPERABLE.

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

ACTIONS

- NOTE -

Separate Condition entry is allowed for each feedwater flow path.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both feedwater flow paths with MFIV or MFCV inoperable.	A.1	Isolate the affected flow path.	72 hours
	<u>AND</u>		
	A.2	Verify affected flow path is isolated.	Once per 7 days
B. One or both feedwater flow paths with associated MFIV and MFCV inoperable.	B.1	Isolate affected flow path.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours
	<u>AND</u>		
	C.3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	--- **- NOTE -** Only required to be performed prior to entry into MODE 2. --- Verify the closure time of each MFIV and MFCV is within limits on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.7 PLANT SYSTEMS

3.7.5 Spent Fuel Pool Water Level

LCO 3.7.5 The spent fuel pool water level shall be ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: When irradiated fuel assemblies are stored in the spent fuel pool.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Spent fuel pool water level < 23 ft.	A.1	Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>		
	A.2	Initiate action to restore water level to ≥ 23 ft.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify the spent fuel pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.6 Main Control Room Emergency Habitability System (VES)

LCO 3.7.6 The VES shall be OPERABLE.

- NOTE -

The main control room envelope (MCRE) boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One valve or damper inoperable.	A.1 Restore valve or damper to OPERABLE status.	7 days
B. MCRE air temperature not within limit.	B.1 Restore MCRE air temperature to within limit.	24 hours
C. VES inoperable due to inoperable MCRE boundary in MODE 1, 2, 3, or 4.	C.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	C.2 Verify mitigating actions ensure MCRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	<u>AND</u>	
	C.3 Restore MCRE boundary to OPERABLE status.	90 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One bank of VES air tanks inoperable.	D.1	Verify that the OPERABLE tanks contain > 245,680 scf of compressed air.	2 hours AND Once per 12 hours thereafter
	AND		
	D.2	Verify VBS MCRE ancillary fans and supporting equipment are available.	24 hours
	AND		
	D.3	Restore VES to OPERABLE status.	7 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met in MODE 1, 2, 3, or 4. OR VES inoperable for reasons other than Condition A, B, C, or D in MODE 1, 2, 3, or 4.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 5.	36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, C, or D not met during movement of irradiated fuel. <u>OR</u> VES inoperable for reasons other than Condition A, B, C, or D during movement of irradiated fuel. <u>OR</u> VES inoperable due to inoperable MCRE boundary during movement of irradiated fuel.	F.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify MCRE air temperature is ≤ 75°F.	24 hours
SR 3.7.6.2	Verify the compressed air storage tanks contain > 327,574 scf of compressed air.	24 hours
SR 3.7.6.3	Verify each VES air delivery isolation valve is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.4	Operate VES for ≥ 15 minutes.	31 days
SR 3.7.6.5	Verify each VES air header manual isolation valve is in an open position.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.6	Verify the air quality of the air storage tanks is within limits.	92 days
SR 3.7.6.7	Verify all MCRE isolation valves are OPERABLE and will close upon receipt of an actual or simulated actuation signal.	24 months
SR 3.7.6.8	Verify each VES pressure relief isolation valve within the MCRE pressure boundary is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.9	Verify each VES pressure relief damper is OPERABLE.	24 months
SR 3.7.6.10	Verify the self-contained pressure regulating valve in each VES air delivery flow path is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.11	Perform required MCRE unfiltered air inleakage testing in accordance with the Main Control Room Envelope Habitability Program.	In accordance with the Main Control Room Envelope Habitability Program
SR 3.7.6.12	Perform required VES Passive Filtration system filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

3.7 PLANT SYSTEMS

3.7.7 Startup Feedwater Isolation and Control Valves

LCO 3.7.7 Each Startup Feedwater Isolation Valve and Control Valve shall be OPERABLE.

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

ACTIONS

- NOTES -

1. Flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each flow path.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more flow paths with one inoperable valve.	A.1	Isolate the affected flow path.	72 hours
	AND		
	A.2	Verify affected flow path is isolated.	Once per 7 days
B. One flow path with two inoperable valves.	B.1	Isolate the affected flow path.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4 with the Reactor Coolant System (RCS) cooling provided by the Normal Residual Heat Removal System (RNS).	24 hours
	AND		
	C.3	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify each startup feedwater isolation and control valve is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.7.2	Verify each startup feedwater isolation and control valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.8 Main Steam Line Leakage

LCO 3.7.8 Main Steam Line leakage through the pipe walls inside containment shall be ≤ 0.5 gpm.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Main Steam Line leakage > 0.5 gpm.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify main steam line leakage into the containment sump ≤ 0.5 gpm.	Per SR 3.4.7.1

3.7 PLANT SYSTEMS

3.7.9 Spent Fuel Pool Makeup Water Sources

LCO 3.7.9 Spent fuel pool makeup water sources shall be OPERABLE.

- NOTES -

1. OPERABILITY of the cask washdown pit is required when the spent fuel pool decay heat > 4.7 MWt and ≤ 7.2 MWt.

2. OPERABILITY of the cask loading pit is required when the spent fuel pool decay heat > 5.6 MWt and ≤ 7.2 MWt.

3. OPERABILITY of the Passive Containment Cooling Water Storage Tank (PCCWST) is required as a spent fuel pool makeup water source when the spent fuel pool decay heat > 7.2 MWt. If the reactor decay heat is > 6.0 MWt, the PCCWST must be exclusively available for containment cooling in accordance with LCO 3.6.6.

APPLICABILITY: During storage of fuel in the spent fuel pool with a decay heat > 4.7 MWt.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required spent fuel pool makeup water sources inoperable.	A.1	Initiate action to restore the required makeup water source(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	-- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 7.2 MWt. -- Verify one passive containment cooling system, motor-operated valve in each flow path is closed and locked, sealed, or otherwise secured in position.	7 days
SR 3.7.9.2	-- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 7.2 MWt. -- Verify the PCCWST volume is ≥ 756,700 gallons.	7 days
SR 3.7.9.3	-- **- NOTE -** Only required to be performed when spent fuel pool decay heat is ≤ 7.2 MWt. -- Verify the water level in the cask washdown pit is ≥ 13.75 ft.	31 days
SR 3.7.9.4	-- **- NOTE -** Only required to be performed when spent fuel pool decay heat is > 5.6 MWt and ≤ 7.2 MWt. -- Verify the water level in the cask loading pit is ≥ 43.9 ft. and in communication with the spent fuel pool.	31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.9.5	Verify the spent fuel pool makeup isolation valves PCS-PL-V009, PCS-PL-V045, PCS-PL-V051, SFS-PL-V042, SFS-PL-V045, SFS-PL-V049, SFS-PL-V066, and SFS-PL-V068 are OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.10 Steam Generator (SG) Isolation Valves

LCO 3.7.10 Each SG power operated relief valve (PORV), PORV block valve, and SG blowdown isolation valve shall be OPERABLE.

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

- NOTE -

PORV OPERABILITY is not required in MODE 4 with Reactor Coolant System (RCS) being cooled by the Normal Residual Heat Removal System (RNS).

ACTIONS

- NOTES -

1. SG blowdown flow path(s) may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG PORV flow paths with one isolation valve inoperable.	A.1 Isolate the flow path by use of at least one closed and deactivated automatic valve. <u>AND</u>	72 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	-- **- NOTES –** 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. -- A.2 Verify the affected flow path is isolated.	Once per 31 days
B. One or more SG blowdown flow paths with one isolation valve inoperable.	B.1 Isolate the flow path by one closed valve. <u>AND</u> B.2 Verify the affected flow path is isolated.	72 hours Once per 7 days
C. One or more SG PORV flow paths with two isolation valves inoperable.	C.1 Isolate the affected flow path by use of at least one closed and deactivated automatic valve.	8 hours
D. One or more SG blowdown flow paths with two isolation valves inoperable.	D.1 Isolate the flow path by one closed valve.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours
	AND	
	E.2 Be in MODE 4 with the RCS cooling provided by the RNS.	24 hours
	AND	
	-- **- NOTE –** Not applicable for inoperable PORV(s). --	
	E.3 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify each SG PORV, PORV block valve, and SG blowdown isolation valve strokes closed.	In accordance with the Inservice Testing Program
SR 3.7.10.2	Verify the isolation time of each PORV block valve and SG blowdown isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.7.10.3	Verify each SG PORV, PORV block valve, and SG blowdown isolation valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool Boron Concentration

LCO 3.7.11 The spent fuel pool boron concentration shall be ≥ 2300 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool storage verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1	Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>		
	A.2.1	Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	<u>OR</u>		
	A.2.2	Initiate action to perform a spent fuel pool storage verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool Storage

LCO 3.7.12 The combination of initial enrichment and burnup of each fuel assembly stored in Region 2 shall be within the limits specified in Figure 3.7.12-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.12-1.	Prior to storing the fuel assembly in Region 2

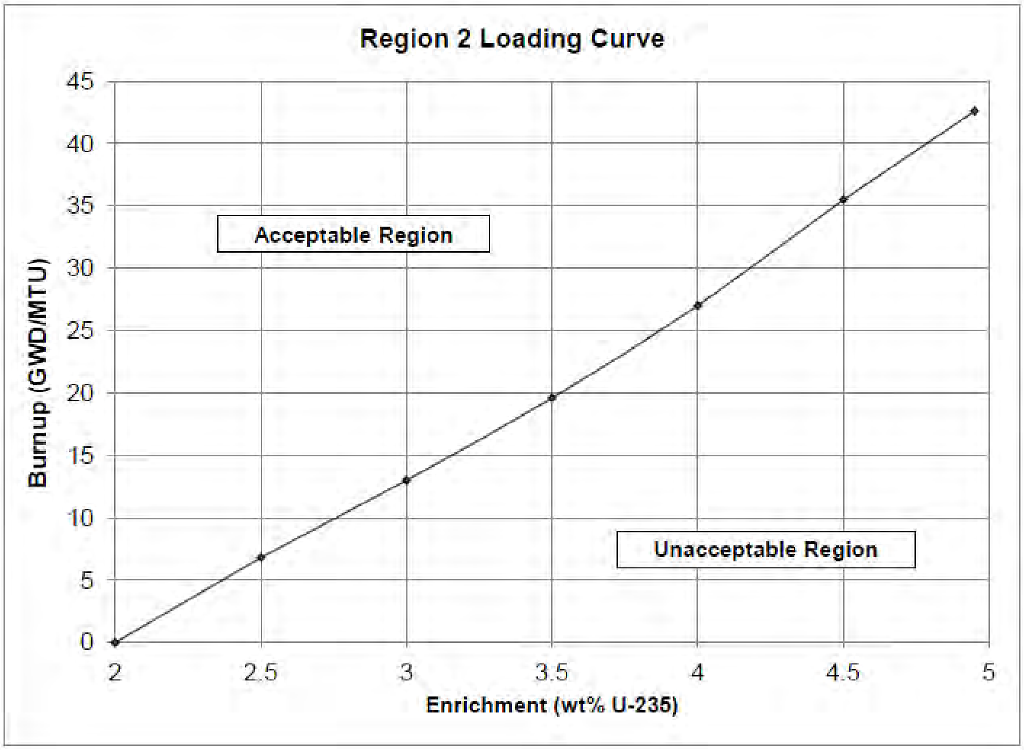


Figure 3.7.12-1

Minimum Fuel Assembly Burnup Versus Initial Enrichment for Region 2 Spent Fuel Cells

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 DC Sources – Operating

LCO 3.8.1 The Division A, B, C, and D Class 1E DC power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more battery chargers in one division inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	6 hours
	<u>AND</u>	
	A.2 Verify battery float current ≤ 2 amps.	Once per 24 hours
	<u>AND</u>	
	A.3 Restore battery charger(s) to OPERABLE status.	7 days
B. One or more battery chargers in two divisions inoperable.	B.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	<u>AND</u>	
	B.2 Verify battery float current ≤ 2 amps.	Once per 24 hours
	<u>AND</u>	
	B.3 Restore battery charger(s) to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more batteries in one division inoperable.	C.1	Restore batteries to OPERABLE status.	6 hours
D. One or more batteries in two divisions inoperable.	D.1	Restore batteries to OPERABLE status.	2 hours
E. One DC electrical power subsystem inoperable for reasons other than Condition A or C.	E.1	Restore DC electrical power subsystem to OPERABLE status.	6 hours
F. Two DC electrical power subsystems inoperable for reasons other than B or D.	F.1	Restore DC electrical power subsystem to OPERABLE status.	2 hours
G. Required Action and associated Completion Time not met.	G.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	G.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	7 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	Verify each battery charger supplies ≥ 200 amps at greater than or equal to the minimum established float voltage for ≥ 8 hours. OR Verify each battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	24 months
SR 3.8.1.3	--- **- NOTE -** The modified performance discharge test in SR 3.8.7.6 may be performed in lieu of SR 3.8.1.3. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 DC Sources – Shutdown

LCO 3.8.2 DC electrical power subsystems shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.6, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required battery chargers in one division inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	6 hours
	<u>AND</u>	
	A.2 Verify battery float current ≤ 2 amps	Once per 24 hours
	<u>AND</u>	
	A.3 Restore battery charger(s) to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more required DC electrical power subsystems inoperable.	B.1	Declare affected required features inoperable.	Immediately
	OR		
	B.2.1	Suspend movement of irradiated fuel assemblies.	Immediately
	AND		
	B.2.2	Suspend operations with a potential for draining the reactor vessel.	Immediately
	AND		
	B.2.3	Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND		
	B.2.4	Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-- **- NOTE -** The following SRs are not required to be performed: SR 3.8.1.2 and SR 3.8.1.3. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.1.1 SR 3.8.1.2 SR 3.8.1.3	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Inverters – Operating

LCO 3.8.3 The Division A, B, C, and D inverters shall be OPERABLE.

- NOTES -

One inverter may be disconnected from its associated DC bus for ≤ 72 hours to perform an equalizing charge on its associated battery, providing:

1. The associated instrument and control bus is energized from its Class 1E constant voltage source transformer; and

2. All other AC instrument and control buses are energized from their associated OPERABLE inverters.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One inverter inoperable.	A.1	------------------------------------ **- NOTE -** Enter applicable Conditions and Required Actions of LCO 3.8.5 “Distribution Systems – Operating” with any instrument and control bus de-energized. ------------------------------------ Restore inverter to OPERABLE status.	 24 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify correct inverter voltage, frequency, and alignment to required AC instrument and control buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 Inverters – Shutdown

LCO 3.8.4 Inverters shall be OPERABLE to support the onsite Class 1E power distribution subsystems required by LCO 3.8.6, “Distribution Systems – Shutdown.”

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required features inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify correct inverter voltage, frequency, and alignments to required AC instrument and control buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 Distribution Systems – Operating

LCO 3.8.5 The following Division A, B, C, and D electrical power distribution subsystems shall be OPERABLE:

a. DC; and

b. AC instrument and control.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC instrument and control division inoperable.	A.1 Restore AC instrument and control division to OPERABLE status.	6 hours
B. One DC division inoperable.	B.1 Restore DC division to OPERABLE status.	6 hours
C. Two AC instrument and control divisions inoperable.	C.1 Restore one AC instrument and control division to OPERABLE status.	2 hours
D. Two DC divisions inoperable.	D.1 Restore one DC division to OPERABLE status.	2 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.	6 hours 36 hours
F. Two inoperable divisions that result in a loss of safety function.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	Verify correct breaker and switch alignments and voltage to required DC and AC instrument and control electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Distribution Systems – Shutdown

LCO 3.8.6 The necessary portions of DC and AC instrument and control electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

- NOTE -

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC or AC instrument and control electrical power distribution subsystems inoperable.	A.1 Declare affected required features inoperable.	Immediately
	OR	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2.2 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.4	Initiate actions to restore required DC and AC instrument and control electrical power distribution subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify correct breaker and switch alignments and voltage to required DC and AC instrument and control electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Battery Parameters

LCO 3.8.7 Battery Parameters for Division A, B, C, and D batteries shall be within limits.

APPLICABILITY: When associated DC electrical power sources are required to be OPERABLE.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries in one division with one or more battery cells float voltage < 2.07 V.	A.1 Perform SR 3.8.1.1.	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.8.7.1.	2 hours
	<u>AND</u>	
	A.3 Restore affected cell voltage ≥ 2.07 V.	24 hours
B. One or more batteries in one division with float current > 2 amps.	B.1 Perform SR 3.8.1.1.	2 hours
	<u>AND</u>	
	B.2 Restore battery float current to ≤ 2 amps.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
---------- **- NOTE -** Required Action C.2 shall be completed if electrolyte level was below the top of plates. ----------	---------- **- NOTE -** Required Actions C.1 and C.2 are only applicable if electrolyte level was below the top of plates. ----------	
C. One or more batteries in one division with one or more cells electrolyte level less than minimum established design limits.	C.1 Restore electrolyte level to above top of plates.	8 hours
	<u>AND</u>	
	C.2 Verify no evidence of leakage.	12 hours
	<u>AND</u>	
	C.3 Restore electrolyte level to greater than or equal to minimum established design limits.	31 days
D. One or more batteries in one division with pilot cell electrolyte temperature less than minimum established design limits.	D.1 Restore battery pilot cell temperature to greater than or equal to minimum established design limits.	12 hours
E. One or more batteries in two or more divisions with battery parameters not within limits.	E.1 Restore battery parameters for batteries in three divisions to within limits.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met. <u>OR</u> One or more batteries in one division with one or more battery cells float voltage < 2.07 V and float current > 2 amps.	F.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	-- **- NOTE -** Not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1. -- Verify each battery float current is ≤ 2 amps.	7 days
SR 3.8.7.2	Verify each battery pilot cell float voltage is ≥ 2.07 V.	31 days
SR 3.8.7.3	Verify each battery connected cell electrolyte level is greater than or equal to minimum established design limits.	31 days
SR 3.8.7.4	Verify each battery pilot cell temperature is greater than or equal to minimum established design limits.	31 days
SR 3.8.7.5	Verify each battery connected cell float voltage is ≥ 2.07 V.	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.7.6	Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentration of the Reactor Coolant System (RCS), the fuel transfer canal, and the refueling cavity shall be maintained within the limit specified in COLR.

APPLICABILITY: MODE 6.

- NOTE -

Applicable to the fuel transfer canal and the refueling cavity only when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2 Initiate actions to restore boron concentration to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Flow Paths

LCO 3.9.2 One valve in each unborated water source flow path shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

- NOTE -

Separate condition entry is allowed for each unborated water source flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------------------------- **- NOTE -** Required Action A.2 must be completed whenever Condition A is entered. -------------------------------- One or more unborated water source flow paths with no valve secured in the closed position.	A.1 Initiate actions to secure one valve in the flow path in the closed position. AND A.2 Perform SR 3.9.1.1.	Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify one valve in each unborated water source flow path is secured in the closed position.	31 days

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2 Suspend operations that would cause introduction into the Reactor Coolant System (RCS), coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	<u>AND</u>	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform a CHANNEL CHECK.	12 hours
SR 3.9.3.2	--- **- NOTE-** Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	24 months

3.9 REFUELING OPERATIONS

3.9.4 Refueling Cavity Water Level

LCO 3.9.4 Refueling Cavity Water Level shall be maintained ≥ 23 ft above the top of the reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

3.9 REFUELING OPERATIONS

3.9.5 Decay Time

LCO 3.9.5 The reactor shall be subcritical for ≥ 48 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical < 48 hours.	A.1 Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify the reactor has been subcritical for ≥ 48 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

4.0 DESIGN FEATURES

4.1 Site

VCSNS is located in Fairfield County, South Carolina, on the eastern side of the Broad River at the Monticello Reservoir and it is approximately 15 miles west of the county seat of Winnsboro and 26 miles northwest of Columbia, the state capital.

4.1.1 Site and Exclusion Boundaries

The Site Boundary is shown in Figure 4.1-2.

The Exclusion Area Boundary is shown in Figure 4.1-2.

4.1.2 Low Population Zone (LPZ)

The LPZ is defined by the 3 mile radius from VCSNS Unit 1 as shown in Figure 4.1-1.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of fuel rods clad with a zirconium based alloy and containing an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium based alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod and Gray Rod Assemblies

The reactor core shall contain 53 Rod Cluster Control Assemblies (RCCAs), each with 24 rodlets/RCCA. The RCCA absorber material shall be silver indium cadmium as approved by the NRC.

Additionally, there are 16 low worth Gray Rod Cluster Assemblies (GRCAs), with 24 rodlets/GRCA, which, in conjunction with the RCCAs, are used to augment mechanical shim (MSHIM) load follow operation.

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 <u>Criticality</u>

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.95 weight percent;

b. $k_{eff} \leq 0.95$ if flooded with unborated water which includes an allowance for uncertainties (Region 1 racks);

c. A nominal 10.93 inch center-to-center distance between fuel assemblies placed in Region 1 of the spent fuel storage racks;

d. A nominal 9.04 inch center-to-center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks;

e. A nominal 11.65 inch center-to-center distance between fuel assemblies placed in the Defective Fuel Cells;

f. New or partially spent fuel assemblies with any discharge burnup may be allowed unrestricted storage in Region 1 and the Defective Fuel Cells of Figure 4.3-1;

g. Partially spent fuel assemblies meeting the initial enrichment and burnup requirements of LCO 3.7.12, "Spent Fuel Pool Storage," may be stored in Region 2 of Figure 4.3-1; and

h. $k_{eff} < 1.0$ if flooded with unborated water and $k_{eff} \leq 0.95$ if flooded with borated water at a minimum soluble boron concentration described in the Bases for LCO 3.7.12 for normal and design basis criticality-related accident conditions, which includes an allowance for uncertainties (Region 2 racks).

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. The maximum k_{eff} value, including all biases and uncertainties, shall be less than or equal to 0.95 with full density unborated water;

4.0 DESIGN FEATURES

c. The maximum k_{eff} value, including all biases and uncertainties, shall be less than or equal to 0.98 with optimum moderation and full reflection conditions; and

d. A nominal 10.90 inch center-to-center distance between fuel assemblies placed in the new fuel storage racks.

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below a minimum water depth of ≥ 23 ft above the surface of the fuel storage racks.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 889 fuel assemblies.

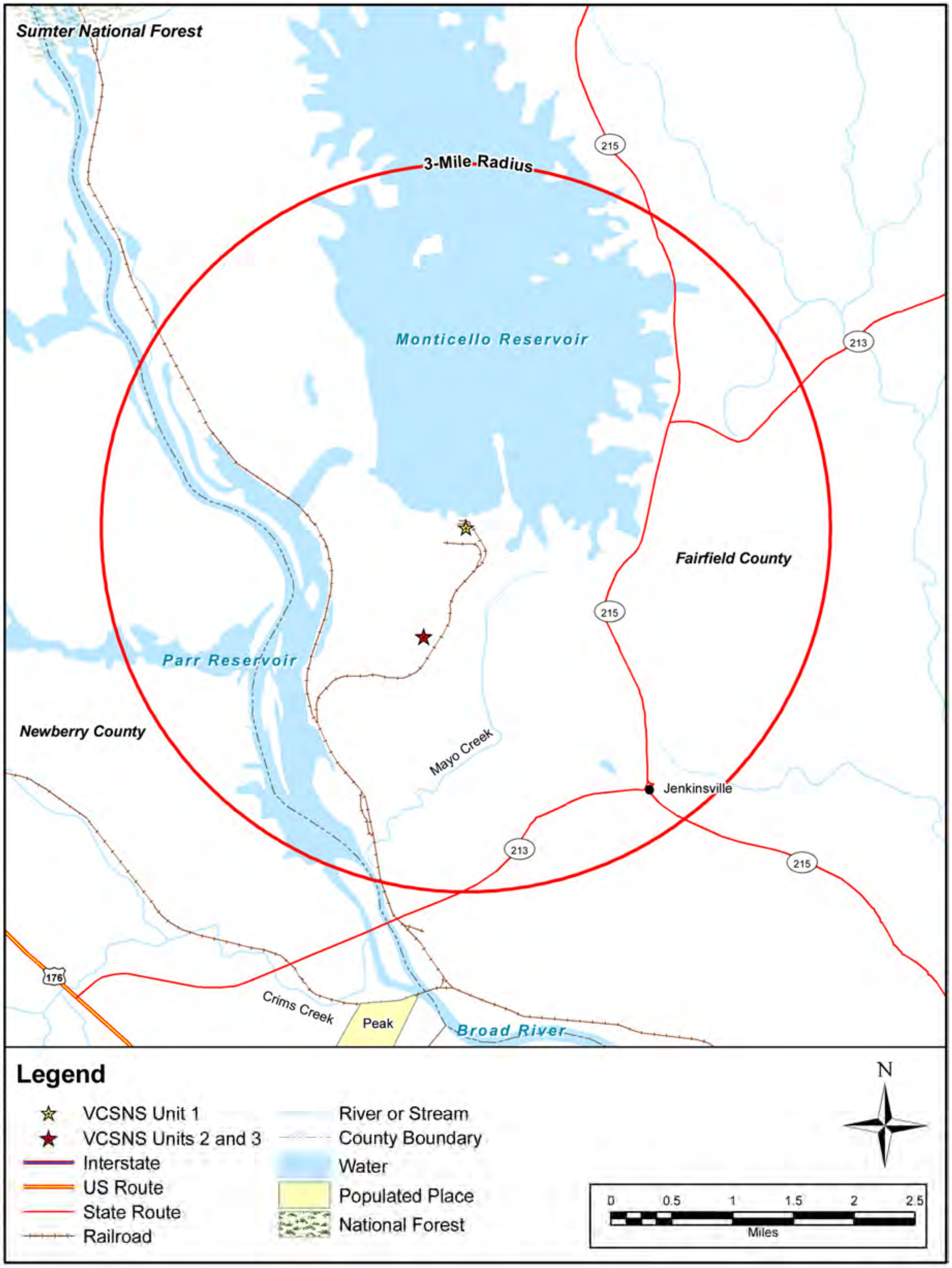


Figure 4.1-1 Low Population Zone Map

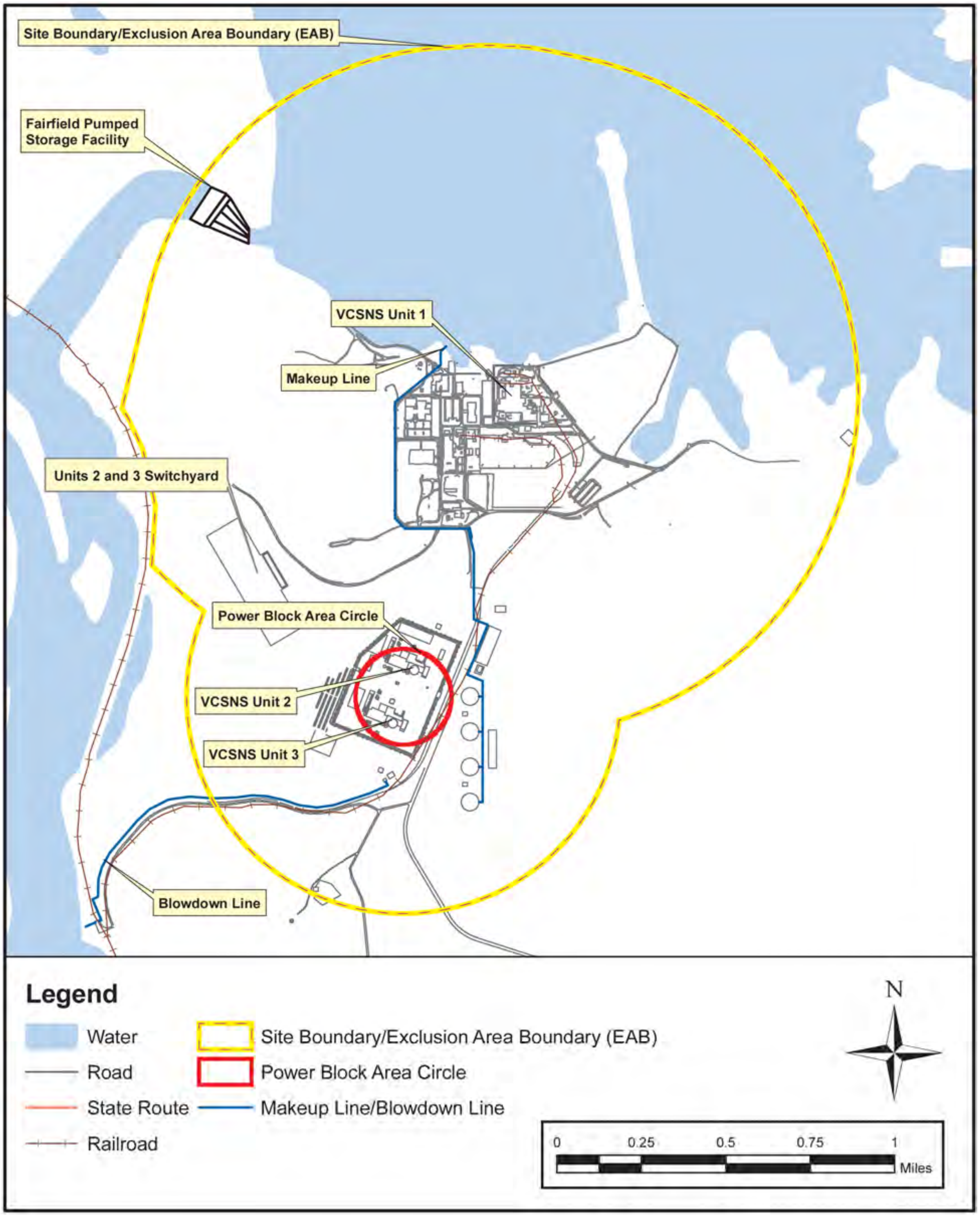
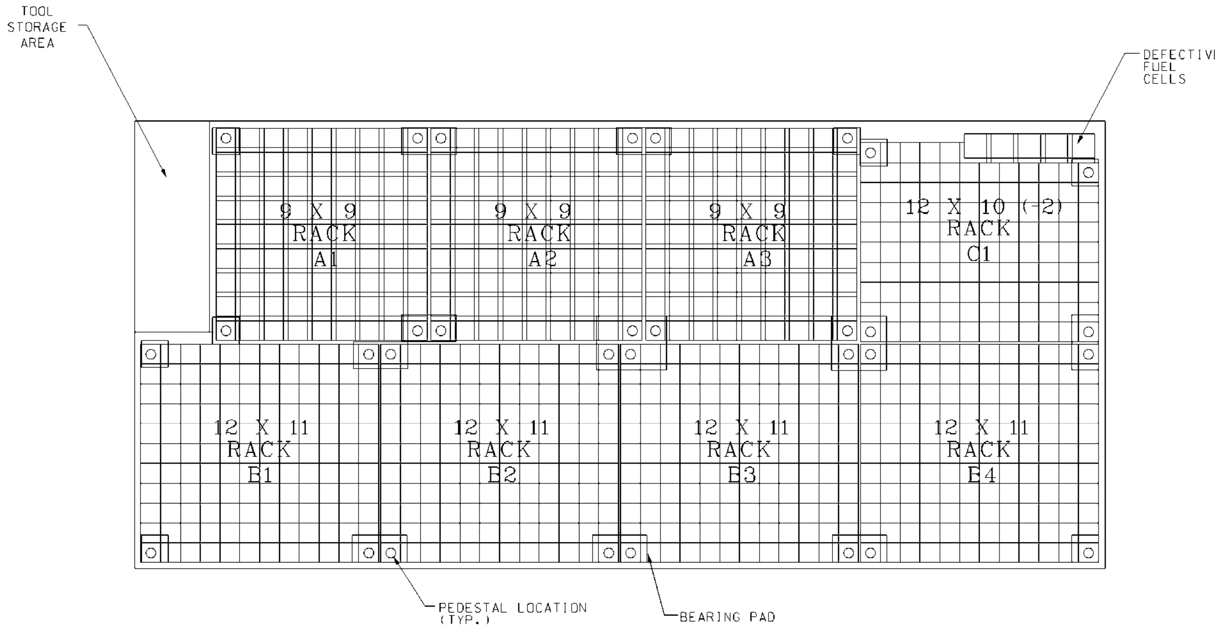


Figure 4.1-2 Exclusion Area Boundary and Site Boundary Map



Region 1 (A1, A2, A3) – 243 locations

Region 2 (B1, B2, B3, B4, C1) – 641 locations

Defective Fuel Cells (DFCs) – 5 locations

Total Storage Locations – 889

Figure 4.3-1

Discrete Two Region Spent Fuel Pool Rack Layout

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Plant Manager shall be responsible for overall unit operations and shall delegate in writing the succession to this responsibility during his absence.

The Plant Manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The Shift Manager shall be responsible for the control room command function. During any absence of the shift manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the shift manager from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the FSAR;

b. The Plant Manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operation pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODE 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

5.2 Organization

5.2.2 Unit Staff (continued)

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. The operations manager or assistant operations manager shall hold an SRO license.

e An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3, 2000, with the following exception:

a. During cold license operator training through the first refueling outage, the Regulatory Position C.1.b of Regulatory Guide 1.8, Revision 2, 1987, applies: cold license operator candidates meet the training elements defined in ANSI/ANS 3.1-1993 but are exempt from the experience requirements defined in ANSI/ANS 3.1-1993.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.1 and Specification 5.6.2.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   i. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   ii. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20. 1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the changed portion of the ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

5.5.2 Radioactive Effluent Control Program

a. This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoints determination in accordance with the methodology in the ODCM;

2. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20;

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public for radioactive materials in liquid effluents released form each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

6. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be in accordance with the following:

   i. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

5.5 Programs and Manuals

5.5.2 Radioactive Effluent Control Program (continued)

ii. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

b. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.3 Inservice Testing Program

This program provides control for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda Terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

5.5 Programs and Manuals

5.5.3 Inservice Testing Program (continued)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities;

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.4 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the

5.5.4 Steam Generator (SG) Program (continued)

assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.7, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following installation.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.5 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.6 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

5.5 Programs and Manuals

5.5.6 Technical Specifications (TS) Bases Control Program (continued)

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.7 Safety Function Determination Program (SFDP)

a. This program ensures loss of safety function is detected and appropriate action taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirement of LCO 3.0.6. The SFDP shall contain the following:

1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system’s Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

5.5.7 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to the support system(s) for the supported systems b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.8 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program, dated September 1995,” as modified by approved exceptions.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 58.3 psig. The containment design pressure is 59 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.10% of primary containment air weight per day.

5.5 Programs and Manuals

5.5.8 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;

2. Air lock testing acceptance criteria are:

i. Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a,

ii. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.9 System Level OPERABILITY Testing Program

The System Level OPERABILITY Testing Program provides requirements for performance tests of passive systems. The System Level Inservice Tests specified in FSAR Section 3.9.6 and FSAR Table 3.9-17 apply when specified by individual Surveillance Requirements.

a. The provisions of SR 3.0.2 are applicable to the test frequencies specified in FSAR Table 3.9-17 for performing system level OPERABILITY testing activities; and

b. The provisions of SR 3.0.3 are applicable to system level OPERABILITY testing activities.

5.5.10 Component Cyclic or Transient Limit

This program provides controls to track the FSAR Table 3.9-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5 Programs and Manuals

5.5.11 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

5.5.12 Main Control Room Envelope Habitability Program

A Main Control Room Envelope (MCRE) Habitability Program shall be established and implemented to ensure that MCRE habitability is maintained such that, with an OPERABLE Main Control Room Emergency Habitability System (VES), MCRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the MCRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the MCRE and the MCRE boundary.

b. Requirements for maintaining the MCRE boundary in its design condition, including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the MCRE boundary into the MCRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” Revision 0, May 2003, and (ii) assessing MCRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the MCRE pressure relative to all external areas adjacent to the MCRE boundary during the pressurization mode of operation of one VES air delivery flow path, operating at the required flow rate of 65 ± 5 scfm, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the MCRE boundary.

5.5 Programs and Manuals

5.5.12 Main Control Room Envelope Habitability Program (continued)

e. The quantitative limits on unfiltered air inleakage into the MCRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of MCRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing MCRE habitability, determining MCRE unfiltered inleakage, and measuring MCRE pressure and assessing the MCRE boundary as required by paragraphs c and d, respectively.

5.5.13 Ventilation Filter Testing Program (VFTP)

a. A program shall be established to implement the following required testing of the VES.

Tests described in Specification 5.5.13.a.1 and 5.5.13.a.2 shall be performed: i) initially, ii) once each 24 months, iii) after partial or complete replacement of a HEPA filter or charcoal adsorber, iv) following detection of, or evidence of, penetration or intrusion of water or other material into any portion of the VES that may have an adverse effect on the functional capability of the filters, and v) following painting, fire, or chemical release in any ventilation zone communicating with the VES that may have an adverse effect on the functional capability of the system.

Tests described in Specification 5.5.13.a.3 shall be performed: i) after each 720 hours of system operation or at least once each 24 months, whichever comes first, ii) following painting, fire, or chemical release in any ventilation zone communicating with the VES that may have an adverse effect on the functional capability of the carbon media, and iii) following detection of, or evidence of, penetration or intrusion of water or other material into any portion of the VES that may have an adverse effect on the functional capability of the carbon media.

Tests described in 5.5.13.a.4 shall be performed once per 24 months.

5.5 Programs and Manuals

5.5.13 Ventilation Filter Testing Program (continued)

1. Demonstrate for the VES that an inplace test of the high efficiency particulate air (HEPA) filter shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at a flow rate at least 600 cfm greater than the VES makeup flow rate.

Ventilation System	Flow Rate
VES	≥ 600 + VES makeup flow rate (cfm)

2. Demonstrate for the VES that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 3, and ASME N510-1989 at a flow rate at least 600 cfm greater than the VES makeup flow rate.

Ventilation System	Flow Rate
VES	≥ 600 + VES makeup flow rate (cfm)

3. Demonstrate for the VES that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 3, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below.

Ventilation System	Penetration	RH
VES	5%	95%

4. Demonstrate for the VES that the pressure drop across the combined HEPA filter, the charcoal adsorber, and the post filter is less than the value specified below when tested at the system flow rate specified below +/- 10%.

ESF Ventilation System	Delta P	Flow Rate
VES	5 in. water gauge	660 cfm

b. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5 Programs and Manuals

5.5.14 Setpoint Program (SP)

a. The Setpoint Program (SP) implements the regulatory requirement of 10 CFR 50.36(c)(1)(ii)(A) that technical specifications will include items in the category of limiting safety system settings (LSSS), which are settings for automatic protective devices related to those variables having significant safety functions.

b. The Nominal Trip Setpoint (NTS), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) for each Technical Specification required automatic protection instrumentation function shall be calculated in conformance with WCAP-16361-P, “Westinghouse Setpoint Methodology for Protection Systems – AP1000,” February 2011.

c. For each Technical Specification required automatic protection instrumentation function, performance of a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST (COT) surveillance “in accordance with the Setpoint Program” shall include the following:

1. The as-found value of the instrument channel trip setting shall be compared with the previously recorded as-left value.

   i. If the as-found value of the instrument channel trip setting differs from the previously recorded as-left value by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated to verify that it is functioning in accordance with its design basis before declaring the surveillance requirement met and returning the instrument channel to service. An Instrument Channel is determined to be functioning in accordance with its design basis if it can be set to within the ALT. This as-found condition shall be entered into the plant’s corrective action program.

   ii. If the as-found value of the instrument channel trip setting is less conservative than the specified AFT, the surveillance requirement is not met and the instrument channel shall be immediately declared inoperable.

5.5 Programs and Manuals

5.5.14 Setpoint Program (SP) (continued)

2. The instrument channel trip setting shall be set to a value within the specified ALT around the specified NTS at the completion of the surveillance; otherwise, the surveillance requirement is not met and the instrument channel shall be immediately declared inoperable.

d. The difference between the instrument channel trip setting as-found value and the previously recorded as-left value for each Technical Specification required automatic protection instrumentation function shall be trended and evaluated to verify that the instrument channel is functioning in accordance with its design basis.

e. The SP shall establish a document containing the current value of the specified NTS, AFT, and ALT for each Technical Specification required automatic protection instrumentation function and references to the calculation documentation. Changes to this document shall be governed by the regulatory requirement of 10 CFR 50.59. In addition, changes to the specified NTS, AFT, and ALT values shall be governed by the approved setpoint methodology. This document, including any revisions or supplements, shall be provided upon issuance to the NRC.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Annual Radiological Environmental Operating Report

- NOTE -

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 Radioactive Effluent Release Report

- NOTE -

A single submittal may be made for a multiple unit station.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1, "Reactor Core SLs";
3.1.1, "SHUTDOWN MARGIN (SDM)";
3.1.3, "Moderator Temperature Coefficient (MTC)";
3.1.5, "Shutdown Bank Insertion Limits";
3.1.6, "Control Bank Insertion Limits";
3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (Constant Axial Offset Control (CAOC) W(Z) Methodology)";
3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)";
3.2.5, "On-Line Power Distribution Monitoring System (OPDMS)-Monitored Parameters";
3.3.1, "Reactor Trip System (RTS) Instrumentation";
3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
3.9.1, "Boron Concentration."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Westinghouse Proprietary) and WCAP-9273-NP-A (Non-Proprietary).

   (Methodology for Specifications 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limits, 3.1.6 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - AXIAL FLUX DIFFERENCE, and 3.9.1 - Boron Concentration.)

2a. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974 (Westinghouse Proprietary) and WCAP-8403 (Non-Proprietary).

   (Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

5.6 Reporting Requirements

5.6.3 <u>CORE OPERATING LIMITS REPORT (COLR)</u> (continued)

2b. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.3 - AXIAL FLUX DIFFERENCE (Constant Axial Offset Control).)

3. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification,” February 1994 (Westinghouse Proprietary) and WCAP-10217-A (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology).)

4. WCAP-12945-P-A, Volumes 1-5, “Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis,” Revision 2, March 1998 (Westinghouse Proprietary) and WCAP-14747 (Non-Proprietary).

(Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor.)

5. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994, Addendum 1, May 1996 (Westinghouse Proprietary), and Addendum 2, March 2001 (Westinghouse Proprietary) and WCAP-12473-A (Non-Proprietary).

(Methodology for Specification 3.2.5 - OPDMS - Monitored Parameters.)

6. APP-GW-GLR-137, Revision 1, “Bases of Digital Overpower and Overtemperature Delta-T (OP□T/OT□T) Reactor Trips,” Westinghouse Electric Company LLC.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, and 3.3.1 – Reactor Trip System (RTS) Instrumentation.)

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Passive Core Cooling Systems limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

3.4.3, “RCS Pressure and Temperature (P/T) Limits”; and
3.4.14, “Low Temperature Overpressure Protection (LTOP).”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.” (Limits for LCO 3.4.3 and LCO 3.4.14).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B of LCO 3.3.17, “Post Accident Monitoring (PAM) Instrumentation,” a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6 Reporting Requirements

5.6.6 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.4, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device’s dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designees, and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displaces radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

APPENDIX B

VIRGIL C. SUMMER NUCLEAR STATION UNIT 2

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) objectives are to ensure compliance with Biological Opinions issued pursuant to the Endangered Species Act of 1973, as amended (ESA), and to ensure that the U.S. Nuclear Regulatory Commission (NRC) is kept informed of other environmental matters. The EPP is intended to be consistent with Federal, State, and local requirements for environmental protection.

2.0 Environmental Protection Issues

In the Final Environmental Impact Statement (FEIS) dated April 2011, the staff considered the environmental impacts associated with the construction and operation of Virgil C. Summer Nuclear Station Unit Nos. 2 and 3. This EPP applies to the licensees' actions affecting the protected environmental resources evaluated in the FEIS and the licensees' actions that may affect any newly discovered protected environmental resources.

2.1 Aquatic Resources Issues

Federal agencies other than the NRC, such as the U.S. Environmental Protection Agency and the U.S. Army Corps of Engineers, have jurisdiction to regulate aquatic resources under the Federal Water Pollution Control Act (Clean Water Act or CWA) and the Rivers and Harbors Appropriation Act of 1899 (RHA). Certain water quality environmental considerations identified in the FEIS, including effluent limitations, monitoring requirements, and mitigation measures, are regulated under the licensees' CWA permits, such as National Pollutant Discharge Elimination System and Section 404 permits, and RHA Section 10 permit. Nothing within this EPP shall be construed to place additional requirements on the regulation of aquatic resources except the imposition of the requirements in a Biological Opinion under the ESA (see Section 2.3). The licensees are required to inform the NRC of events or situations concerning aquatic resources consistent with the provisions of 10 CFR 50.72(b)(2)(xi), and this EPP does not expand any reporting requirement required by that regulation.

2.2 Terrestrial Resources Issues

Several statutes govern the regulation of terrestrial resources. For example, the U.S. Fish and Wildlife Service (FWS) regulates matters involving migratory birds and their nests in accordance with the Migratory Bird Treaty Act. Activities affecting migratory birds or their nests may require permits under the Migratory Bird Treaty Act. The FWS also regulates matters involving the protection and taking of bald and golden eagles in accordance with the Bald and Golden Eagle Protection Act. The licensees shall inform NRC of any events or situations concerning terrestrial resources consistent with the provisions of 10 CFR 50.72(b)(2)(xi), and this EPP does not expand any reporting requirement required by that regulation.

2.3 Endangered Species Act of 1973

The NRC may be required to protect some aquatic resources and terrestrial resources in accordance with the ESA. If a Biological Opinion is issued to the NRC in accordance with ESA Section 7 prior to the issuance of the combined license, the licensees shall comply with the terms and conditions set forth in the Incidental Take Statement of the Biological Opinion. If any Federally listed species or critical habitat occurs in an area affected by construction or operation of the plant that was not previously identified as occurring in such areas, including species and critical habitat that were not previously Federally listed, the licensees shall inform the NRC within four hours of discovery. The time of discovery is identified as the specific time when a decision is made to notify another agency or to issue a press release. Similarly, the licensees shall inform the NRC within four hours of discovery of any take, as defined in the ESA, of a Federally listed species or destruction or adverse modification of critical habitat. The four-hour discovery notifications shall be made to the NRC Operations Center via the Emergency Notification System. The licensees shall provide any necessary information to the NRC if the NRC initiates or reinitiates consultation under the ESA.

Unusual Event - The licensees shall inform the NRC of any onsite mortality, injury, or unusual occurrence of any species protected by the ESA within four hours of discovery, followed by a written report in accordance with Section 4.1. The time of discovery is identified as the specific time when a decision is made to notify another agency or to issue a press release. Such incidents shall be reported regardless of the licensees' assessment of causal relation to plant construction or operation.

3.0 Consistency Requirements

The licensees shall notify the NRC of proposed changes to permits or certifications concerning aquatic or terrestrial resources by providing the NRC with a copy of the proposed change(s) at the same time it is submitted to the permitting agency. The licensees shall provide the NRC with a copy of the application for renewal of permits or certifications at the same time the application is submitted to the permitting agency.

Changes to or renewals of such permits or certifications shall be reported to the NRC within 30 days following the later of the date the change or renewal is approved or the date the change becomes effective. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

4.0 Administrative Procedures

4.1 Plant Reporting Requirements: Non-routine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of any unusual event described in Section 2.3 of this EPP. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics at the time of the event, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection, which also require reports to other Federal, State, or local agencies, shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

4.2 Review and Audit

The licensees shall provide for review and audit of compliance with Section 2.3 of this EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organizational structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

4.3 Records Retention

Records required by this EPP shall be made and retained in a manner convenient for review and inspection. These records shall be made available to the NRC on request. The records, data, and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

4.4 Changes in Environmental Protection Plan

A request for a change in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

The licensees shall request a license amendment to incorporate the requirements of any Terms and Conditions set forth in the Incidental Take Statement of applicable Biological Opinions issued subsequent to the effective date of this EPP

APPENDIX C

VIRGIL C. SUMMER NUCLEAR STATION UNIT 2

INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

Table of Contents

1.0 Introduction

1.1 Definitions

The following definitions apply to terms used in the design descriptions and associated inspections, tests, analyses, and acceptance criteria (ITAAC).

Acceptance Criteria means the performance, physical condition, or analysis result for a structure, system, or component that demonstrates that the design or program commitment is met.

Analysis means a calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar structures, systems, or components.

As-built means the physical properties of a structure, system, or component following the completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, determination of physical properties of the as-built structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing does not alter the properties.

Column Line is the designation applied to a plant reference grid used to define the location of building walls and columns. Column lines may not represent the center line of walls and columns.

Design Commitment means that portion of the design description that is verified by ITAAC.

Design Description means that portion of the design that is certified.

Design Plant Grade means the elevation of the soil around the nuclear island assumed in the design of the AP1000, i.e., floor elevation 100'-0".

Division (for electrical systems or electrical equipment) is the designation applied to a given safety-related system or set of components that is physically, electrically, and functionally independent from other redundant sets of components.

Floor Elevation is the designation applied to name a floor. The actual elevation may vary due to floor slope and layout requirements.

Functional Arrangement (for a system) means the physical arrangement of systems and components to provide the service for which the system is intended, and which is described in the system design description.

Inspect or **Inspection** means visual observations, physical examinations, or reviews of records based on visual observation or physical examination that compare a) the structure, system, or component condition to one or more design commitments or b) the program implementation elements to one or more program commitments, as applicable. Examples include walkdowns, configuration checks, measurements of dimensions, or nondestructive examinations.

Inspect for Retrievability of a display means to visually observe that the specified information appears on a monitor when summoned by the operator.

ITAAC number is a unique number based on three character strings. The first string represents the source of the ITAAC where a C or E denotes the ITAAC source is from the combined license or early site permit respectively. No alpha character denotes the ITAAC source is from the design control document (DCD). The second string represents the chapter, section, and subsection where the ITAAC table is located within the source document and contains three or more numbers separated by decimals. If the source document is not numbered, the string is based on the ITAAC table number within this appendix. The third string identifies the location of the ITAAC within the table and will vary in length and composition based on the source table numbering convention.

$\mathbf{L_a}$ is the maximum allowable containment leakage as defined in 10 CFR 50 Appendix J.

Physical Arrangement (for a structure) means the arrangement of the building features (e.g., floors, ceilings, walls, and basemat) and of the structures, systems, and components within, which are described in the building design description.

Program Commitment means that portion of the program description that is verified by ITAAC. The bracketed, alphanumerical designations included in the emergency planning ITAAC identify the evaluation criteria (i.e., program elements) from NUREG-0654/FEMA-REP-1 Planning Standards that were used to develop the specific generic ITAAC in NUREG-0800, Table 14.3.10-1.

Qualified for Harsh Environment means that equipment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of its safety function, for the time required to perform the safety function. These environmental conditions include applicable time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and their synergistic effects which have a significant effect on the equipment performance. Equipment identified in the Design Description as being Qualified for Harsh Environment includes the:

a. equipment itself
b. sensors, switches and lubricants that are an integral part of the equipment
c. electrical components connected to the equipment (wiring, cabling and terminations)

Items b and c are Qualified for Harsh Environment only when they are necessary to support operation of the equipment to meet its safety-related function listed in the Design Description table and to the extent such equipment is located in a harsh environment during or following a design basis accident.

Sensor means a transmitter, resistance temperature detector, thermocouple or other transducer, plus associated cables, connectors, preamplifiers, reference junction boxes, or other signal processing equipment that is located in the immediate proximity of the sensor and subject to the same environmental conditions.

Site Grade means the as-built elevation of the soil to the west side of the nuclear island. Adjacent buildings are located on the other sides of the nuclear island.

Tag Number in the ITAACs represents the complete tag number or a portion of the tag number used to identify the actual hardware (or associated software). For instrumentation, the tag

number identified in the ITAACs does not include the type of instrument (for example, the Containment Exhaust Fan A Flow Sensor, VFS-11A, does not include the designators FE [flow element] or FT [flow transmitter], which would appear on the actual hardware or in the associated software). This is because the designator VFS-11A and the equipment description are sufficient to uniquely identify the channel associated with the designated instrument function, and this method of identification eliminates the need to list every portion of the instrumentation channel required to perform the function. In most cases, the channel number includes physical hardware. There are, however, a few places where the channel number represents only a calculation in software. In those cases, the channel data can be displayed. In many instances, the word "sensor" is used in the equipment description to identify that the item is an instrument.

Test means the actuation, operation, or establishment of specified conditions to evaluate the performance or integrity of as-built structures, systems, or components, unless explicitly stated otherwise.

Transfer Open (Closed) means to move from a closed (open) position to an open (closed) position.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of the same type and manufacturer. A type test is not necessarily a test of the as-built structures, systems, or components.

UA of a heat exchanger means the product of the heat transfer coefficient and the surface area.

1.2 General Provisions

The following general provisions are applicable to the design descriptions and associated ITAAC.

Treatment of Individual Items

The absence of any discussion or depiction of an item in the design description or accompanying figures shall not be construed as prohibiting a licensee from utilizing such an item, unless it would prevent an item from performing its safety functions as discussed or depicted in the design description or accompanying figures.

If an inspections, tests, or analyses (ITA) requirement does not specify the temperature or other conditions under which a test must be run, then the test conditions are not constrained.

When the term "operate," "operates," or "operation" is used with respect to an item discussed in the acceptance criteria, it refers to the actuation and running of the item. When the term "exist," "exists," or "existence" is used with respect to an item discussed in the acceptance criteria, it means that the item is present and meets the design commitment.

Implementation of ITAAC

The ITAAC are provided in tables with the following three-column format:

Design (or Program) Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each design or program commitment in the left-hand column of the ITAAC tables has an associated ITA requirement specified in the middle column of the tables.

The identification of a separate ITA entry for each design or program commitment shall not be construed to require that separate inspections, tests, or analyses must be performed for each design or program commitment. Instead, the activities associated with more than one ITA entry may be combined, and a single inspection, test, or analysis may be sufficient to implement more than one ITA entry.

An ITA may be performed by the licensee of the plant or by its authorized vendors, contractors, or consultants. Furthermore, an ITA may be performed by more than a single individual or group, may be implemented through discrete activities separated by time, and may be performed at any time prior to fuel load (including before issuance of the combined license for those ITAACs that do not necessarily pertain to as-installed equipment). Additionally, an ITA may be performed as part of the activities that are required to be performed under 10 CFR Part 50 (including, for example, the quality assurance (QA) program required under Appendix B to Part 50); therefore, an ITA need not be performed as a separate or discrete activity.

Many of the acceptance criteria include the words “A report exists and concludes that…” When these words are used, it indicates that the ITAAC for that design commitment will be met when it is confirmed that appropriate documentation exists and the documentation shows that the design commitment is met. Appropriate documentation can be a single document or a collection of documents that show that the stated acceptance criteria are met. Examples of appropriate documentation include design reports, test reports, inspection reports, analysis reports, evaluation reports, design and manufacturing procedures, certified data sheets, commercial dedication procedures and records, quality assurance records, calculation notes, and equipment qualification data packages. For plants at sites which are qualified using the hard rock high frequency (HRHF) ground motion response spectra (GMRS), high frequency seismic screening and qualification testing required as a result of the evaluation of potential high frequency sensitive components is included in the equipment qualification data packages.

Many entries in the ITA column of the ITAAC tables include the words “Inspection will be performed for the existence of a report verifying…” When these words are used it indicates that the ITA is tests, type tests, analyses, or a combination of tests, type tests, and analyses and a report will be produced documenting the results. This report will be available to inspectors.

Many ITAAC are only a reference to another location, either a section, subsection, or ITAAC table entry (for example, “See ITAAC table entry…”). A reference to another ITAAC location is always in both the ITA and acceptance criteria columns for a design commitment. This reference is an indication that the ITA and acceptance criteria for that design commitment are satisfied when the referenced ITA are completed and the acceptance criteria for the referenced sections, subsections, or table entries are satisfied. If a complete section is referenced, this indicates that all the ITA and acceptance criteria in that section must be met before the referencing design commitment is satisfied.

Discussion of Matters Related to Operations

In some cases, the design descriptions in this document refer to matters that relate to operation, such as normal valve or breaker alignment during normal operation modes. Such discussions are provided solely to place the design description provisions in context (for example, to explain automatic features for opening or closing valves or breakers upon off-normal conditions). Such discussions shall not be construed as requiring operators during operation to take any particular action (for example, to maintain valves or breakers in a particular position during normal operation).

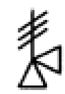
Interpretation of Figures

In many but not all cases, the design descriptions in Section 2 include one or more figures. The figures may represent a functional diagram, general structural representation, or another general illustration. For instrumentation and control (I&C) systems, figures may also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, the figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on the figures, provided that those safety functions discussed in the design description pertaining to the figure are not adversely affected.

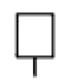
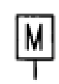
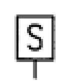
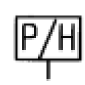
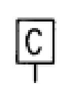
1.3 Figure Legend

The conventions used in this section are for figures described in the design description. The figure legend is provided for information.

VALVES

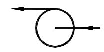
Valve	
Check Valve	
Relief Valve	

VALVE OPERATORS

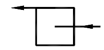
Operator Of Unspecified Type	
Motor Operator	
Solenoid Operator	
Pneumatic/Hydraulic Operator	
Pneumatic Operator	
Squib Valve	

MECHANICAL EQUIPMENT

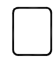
Centrifugal Pump



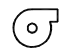
Pump Type Not Specified



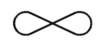
Tank



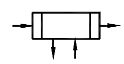
Centrifugal Fan



Axial Fan



Heat Exchanger

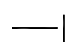


Vent

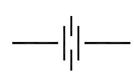
Drain

Pipe Cap

Blind Flange



Orifice



DAMPERS

Gravity Or Manually Operated Damper 

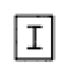
Remotely Operated Damper 

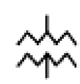
ELECTRICAL EQUIPMENT

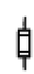
Battery 

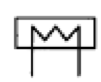
Circuit Breaker 

Disconnect Switch 

Isolation 

Transformer 

Fuse 

Heater 

Generator 

MISCELLANEOUS

A component that is part of the system functional arrangement shown on the figure and is included in the design commitments for the system.

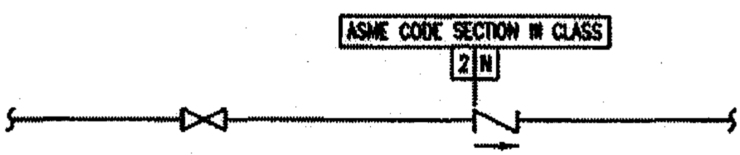
A component that is part of the system functional arrangement shown on the figure.

A system or component of another system that is not part of the system functional arrangement shown on the figure.

A functional connection to another system that is not part of the system functional arrangement shown on the figure.

ASME CODE CLASS BREAK

An ASME Code class break is identified by a single line to the designated location for the class break, as shown in the example below (see note 1).



NOTES:

1. The header, “ASME Code Section III Class,” must appear at least once on each figure on which ASME class breaks are shown, but need not appear at every class break shown on a figure.

   [N] Indicates Non-ASME Code Section III

1.4 List of Acronyms and Abbreviations

The acronyms presented in this section are provided for information.

ac	Alternating Current
AC	Acceptance Criteria
ADS	Automatic Depressurization System
AHU	Air Handling Units
ANS	Alert and Notification System
ASME	American Society of Mechanical Engineers
atm	Atmoshpere
BTU	British Thermal Unit
CAS	Compressed and Instrument Air System
CAV	Cumulative Absolute Velocity
cc	Cubic Centimeter
CCS	Component Cooling Water System
CDE	Committed Dose Equivalent
CDS	Condensate System
cfm	Cubic Feet per Minute
CFR	Code of Federal Regulations
Ci	Curie
CIM	Component Interface Module
CMT	Core Makeup Tank
CNS	Containment System
COL	Combined License
CRDM	Control Rod Drive Mechanism
CSA	Control Support Area
CST	Condensate Storage Tank
CVS	Chemical and Volume Control System
CWS	Circulating Water System
DAS	Diverse Actuation System
DBT	Design Basis Threat
dc	Direct Current
DCD	Design Control Document
DDS	Data Display and Processing System
DOS	Standby Diesel Fuel Oil System
D-RAP	Design Reliability Assurance Program
DTS	Demineralized Water Treatment System
DVI	Direct Vessel Injection
DWS	Demineralized Water Transfer and Storage System
EAL	Emergency Action Level
ECS	Main ac Power System
EDS	Non-Class 1E dc and Uninterruptible Power Supply System
EFS	Communication System

List of Acronyms and Abbreviations (cont.)

EGS	Grounding and Lightening Protection System
EIP	Emergency Implementing Procedure
El.	Elevation
ELS	Plant Lighting System
EMI	Electromagnetic Interference
ENC	Emergency News Center
EOC	Emergency Operations Center
EOF	Emergency Operations Facility
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
ERDS	Emergency Response Data System
ESD	Electrostatic Discharge
F	Fahrenheit
FE	Flow Element
FHM	Fuel Handling Machine
FHS	Fuel Handling and Refueling System
FPS	Fire Protection System
ft	Feet
FT	Flow Transmitter
FWS	Main and Startup Feedwater System
GEMA	Georgia Emergency Management Agency
GMRS	Ground Motion Response Spectra
gpm	Gallons per Minute
GRCA	Gray Rod Cluster Assemblies
GSU	Generator Stepup Transformer
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HL	Hot Leg
hr	Hour
HRHF	Hard Rock High Frequency
HSI	Human-System Interface
HVAC	Heating, Ventilation, and Air Conditioning
HX	Heat Exchanger
Hz	Hertz
I&C	Instrumentation and Control
IDS	Class 1E dc and Uninterruptible Power Supply System
IIS	In-core Instrumentation System
in	Inches
I&C	Instrumentation and Control
IRC	Inside Reactor Containment
IRWST	In-containment Refueling Water Storage Tank
ITA	Inspections, Tests, Analyses

List of Acronyms and Abbreviations (cont.)

ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
lb/hr	Pounds per Hour
JIC	Joint Information Center
kW	Kilowatt
LBB	Leak Before Break
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Overpressure Protection
MBtu	Million British Thermal Units
MCC	Motor Control Center
MCR	Main Control Room
MHS	Mechanical Handling System
MOV	Motor-operated Valve
MSIV	Main Steam Isolation Valve
MSS	Main Steam System
MTS	Main Turbine System
MW	Megawatt
MWe	Megawatt Electric
MWt	Megawatt Thermal
NI	Nuclear Island
NRC	U.S. Nuclear Regulatory Commission
OCS	Operation and Control Centers System
ODCM	Offsite Dose Calculation Manual
ORC	Outside Reactor Containment
OSC	Operations Support Center
PAG	Protection Action Guide
PAR	Protective Action Recommendation
PCCAWST	Passive Containment Cooling Ancillary Water Storage Tank
PCCWST	Passive Containment Cooling Water Storage Tank
PCS	Passive Containment Cooling System
PGS	Plant Gas System
pH	Potential of Hydrogen
PLS	Plant Control System
PMS	Protection and Safety Monitoring System
PNS	Prompt Notification System
PORV	Power-operated Relief Valve
PRHR	Passive Residual Heat Removal
psia	Pounds per Square Inch Absolute
psig	Pounds per Square Inch Gauge
PSS	Primary Sampling System
pu	Per Unit
PWS	Potable Water System
PXS	Passive Core Cooling System

List of Acronyms and Abbreviations (cont.)

QA	Quality Assurance
RAP	Reliability Assurance Program
R/hr	Roentgen per hour
RAT	Reserve Auxiliary Transformer
RC	Reinforced Concrete
RCCA	Rod Cluster Control Assembly
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFI	Radio Frequency Interference
RM	Refueling Machine
RMS	Radiation Monitoring System
RNS	Normal Residual Heat Removal System
RPV	Reactor Pressure Vessel
RSR	Remote Shutdown Room
RSW	Remote Shutdown Workstation
RTD	Resistance Temperature Detector
RXS	Reactor System
RV	Reactor Vessel
SC	Steel and Concrete
SCEMD	South Carolina Emergency Management Division
scf	Standard Cubic Feet
scfm	Standard Cubic Feet per Minute
SDS	Sanitary Drainage System
SFHT	Spent Fuel Handling Tool
SFP	Spent Fuel Pool
SFS	Spent Fuel Pool Cooling System
SG	Steam Generator
SGS	Steam Generator System
SJS	Seismic Monitoring System
SMS	Special Monitoring System
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SWC	Surge Withstand Capability
SWS	Service Water System
TEDE	Total Effective Dose Equivalent
TSC	Technical Support Center
UAT	Unit Auxiliary Transformer
UPS	Uninterruptible Power Supply
V	Volt
VAS	Radiologically Controlled Area Ventilation System
VBS	Nuclear Island Nonradioactive Ventilation System

List of Acronyms and Abbreviations (cont.)

VCS	Containment Recirculation Cooling System
VCSNS	Virgil C. Summer Nuclear Station
Vdc	Direct Current Voltage
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VHS	Health Physics and Hot Machine Shop Areas
VLS	Containment Hydrogen Control System
VRS	Radwaste Building HVAC System
VWS	Central Chilled Water System
VXS	Annex/Auxiliary Building Nonradioactive Ventilation System
VZS	Diesel Generator Building Ventilation System
wg	Water Gauge
WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
WSS	Solid Radwaste System
WWS	Waste Water System
WRS	Radioactive Waste Drain System
ZOI	Zone of Influence
ZOS	Onsite Standby Power System

2.0 System Based Design Descriptions and ITAAC

2.1 Reactor

2.1.1 Fuel Handling and Refueling System

Design Description

The fuel handling and refueling system (FHS) transfers fuel assemblies and core components during fueling operations and stores new and spent fuel assemblies in the new and spent fuel storage racks. The refueling machine (RM) and the fuel transfer tube are operated during refueling mode. The fuel handling machine (FHM) is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the FHS are as shown in Table 2.1.1-2.

1. The functional arrangement of the FHS is as described in the Design Description of this Section 2.1.1.
2. The FHS has the RM, the FHM, and the new and spent fuel storage racks.
3. The FHS preserves containment integrity by isolation of the fuel transfer tube penetrating containment.
4. The RM and FHM/spent fuel handling tool (SFHT) gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers.
5. The lift height of the RM mast and FHM hoist(s) is limited such that the minimum required depth of water shielding is maintained.
6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.
7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.

Table 2.1.1-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1	2.1.01.01	1. The functional arrangement of the FHS is as described in the Design Description of this Section 2.1.1.	Inspection of the as-built system will be performed.	The as-built FHS conforms with the functional arrangement as described in the Design Description of this Section 2.1.1.
2	2.1.01.02	2. The FHS has the refueling machine (RM), the fuel handling machine (FHM), and the new and spent fuel storage racks.	Inspection of the system will be performed.	The FHS has the RM, the FHM, and the new and spent fuel storage racks.
3	2.1.01.03	3. The FHS preserves containment integrity by isolation of the fuel transfer tube penetrating containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
4	2.1.01.04	4. The RM and FHM/spent fuel handling tool (SFHT) gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers.	The RM and FHM/SFHT gripper assemblies will be tested by operating the open controls of the gripper while suspending a dummy fuel assembly.	The RM and FHM/SFHT gripper assemblies will not open while suspending a dummy test assembly.
5	2.1.01.05	5. The lift height of the RM mast and FHM hoist(s) is limited such that the minimum required depth of water shielding is maintained.	The RM and FHM will be tested by attempting to raise a dummy fuel assembly.	The bottom of the dummy fuel assembly cannot be raised to within 24 ft, 6 in. of the operating deck floor.
6	2.1.01.06.i	6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.	i) Inspection will be performed to verify that the RM and FHM are located on the nuclear island.	i) The RM and FHM are located on the nuclear island.
7	2.1.01.06.ii	6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.	ii) Type test, analysis, or a combination of type tests and analyses of the RM and FHM will be performed.	ii) A report exists and concludes that the RM and FHM can withstand seismic design basis dynamic loads without loss of load carrying or structural integrity functions.
8	2.1.01.07.i	7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.	i) Analyses will be performed to calculate the effective neutron multiplication factor in the new and spent fuel storage racks during normal conditions.	i) The calculated effective neutron multiplication factor for the new and spent fuel storage racks meets the requirements of 10 CFR 50.68(1) limits under normal conditions.

Table 2.1.1-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
9	2.1.01.07.ii	7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.	ii) Inspection will be performed to verify that the new and spent fuel storage racks are located on the nuclear island.	ii) The new and spent fuel storage racks are located on the nuclear island.
10	2.1.01.07.iii	7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.	iii) Seismic analysis of the new and spent fuel storage racks will be performed.	iii) A report exists and concludes that the new and spent fuel racks can withstand seismic design basis dynamic loads and maintain the calculated effective neutron multiplication factor required by 10 CFR 50.68(1) limits.
11	2.1.01.07.iv	7. The new and spent fuel storage racks maintain the effective neutron multiplication factor required by 10 CFR 50.68 limits during normal operation, design basis seismic events, and design basis dropped spent fuel assembly accidents over the spent fuel storage racks.	iv) Analysis of the spent fuel storage racks under design basis dropped spent fuel assembly loads will be performed.	iv) A report exists and concludes that the spent fuel racks can withstand design basis dropped spent fuel assembly loads and maintain the calculated effective neutron multiplication factor required by 10 CFR 50.68(1) limits.

Note:

1. The requirements of 10 CFR 50.68 are summarized as follows:
   - For new fuel storage racks:
     - The effective neutron multiplication factor (K-effective) must not exceed 0.95 when flooded with unborated water and
     - K-effective must not exceed 0.98 with optimum moderator conditions.
   - For spent fuel storage racks:
     - If methodology does not take credit for soluble boron:
       - K-effective must not exceed 0.95 when flooded with unborated water.
     - Or if methodology takes credit for soluble boron:
       - K-effective must not exceed 0.95 when flooded with borated water and
       - K-effective must remain below 1.0 when flooded with unborated water.

Table 2.1.1-2		
Component Name	**Tag No.**	**Component Location**
Refueling Machine	FHS-FH-01	Containment
Fuel Handling Machine	FHS-FH-02	Auxiliary Building
Spent Fuel Storage Racks	FHS-FS-02	Auxiliary Building
New Fuel Storage Racks	FHS-FS-01	Auxiliary Building
Fuel Transfer Tube	FHS-FT-01	Auxiliary Building/Containment

2.1.2 Reactor Coolant System

Design Description

The reactor coolant system (RCS) removes heat from the reactor core and transfers it to the secondary side of the steam generators for power generation. The RCS contains two vertical U-tube steam generators, four sealless reactor coolant pumps (RCPs), and one pressurizer.

The RCS is as shown in Figure 2.1.2-1 and the component locations of the RCS are as shown in Table 2.1.2-5.

1. The functional arrangement of the RCS is as described in the Design Description of this Section 2.1.2.
2. a) The components identified in Table 2.1.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.1.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.1.2-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.1.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.1.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.
5. a) The seismic Category I equipment identified in Table 2.1.2-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.1.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.1.2-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

7. a) The Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.1.2-1 are powered from their respective Class 1E division.

   c) Separation is provided between RCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The RCS provides the following safety-related functions:

   a) The pressurizer safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

   b) The reactor coolant pumps (RCPs) have a rotating inertia to provide RCS flow coastdown on loss of power to the pumps.

   c) Each RCP flywheel assembly can withstand a design overspeed condition.

   d) The RCS provides automatic depressurization during design basis events.

   e) The RCS provides emergency letdown during design basis events.

9. The RCS provides the following nonsafety-related functions:

   a) The RCS provides circulation of coolant to remove heat from the core.

   b) The RCS provides the means to control system pressure.

   c) The pressurizer heaters trip after a signal is generated by the PMS.

10. Safety-related displays identified in Table 2.1.2-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.1.2-1 to perform active functions.

    b) The valves identified in Table 2.1.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

    c) The valves identified in Table 2.1.2-1 as having diverse actuation system (DAS) control perform an active safety function after receiving a signal from DAS.

12. a) The valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.1.2-1 assume the indicated loss of motive power position.

13. a) Controls exist in the MCR to trip the RCPs.

    b) The RCPs trip after receiving a signal from the PMS.

    c) The RCPs trip after receiving a signal from the DAS.

14. Controls exist in the MCR to cause the components identified in Table 2.1.2-3 to perform the listed function.

15. Displays of the parameters identified in Table 2.1.2-3 can be retrieved in the MCR.

Table 2.1.2-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
Steam Generator 1	RCS-MB-01	Yes	Yes	-	-/-	-	-	-	-
Steam Generator 2	RCS-MB-02	Yes	Yes	-	-/-	-	-	-	-
RCP 1A	RCS-MP-01A	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
RCP 1B	RCS-MP-01B	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
RCP 2A	RCS-MP-02A	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
RCP 2B	RCS-MP-02B	Yes	Yes	-	No/No	No	Yes/Yes (pump trip)	No	-
Pressurizer	RCS-MV-02	Yes	Yes	-	No/No (heaters)	-	Yes/No (heater trip)	No	-
Automatic Depressurization System (ADS) Sparger A	PXS-MW-01A	Yes	Yes	-	-/-	-	-/-	-	-
ADS Sparger B	PXS-MW-01B	Yes	Yes	-	-/-	-	-/-	-	-
Pressurizer Safety Valve	RCS-PL-V005A	Yes	Yes	No	-/-	No	-/-	Transfer Open/ Transfer Closed	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
Pressurizer Safety Valve	RCS-PL-V005B	Yes	Yes	No	-/-	No	-/-	Transfer Open/ Transfer Closed	-
First-stage ADS Motor-operated Valve (MOV)	RCS-PL-V001A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
First-stage ADS MOV	RCS-PL-V001B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Second-stage ADS MOV	RCS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Second-stage ADS MOV	RCS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS MOV	RCS-PL-V003A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS MOV	RCS-PL-V003B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004C	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS Squib Valve	RCS-PL-V004D	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
ADS Discharge Header A Vacuum Relief Valve	RCS-PL-V010A	Yes	Yes	No	Yes/Yes	No	No/No	Transfer Open	-
ADS Discharge Header B Vacuum Relief Valve	RCS-PL-V010B	Yes	Yes	No	Yes/Yes	No	No/No	Transfer Open	-
First-stage ADS Isolation MOV	RCS-PL-V011A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
First-stage ADS Isolation MOV	RCS-PL-V011B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Second-stage ADS Isolation MOV	RCS-PL-V012A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
Second-stage ADS Isolation MOV	RCS-PL-V012B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS Isolation MOV	RCS-PL-V013A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Third-stage ADS Isolation MOV	RCS-PL-V013B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
Fourth-stage ADS MOV	RCS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Fourth-stage ADS MOV	RCS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Fourth-stage ADS MOV	RCS-PL-V014C	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Fourth-stage ADS MOV	RCS-PL-V014D	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	None	As Is
Reactor Vessel Head Vent Valve	RCS-PL-V150A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed / Transfer Open	Closed
Reactor Vessel Head Vent Valve	RCS-PL-V150B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed / Transfer Open	Closed
Reactor Vessel Head Vent Valve	RCS-PL-V150C	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed / Transfer Open	Closed
Reactor Vessel Head Vent Valve	RCS-PL-V150D	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed / Transfer Open	Closed
RCS Hot Leg 1 Flow Sensor	RCS-101A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Flow Sensor	RCS-101B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Flow Sensor	RCS-101C	-	Yes	-	Yes/Yes	No	-/-	-	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
RCS Hot Leg 1 Flow Sensor	RCS-101D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Flow Sensor	RCS-102D	-	Yes	-	Yes/Yes	No	-/-	-	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
RCS Cold Leg 1A Narrow Range Temperature Sensor	RCS-121A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1B Narrow Range Temperature Sensor	RCS-121B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1B Narrow Range Temperature Sensor	RCS-121C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1A Narrow Range Temperature Sensor	RCS-121D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2B Narrow Range Temperature Sensor	RCS-122A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2A Narrow Range Temperature Sensor	RCS-122B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2A Narrow Range Temperature Sensor	RCS-122C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 2B Narrow Range Temperature Sensor	RCS-122D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Cold Leg 1A Dual Range Temperature Sensor	RCS-125A	-	Yes	-	Yes/Yes	Yes (Wide Range)	-/-	-	-
RCS Cold Leg 2A Dual Range Temperature Sensor	RCS-125B	-	Yes	-	Yes/Yes	Yes (Wide Range)	-/-	-	-
RCS Cold Leg 1B Dual Range Temperature Sensor	RCS-125C	-	Yes	-	Yes/Yes	Yes (Wide Range	-/-	-	-
RCS Cold Leg 2B Dual Range Temperature Sensor	RCS-125D	-	Yes	-	Yes/Yes	Yes (Wide Range)	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-131A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-131B	-	Yes	-	Yes/Yes	No	-/-	-	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-131C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-131D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-132A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-132B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-132C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-132D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-133A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-133B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Narrow Range Temperature Sensor	RCS-133C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 2 Narrow Range Temperature Sensor	RCS-133D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCS Hot Leg 1 Wide Range Temperature Sensor	RCS-135A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Wide Range Temperature Sensor	RCS-135B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Wide Range Pressure Sensor	RCS-140D	-	Yes	-	Yes/Yes	Yes	-/-	-	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
RCS Hot Leg 1 Level Sensor	RCS-160A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCS Hot Leg 2 Level Sensor	RCS-160B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Passive Residual Heat Removal (PRHR) Return Line Temperature Sensor	RCS-161	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Pressure Sensor	RCS-191D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Reference Leg Temperature Sensor	RCS-193D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Pressurizer Level Sensor	RCS-195D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RCP 1A Bearing Water Temperature Sensor	RCS-211A	-	Yes	-	Yes/Yes	No	-/-	-	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
RCP 1A Bearing Water Temperature Sensor	RCS-211B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1A Bearing Water Temperature Sensor	RCS-211C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1A Bearing Water Temperature Sensor	RCS-211D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1B Bearing Water Temperature Sensor	RCS-212D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213A	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2A Bearing Water Temperature Sensor	RCS-213D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214A	-	Yes	-	Yes/Yes	No	-/-	-	-

Table 2.1.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
RCP 2B Bearing Water Temperature Sensor	RCS-214B	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214C	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2B Bearing Water Temperature Sensor	RCS-214D	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1A Pump Speed Sensor	RCS-281	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 1B Pump Speed Sensor	RCS-282	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2A Pump Speed Sensor	RCS-283	-	Yes	-	Yes/Yes	No	-/-	-	-
RCP 2B Pump Speed Sensor	RCS-284	-	Yes	-	Yes/Yes	No	-/-	-	-

Note: Dash (-) indicates not applicable.

Table 2.1.2-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
Hot Legs	RCS-L001A RCS-L001B	Yes	Yes	Yes
Cold Legs	RCS-L002A RCS-L002B RCS-L002C RCS-L002D	Yes	Yes	Yes
Pressurizer Surge Line	RCS-L003	Yes	Yes	Yes
ADS Inlet Headers	RCS-L004A/B RCS-L006A/B RCS-L030A/B RCS-L020A/B	Yes	Yes	Yes
Safety Valve Inlet Piping	RCS-L005A RCS-L005B	Yes	Yes	Yes

Table 2.1.2-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
Safety Valve Discharge Piping	RCS-L050A/B RCS-L051A/B	Yes	No	Yes
	RCS-L064A/B	Yes	No	No
ADS First-stage Valve Inlet Piping	RCS-L010A/B RCS-L011A/B	Yes	No	Yes
ADS Second-stage Valve Inlet Piping	RCS-L021A/B RCS-L022A/B	Yes	Yes No	Yes
ADS Third-stage Valve Inlet Piping	RCS-L131 RCS-L031A/B RCS-L032A/B	Yes	Yes Yes No	Yes
ADS Outlet Piping	RCS-L012A/B RCS-L023A/B RCS-L033A/B RCS-L061A/B RCS-L063A/B RCS-L200 RCS-L069A/B PXS-L130A/B	Yes	No	Yes
	RCS-L240A/B	Yes	No	No
ADS Fourth-stage Inlet Piping	RCS-L133A/B RCS-L135A/B RCS-L136A/B RCS-L137A/B	Yes	Yes	Yes
Pressurizer Spray Piping	RCS-L106 RCS-L110A/B RCS-L212A/B RCS-L213 RCS-L215	Yes	No	No
RNS Suction Piping	RCS-L139 RCS-L140	Yes	Yes	No
CVS Purification Piping	RCS-L111 RCS-L112	Yes	No	No

Table 2.1.2-3			
Equipment	**Tag No.**	**Display**	**Control Function**
RCP 1A Breaker (Status)	ECS-ES-31	Yes	-
RCP 1A Breaker (Status)	ECS-ES-32	Yes	-
RCP 1B Breaker (Status)	ECS-ES-41	Yes	-

Table 2.1.2-3			
Equipment	**Tag No.**	**Display**	**Control Function**
RCP 1B Breaker (Status)	ECS-ES-42	Yes	-
RCP 2A Breaker (Status)	ECS-ES-51	Yes	-
RCP 2A Breaker (Status)	ECS-ES-52	Yes	-
RCP 2B Breaker (Status)	ECS-ES-61	Yes	-
RCP 2B Breaker (Status)	ECS-ES-62	Yes	-
Pressurizer Heaters	RCS-EH-03	Yes	On/Off
Pressurizer Heaters	RCS-EH-04A	Yes	On/Off
Pressurizer Heaters	RCS-EH-04B	Yes	On/Off
Pressurizer Heaters	RCS-EH-04C	Yes	On/Off
Pressurizer Heaters	RCS-EH-04D	Yes	On/Off
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004A	Yes	-
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004B	Yes	-
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004C	Yes	-
Fourth-stage ADS Squib Valve (Position Indication)	RCS-PL-V004D	Yes	-
Pressurizer Safety Valve (Position Indication)	RCS-PL-V005A	Yes	-
Pressurizer Safety Valve (Position Indication)	RCS-PL-V005B	Yes	-
Pressurizer Spray Valve (Position Indication)	RCS-PL-V110A	Yes	-
Pressurizer Spray Valve (Position Indication)	RCS-PL-V110B	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150A	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150B	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150C	Yes	-
Reactor Vessel Head Vent Valve (Position Indication)	RCS-PL-V150D	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
12	2.1.02.01	1. The functional arrangement of the RCS is as described in the Design Description of this Section 2.1.2.	Inspection of the as-built system will be performed.	The as-built RCS conforms with the functional arrangement described in the Design Description of this Section 2.1.2.
13	2.1.02.02a	2.a) The components identified in Table 2.1.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.1.2-1 as ASME Code Section III.
14	2.1.02.02b	2.b) The piping identified in Table 2.1.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME code Section III design reports exist for the as-built piping identified in Table 2.1.2-2 as ASME Code Section III.
15	2.1.02.03a	3.a) Pressure boundary welds in components identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
16	2.1.02.03b	3.b) Pressure boundary welds in piping identified in Table 2.1.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
17	2.1.02.04a	4.a) The components identified in Table 2.1.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.1.2-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
18	2.1.02.04b	4.b) The piping identified in Table 2.1.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.1.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
19	2.1.02.05a.i	5.a) The seismic Category I equipment identified in Table 2.1.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.1.2-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.1.2-1 is located on the Nuclear Island.
20	2.1.02.05a.ii	5.a) The seismic Category I equipment identified in Table 2.1.2-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
21	2.1.02.05a.iii	5.a) The seismic Category I equipment identified in Table 2.1.2-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
22	2.1.02.05b	5.b) Each of the lines identified in Table 2.1.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.1.2-2 for which functional capability is required meets the requirements for functional capability.
23	2.1.02.06	6. Each of the as-built lines identified in Table 2.1.2-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from dynamic effects of a pipe break. Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built RCS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
24	2.1.02.07a.i	7.a) The Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
25	2.1.02.07a.ii	7.a) The Class 1E equipment identified in Table 2.1.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.1.2-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
26	2.1.02.07b	7.b) The Class 1E components identified in Table 2.1.2-1 are powered from their respective Class 1E division.	Testing will be performed on the RCS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.1.2-1 when the assigned Class 1E division is provided the test signal.
27	2.1.02.07c	7.c) Separation is provided between RCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d).	See ITAAC Table 3.3-6, item 7.d).
28	2.1.02.08a.i	8.a) The pressurizer safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	i) Inspections will be conducted to confirm that the value of the vendor code plate rating is greater than or equal to system relief requirements.	i) The sum of the rated capacities recorded on the valve ASME Code plates of the safety valves exceeds 1,500,000 lb/hr.
29	2.1.02.08a.ii	8.a) The pressurizer safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	ii) Testing and analysis in accordance with ASME Code Section III will be performed to determine set pressure.	ii) A report exists and concludes that the safety valves set pressure is 2485 psig ± 25 psi.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
30	2.1.02.08b	8.b) The RCPs have a rotating inertia to provide RCS flow coastdown on loss of power to the pumps.	A test will be performed to determine the pump flow coastdown curve.	The pump flow coastdown will provide RCS flows greater than or equal to the flow shown in Figure 2.1.2-2, "Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down."
31	2.1.02.08c	8.c) Each RCP flywheel assembly can withstand a design overspeed condition.	Shop testing of each RCP flywheel assembly will be performed at the vendor facility at overspeed conditions.	Each RCP flywheel assembly has passed an overspeed condition of no less than 125% of operating speed.
32	2.1.02.08d.i	8.d) The RCS provides automatic depressurization during design basis events.	i) A low pressure flow test and associated analysis will be conducted to determine the total piping flow resistance of each ADS valve group connected to the pressurizer (i.e., ADS Stages 1-3) from the pressurizer through the outlet of the downstream ADS control valves. The reactor coolant system will be at cold conditions with the pressurizer full of water. The normal residual heat removal pumps will be used to provide injection flow into the RCS discharging through the ADS valves. Inspections and associated analysis of the piping flow paths from the discharge of the ADS valve groups connected to the pressurizer (i.e., ADS Stages 1-3) to the spargers will be conducted to verify the line routings are consistent with the line routings used for design flow resistance calculations.	i) The calculated ADS piping flow resistance from the pressurizer through the sparger with all valves of each ADS group open is $\leq$ 2.91E-6 ft/gpm^2.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
33	2.1.02.08d.ii	8.d) The RCS provides automatic depressurization during design basis events.	ii) Inspections and associated analysis of each fourth-stage ADS valve group (four valves and associated piping connected to each hot leg) will be conducted to verify the line routing is consistent with the line routing used for design flow resistance calculations.	ii) The calculated flow resistance for each group of fourth-stage ADS valves and piping with all valves open is: Loop 1: $\leq 1.70\text{x}10^{-7}$ ft/gpm^2 Loop 2: $\leq 1.57\text{x}10^{-7}$ ft/gpm^2
34	2.1.02.08d.iii	8.d) The RCS provides automatic depressurization during design basis events.	iii) Inspections of each fourth-stage ADS valve will be conducted to determine the as-manufactured flow area through each valve.	iii) The as-manufactured flow area through each fourth-stage ADS valve is $\geq$ 67 in^2.
35	2.1.02.08d.iv	8.d) The RCS provides automatic depressurization during design basis events.	iv) Type tests and analysis will be performed to determine the effective flow area through each stage 1,2,3 ADS valve.	iv) A report exists and concludes that the effective flow area through each stage 1 ADS valve $\geq$ 4.6 in^2 and each stage 2,3 ADS valve is $\geq$ 19 in^2.
36	2.1.02.08d.v	8.d) The RCS provides automatic depressurization during design basis events.	v) Inspections of the elevation of the ADS stage 4 valve discharge will be conducted.	v) The minimum elevation of the bottom inside surface of the outlet of these valves is greater than plant elevation 110 feet.
37	2.1.02.08d.vi	8.d) The RCS provides automatic depressurization during design basis events.	vi) Inspections of the ADS stage 4 valve discharge will be conducted.	vi) The discharge of the ADS stage 4 valves is directed into the steam generator compartments.
38	2.1.02.08d.vii	8.d) The RCS provides automatic depressurization during design basis events.	vii) Inspection of each ADS sparger will be conducted to determine the flow area through the sparger holes.	vii) The flow area through the holes in each ADS sparger is $\geq$ 274 in^2.
39	2.1.02.08d.viii	8.d) The RCS provides automatic depressurization during design basis events.	viii) Inspection of the elevation of each ADS sparger will be conducted.	viii) The centerline of the connection of the sparger arms to the sparger hub is $\leq$ 11.5 feet below the IRWST overflow level.
40	2.1.02.08e	8.e) The RCS provides emergency letdown during design basis events.	Inspections of the reactor vessel head vent valves and inlet and outlet piping will be conducted.	A report exists and concludes that the capacity of the reactor vessel head vent is sufficient to pass not less than 8.2 lbm/sec at 1250 psia in the RCS.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
41	2.1.02.09a	9.a) The RCS provides circulation of coolant to remove heat from the core.	Testing and analysis to measure RCS flow with four reactor coolant pumps operating at no-load RCS pressure and temperature conditions will be performed. Analyses will be performed to convert the measured pre-fuel load flow to post-fuel load flow with 10-percent steam generator tube plugging.	The calculated post-fuel load RCS flow rate is $\geq$ 301,670 gpm.
42	2.1.02.09b.i	9.b) The RCS provides the means to control system pressure.	i) Inspections will be performed to verify the rated capacity of pressurizer heater backup groups A and B.	i) Pressurizer heater backup groups A and B each has a rated capacity of at least 168 kW.
43	2.1.02.09b.ii	9.b) The RCS provides the means to control system pressure.	ii) Tests will be performed to verify that the pressurizer spray valves can open and close when operated from the MCR.	ii) Controls in the MCR operate to cause the pressurizer spray valves to open and close.
44	2.1.02.09c	9.c) The pressurizer heaters trip after a signal is generated by the PMS.	Testing will be performed to confirm trip of the pressurizer heaters identified in Table 2.1.2-3.	The pressurizer heaters identified in Table 2.1.2-3 trip after a signal is generated by the PMS.
45	2.1.02.10	10. Safety-related displays identified in Table 2.1.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.1.2-1 can be retrieved in the MCR.
46	2.1.02.11a.i	11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.1.2-1 to perform active functions.	i) Testing will be performed on the squib valves identified in Table 2.1.2-1 using controls in the MCR without stroking the valve.	i) Controls in the MCR operate to cause a signal at the squib valve electrical leads which is capable of actuating the squib valve.
47	2.1.02.11a.ii	11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.1.2-1 to perform active functions.	ii) Stroke testing will be performed on the other remotely operated valves listed in Table 2.1.2-1 using controls in the MCR.	ii) Controls in the MCR operate to cause the remotely operated valves (other than squib valves) to perform active functions.
48	2.1.02.11b.i	11.b) The valves identified in Table 2.1.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	i) Testing will be performed on the squib valves identified in Table 2.1.2-1 using real or simulated signals into the PMS without stroking the valve.	i) The squib valves receive a signal at the valve electrical leads that is capable of actuating the squib valve.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
49	2.1.02.11b.ii	11.b) The valves identified in Table 2.1.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	ii) Testing will be performed on the other remotely operated valves identified in Table 2.1.2-1 using real or simulated signals into the PMS.	ii) The other remotely operated valves identified in Table 2.1.2-1 as having PMS control perform the active function identified in the table after receiving a signal from PMS.
50	2.1.02.11b.iii	11.b) The valves identified in Table 2.1.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	iii) Testing will be performed to demonstrate that remotely operated RCS valves RCS-V001A/B, V002A/B, V003A/B, V011A/B, V012A/B, V013A/B open within the required response times.	iii) These valves open within the following times after receipt of an actuation signal: V001A/B $\leq$ 40 sec V002A/B, V003A/B $\leq$ 100 sec V011A/B $\leq$ 30 sec V012A/B, V013A/B $\leq$ 60 sec
51	2.1.02.11c.i	11.c) The valves identified in Table 2.1.2-1 as having DAS control perform an active safety function after receiving a signal from DAS.	i) Testing will be performed on the squib valves identified in Table 2.1.2-1 using real or simulated signals into the DAS without stroking the valve.	i) The squib valves receive a signal at the valve electrical leads that is capable of actuating the squib valve.
52	2.1.02.11c.ii	11.c) The valves identified in Table 2.1.2-1 as having DAS control perform an active safety function after receiving a signal from DAS.	ii) Testing will be performed on the other remotely operated valves identified in Table 2.1.2-1 using real or simulated signals into the DAS.	ii) The other remotely operated valves identified in Table 2.1.2-1 as having DAS control perform the active function identified in the table after receiving a signal from DAS.
53	2.1.02.12a.i	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.1.2-1 under design conditions.
54	2.1.02.12a.ii	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
55	2.1.02.12a.iii	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under pre-operational flow, differential pressure and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.1.2-1 under pre-operational test conditions.
56	2.1.02.12a.iv	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	iv) Tests or type tests of squib valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	iv) A test report exists and concludes that each squib valve changes position as indicated in Table 2.1.2-1 under design conditions.
57	2.1.02.12a.v	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	v) Inspection will be performed for the existence of a report verifying that the as-built squib valves are bounded by the tests or type tests.	v) A report exists and concludes that the as-built squib valves are bounded by the tests or type tests.
58	2.1.02.12a.vi	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	vi) See item 8.d.i in this table.	vi) See item 8.d.i in this table. The ADS stage 1-3 valve flow resistances are verified to be consistent with the ADS stage 1-3 path flow resistances.
59	2.1.02.12a.vii	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	vii) See item 8.d.ii in this table.	vii) See item 8.d.ii in this table. The ADS stage 4 valve flow resistances are verified to be consistent with the ADS stage 4 path flow resistances.
60	2.1.02.12a.viii	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	viii) See item 8.d.iii in this table.	viii) See item 8.d.iii in this table.
61	2.1.02.12a.ix	12.a) The automatic depressurization valves identified in Table 2.1.2-1 perform an active safety-related function to change position as indicated in the table.	ix) See item 8.d.iv in this table.	ix) See item 8.d.iv in this table.
62	2.1.02.12b	12.b) After loss of motive power, the remotely operated valves identified in Table 2.1.2-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.1.2-1 assumes the indicated loss of motive power position.

Table 2.1.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
63	2.1.02.13a	13.a) Controls exist in the MCR to trip the RCPs.	Testing will be performed on the RCPs using controls in the MCR.	Controls in the MCR operate to trip the RCPs.
64	2.1.02.13b	13.b) The RCPs trip after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The RCPs trip after receiving a signal from the PMS.
65	2.1.02.13c	13.c) The RCPs trip after receiving a signal from the DAS.	Testing will be performed using real or simulated signals into the DAS.	The RCPs trip after receiving a signal from the DAS.
66	2.1.02.14	14. Controls exist in the MCR to cause the components identified in Table 2.1.2-3 to perform the listed function.	Testing will be performed on the components in Table 2.1.2-3 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.1.2-3 to perform the listed functions.
67	2.1.02.15	15. Displays of the parameters identified in Table 2.1.2-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the RCS parameters in the MCR.	The displays identified in Table 2.1.2-3 can be retrieved in the MCR.

Table 2.1.2-5		
Component Name	**Tag No.**	**Component Location**
Steam Generator 1	RCS-MB-01	Containment
Steam Generator 2	RCS-MB-02	Containment
Reactor Coolant Pump 1A	RCS-MP-01A	Containment
Reactor Coolant Pump 1B	RCS-MP-01B	Containment
Reactor Coolant Pump 2A	RCS-MP-02A	Containment
Reactor Coolant Pump 2B	RCS-MP-02B	Containment
Pressurizer	RCS-MV-02	Containment
ADS Sparger A	PXS-MW-01A	Containment
ADS Sparger B	PXS-MW-01B	Containment

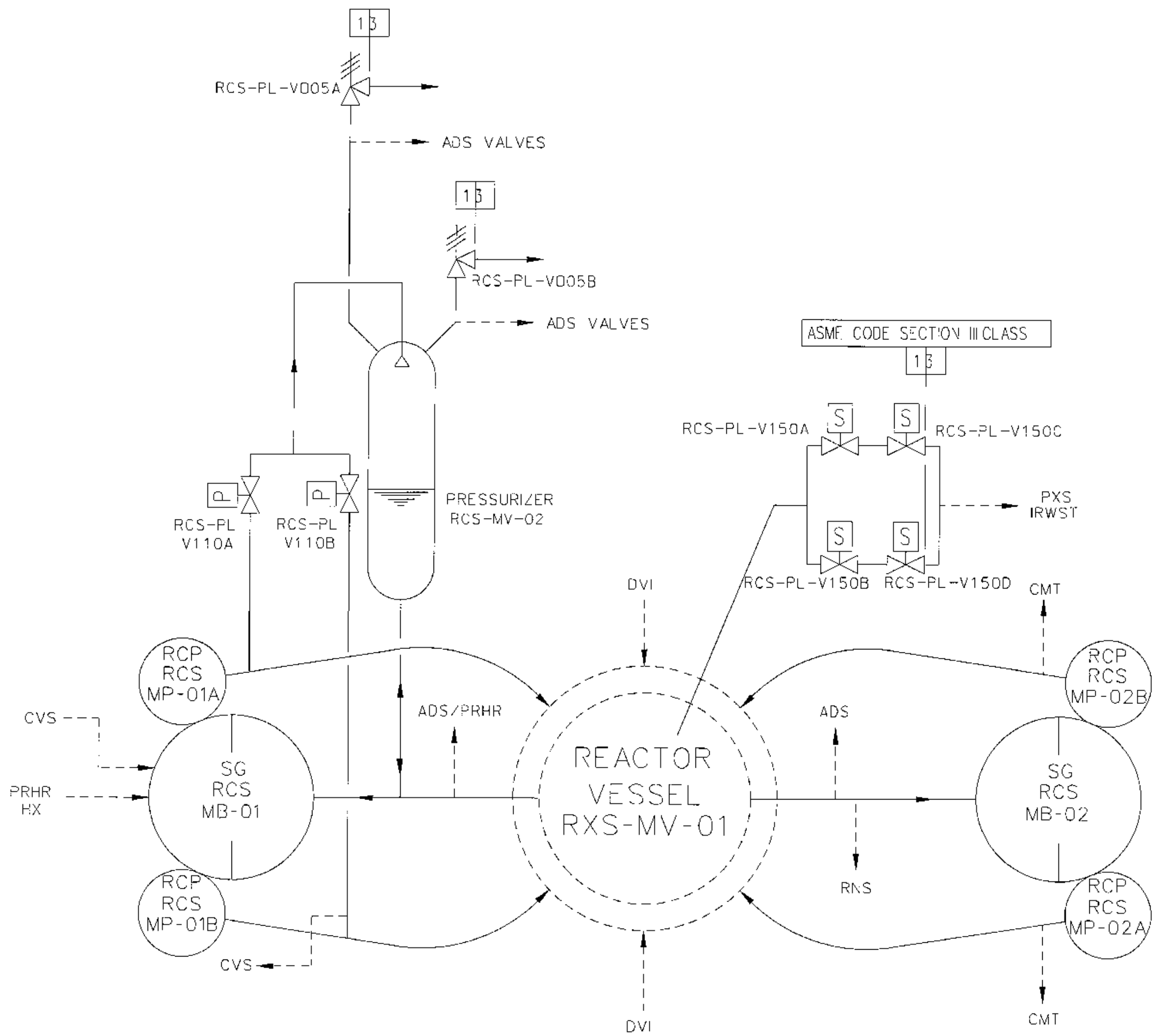


Figure 2.1.2-1 (Sheet 1 of 2)
Reactor Coolant System

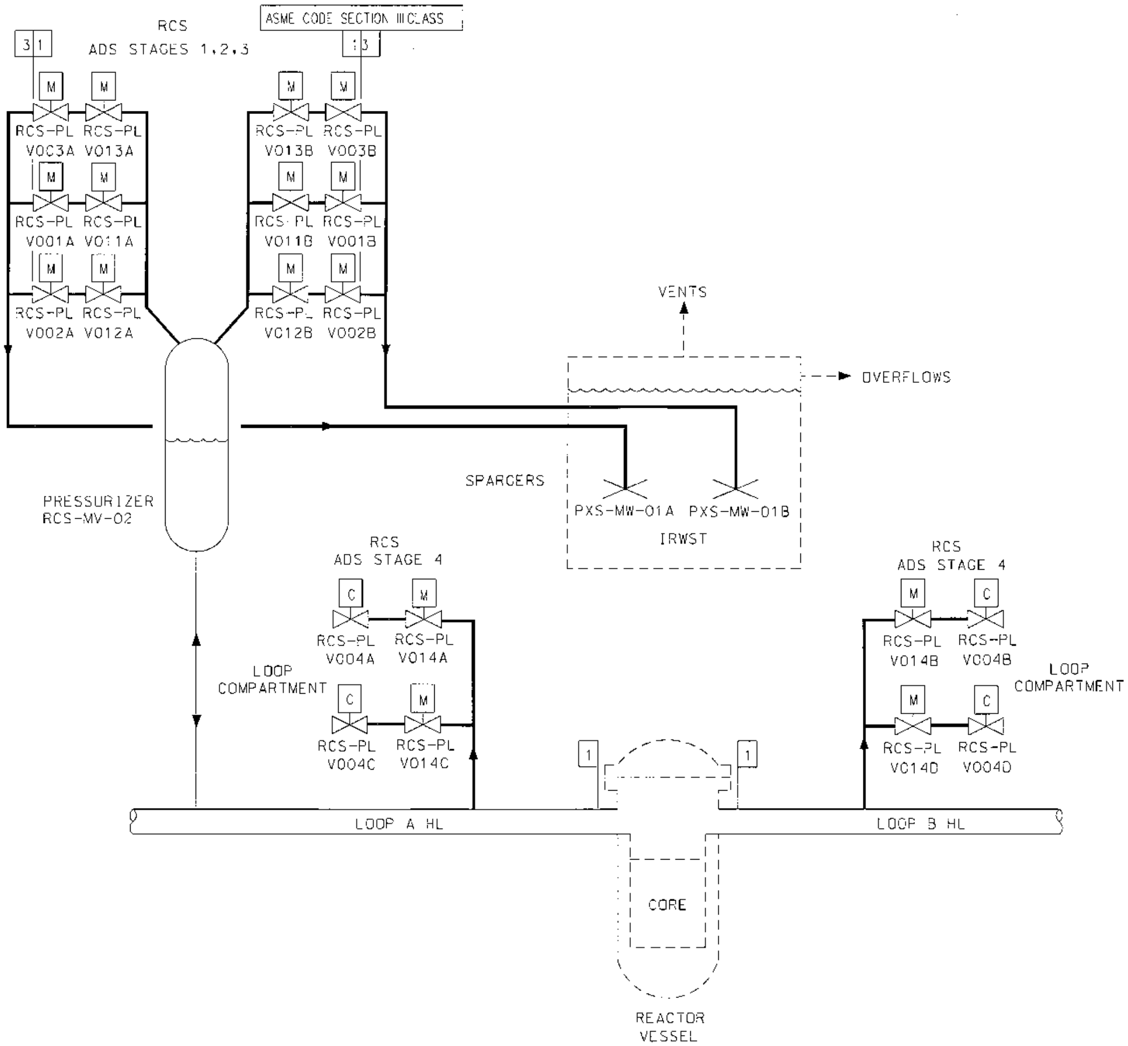


Figure 2.1.2-1 (Sheet 2 of 2)
Reactor Coolant System

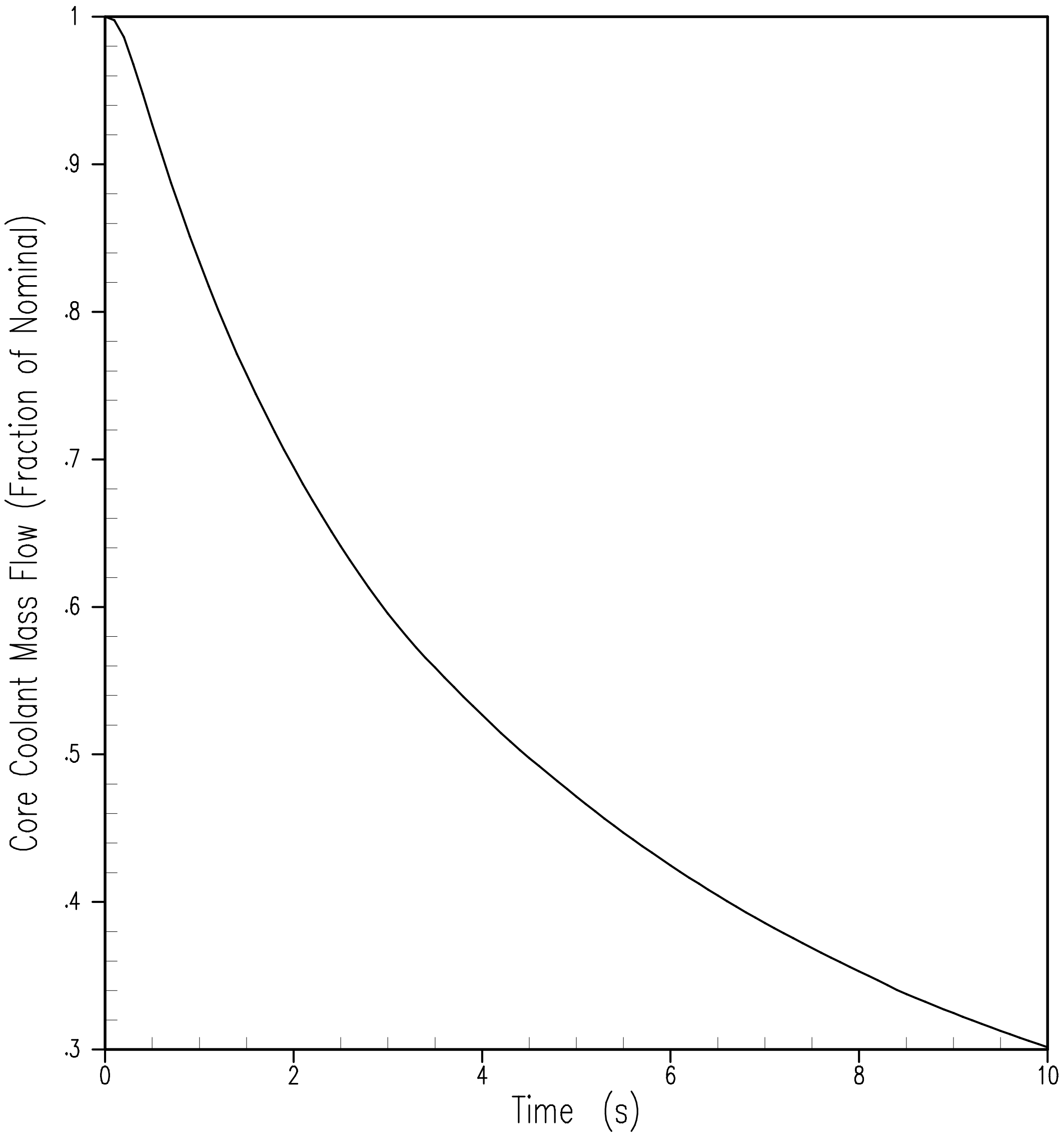


Figure 2.1.2-2
Flow Transient for Four Cold Legs
in Operation, Four Pumps Coasting Down

2.1.3 Reactor System

Design Description

The reactor system (RXS) generates heat by a controlled nuclear reaction and transfers the heat generated to the reactor coolant, provides a barrier that prevents the release of fission products to the atmosphere and a means to insert negative reactivity into the reactor core and to shutdown the reactor core.

The reactor core contains a matrix of fuel rods assembled into fuel assemblies using structural elements. Rod cluster control assemblies (RCCAs) are positioned and held within the fuel assemblies by control rod drive mechanisms (CRDMs). The CRDMs unlatch upon termination of electrical power to the CRDM thereby releasing the RCCAs. The fuel assemblies and RCCAs are designed in accordance with the principal design requirements.

The RXS is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the RXS are as shown in Table 2.1.3-3.

1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.
   a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.
   b) The rod cluster control and drive rod arrangement is as shown in Figure 2.1.3-2.
   c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.
2. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
3. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.
4. The pressure boundary components (reactor vessel [RV], control rod drive mechanisms [CRDMs], and incore instrument QuickLoc assemblies) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
5. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.
6. The reactor internals will withstand the effects of flow induced vibration.
7. The reactor vessel direct injection nozzle limits the blowdown of the reactor coolant system (RCS) following the break of a direct vessel injection line.

9. a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
   b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.

c. Separation is provided between RXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.
11. The reactor pressure vessel (RPV) beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.
12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the main control room (MCR).
13. The fuel assemblies and rod cluster control assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.
14. A top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.

Table 2.1.3-1					
Equipment Name	**Tag No.**	**ASME Code Section III Classification**	**Seismic Cat. I**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**
RV	RXS-MV-01	Yes	Yes	-	-
Reactor Upper Internals Assembly	RXS-MI-01	Yes	Yes	-	-
Reactor Lower Internals Assembly	RXS-MI-02	Yes	Yes	-	-
Fuel Assemblies (157 locations)	RXS-FA-A07/A08/A09/B05/B06/B07/B08/ B09/B10/B11/C04/C05/C06/C07/C08/C09/C10/ C11/C12/D03/D04/D05/D06/D07/D08/D09/ D10/D11/D12/D13/E02/E03/E04/E05/E06/E07/ E08/E09/E10/E11/E12/E13/E14/F02/F03/F04/ F05/F06/F07/F08/F09/F10/F11/F12/F13/F14/ G01/G02/G03/G04/G05/G06/G07/G08/G09/ G10/G11/G12/G13/G14/G15/H01/H02/H03/ H04/H05/H06/H07/H08/H09/H10/H11/H12/ H13/H14/H15/J01/J02/J03/J04/J05/J06/J07/J08/ J09/J10/J11/J12/J13/J14/J15/K02/K03/K04/ K05/K06/K07/K08/K09/K10/K11/K12/K13/ K14/L02/L03/L04/L05/L06/L07/L08/L09/L10/ L11/L12/L13/L14/M03/M04/M05/M06/M07/ M08/M09/M10/M11/M12/M13/N04/N05/N06/ N07/N08/N09/N10/N11/N12/P05/P06/P07/P08/ P09/P10/P11/ R07/R08/R09	No(1)	Yes	-	-

Note: Dash (-) indicates not applicable.

1. Fuel assemblies are designed using ASME Section III as a general guide.

Table 2.1.3-1 (cont.)					
Equipment Name	**Tag No.**	**ASME Code Section III Classification**	**Seismic Cat. I**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**
Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/C09/C11/D06/ D08/D10/E03/E05/E07/E09/E11/E13/F02/F04/ F12/F14/G03/G05/G07/G09/G11/G13/H04/ H08/H12/J03/J05/J07/J09/J11/J13/K02/K04/ K12/K14/L03/L05/L07/L09/L11/L13/M06/ M08/M10/N05/N07/N09/N11/P06/P10	No(1)	Yes	-	-
Gray Rod Cluster Assemblies (GRCAs) (16 locations)	RXS-FG-B08/D04/D12/F06/F08/F10/H02/H06/ H10/H14/K06/K08/K10/M04/M12/P08	No(1)	Yes	-	-
Control Rod Drive Mechanisms (CRDMs) (69 Locations)	RXS-MV-11B06/11B08/11B10/11C05/11C07/ 11C09/11C11/11D04/11D06/11D08/11D10/ 11D12/11E03/11E05/11E07/11E09/11E11/ 11E13/11F02/11F04/11F06/11F08/11F10/ 11F12/11F14/11G03/11G05/11G07/11G09/ 11G11/11G13/11H02/11H04/11H06/11H08/ 11H10/11H12/11H14/11J03/11J05/11J07/ 11J09/11J11/11J13/11K02/11K04/11K06/ 11K08/11K10/11K12/11K14/11L03/11L05/ 11L07/11L09/11L11/11L13/11M04/11M06/ 11M08/11M10/11M12/11N05/11N07/11N09/ 11N11/11P06/11P08/11P10	Yes	Yes	No/No	No
Incore Instrument QuickLoc Assemblies (8 Locations)	RXS-MY-Y11 through Y18	Yes	Yes	-	-

Note: Dash (-) indicates not applicable.

1. Fuel assemblies are designed using ASME Section III as a general guide.

Table 2.1.3-1 (cont.)					
Equipment Name	**Tag No.**	**ASME Code Section III Classification**	**Seismic Cat. I**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**
Source Range Detectors (4)	RXS-JE-NE001A/NE001B/NE001C/NE001D	-	Yes	Yes/Yes	No
Intermediate Range Detectors (4)	RXS-JE-NE002A/NE002B/NE002C/NE002D	-	Yes	Yes/Yes	Yes
Power Range Detectors – Lower (4)	RXS-JE-NE003A/NE003B/NE003C/NE003D	-	Yes	Yes/Yes	No
Power Range Detectors – Upper (4)	RXS-JE-NE004A/NE004B/NE004C/NE004D	-	Yes	Yes/Yes	No

Note: Dash (-) indicates not applicable.

Table 2.1.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
68	2.1.03.01	1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.	Inspection of the as-built system will be performed.	The as-built RXS conforms with the functional arrangement as described in the Design Description of this Section 2.1.3.
69	2.1.03.02a	2.a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the fuel assembly and control rod drive mechanism pattern shown in Figure 2.1.3-1.
70	2.1.03.02b	2.b) The control assemblies (rod cluster and gray rod) and drive rod arrangement is as shown in Figure 2.1.3-2.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the control assemblies (rod cluster and gray rod) and drive rod arrangement shown in Figure 2.1.3-2.
71	2.1.03.02c	2.c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the reactor vessel arrangement shown in Figure 2.1.3-3.
72	2.1.03.03	3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.1.3-1 as ASME Code Section III.
73	2.1.03.04	4. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
74	2.1.03.05	5. The pressure boundary components (RV, CRDMs, and incore instrument QuickLoc assemblies) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components of the RXS required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the pressure boundary components (RV, CRDMs, and incore instrument QuickLoc assemblies) conform with the requirements of the ASME Code Section III.

Table 2.1.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
75	2.1.03.06.i	6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island.
76	2.1.03.06.ii	6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
77	2.1.03.06.iii	6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
78	2.1.03.07.i	7. The reactor internals will withstand the effects of flow induced vibration.	i) A vibration type test will be conducted on the (first unit) reactor internals representative of AP1000.	i) A report exists and concludes that the (first unit) reactor internals have no observable damage or loose parts as a result of the vibration type test.
79	2.1.03.07.ii	7. The reactor internals will withstand the effects of flow induced vibration.	ii) A pre-test inspection, a flow test and a post-test inspection will be conducted on the as-built reactor internals.	ii) The as-built reactor internals have no observable damage or loose parts.
80	2.1.03.08	8. The reactor vessel direct vessel injection nozzle limits the blowdown of the RCS following the break of a direct vessel injection line.	An inspection will be conducted to verify the flow area of the flow limiting venturi within each direct vessel injection nozzle.	The throat area of the direct vessel injection line nozzle flow limiting venturi is less than or equal to 12.57 in^2.

Table 2.1.3-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
81	2.1.03.09a.i	9.a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analysis, or a combination of type tests and analysis will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
82	2.1.03.09a.ii	9.a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.1.3-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
83	2.1.03.09b	9.b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.	Testing will be performed by providing simulated test signals in each Class 1E division.	A simulated test signal exists for Class 1E equipment identified in Table 2.1.3-1 when the assigned Class 1E division is provided the test signal.
84	2.1.03.09c	9.c) Separation is provided between RXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
85	2.1.03.10	10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.	Inspection of the reactor lower internals assembly for the presence of capsules will be performed.	At least eight capsules are in the reactor lower internals assembly.
86	2.1.03.11	11. The RPV beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.	Manufacturing tests of the Charpy V-Notch specimen of the RPV beltline material will be performed.	A report exists and concludes that the initial RPV beltline Charpy upper-shelf energy is no less than 75 ft-lb.

Table 2.1.3-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
87	2.1.03.12	12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.1.3-1 can be retrieved in the MCR.
88	2.1.03.13	13. The fuel assemblies and rod cluster control assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the established design requirements.	An analysis is performed of the reactor core design.	A report exists and concludes that the fuel assemblies and rod cluster control assemblies intended for the initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.
89	2.1.03.14	14. A top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.	A preservice visual examination of the reactor vessel head top surface and penetration nozzles will be performed.	A report exists that documents the results of the top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle.

Table 2.1.3-3		
Component Name	**Tag No.**	**Component Location**
RV	RXS-MV-01	Containment
Reactor Upper Internals Assembly	RXS-MI-01	Containment
Reactor Lower Internals Assembly	RXS-MI-02	Containment
Fuel Assemblies (157 locations)	RXS-FA-A07/A08/A09/B05/ B06/B07/B08/B09/B10/B11/ C04/C05/C06/C07/C08/C09/ C10/C11/C12/D03/D04/D05/ D06/D07/D08/D09/D10/D11/ D12/D13/E02/E03/E04/E05/ E06/E07/E08/E09/E10/E11/E12/ E13/E14/F02/F03/F04/F05/F06/ F07/F08/F09/F10/F11/F12/F13/ F14/G01/G02/G03/G04/G05/ G06/G07/G08/G09/G10/G11/ G12/G13/G14/G15/H01/H02/ H03/H04/H05/H06/H07/H08/	Containment (located in auxiliary building prior to fuel loading)

Table 2.1.3-3		
Component Name	**Tag No.**	**Component Location**
	H09/H10/H11/H12/H13/H14/ H15/J01/J02/J03/J04/J05/J06/ J07/J08/J09/J10/J11/J12/J13/ J14/J15/K02/K03/K04/K05/ K06/K07/K08/K09/K10/K11/ K12/K13/K14/L02/L03/L04/ L05/L06/L07/L08/L09/L10/L11/ L12/L13/L14/M03/M04/M05/ M06/M07/M08/M09/M10/M11/ M12/M13/N04/N05/N06/N07/ N08/N09/N10/N11/N12/P05/ P06/P07/P08/P09/P10/P11/R07/ R08/R09	
Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/ C09/C11/D06/D08/D10/E03/ E05/E07/E09/E11/E13/F02/F04/ F12/F14/G03/G05/G07/G09/ G11/G13/H04/H08/H12/J03/ J05/J07/J09/J11/J13/K02/K04/ K12/K14/L03/L05/L07/L09/ L11/L13/M06/M08/M10/N05/ N07/N09/N11/P06/P10	Containment (located in auxiliary building prior to fuel loading)
Gray Rod Cluster Assemblies (GRCAs) (16 locations)	RXS-FG-B08/D04/D12/F06/ F08/F10/H02/H06/H10/H14/ K06/K08/K10/M04/M12/P08	Containment (located in auxiliary building prior to fuel loading)
Control Rod Drive Mechanisms (CRDMs) (69 Locations)	RXS-MV-11B06/11B08/ 11B10/11C05/11C07/11C09/ 11C11/11D04/11D06/11D08/ 11D10/11D12/11E03/11E05/ 11E07/11E09/11E11/11E13/ 11F02/11F04/11F06/11F08/ 11F10/11F12/11F14/11G03/ 11G05/11G07/11G09/11G11/ 11G13/11H02/11H04/11H06/ 11H08/11H10/11H12/11H14/ 11J03/11J05/11J07/11J09/11J11/ 11J13/11K02/11K04/11K06/ 11K08/11K10/11K12/11K14/ 11L03/11L05/11L07/11L09/ 11L11/11L13/11M04/11M06/ 11M08/11M10/11M12/11N05/ 11N07/11N09/11N11/11P06/ 11P08/11P10	Containment
Incore Instrument QuickLoc Assemblies (8 Locations)	RXS-MY-Y11 through Y18	Containment

Table 2.1.3-3		
Component Name	**Tag No.**	**Component Location**
Source Range Detectors (4)	RXS-JE-NE001A/NE001B/ NE001C/NE001D	Containment
Intermediate Range Detectors (4)	RXS-JE-NE002A/NE002B/ NE002C/NE002D	Containment
Power Range Detectors – Lower (4)	RXS-JE-NE003A/NE003B/ NE003C/NE003D	Containment
Power Range Detectors – Upper (4)	RXS-JE-NE004A/NE004B/ NE004C/NE004D	Containment

Table 2.1.3-4
Key Dimensions and Acceptable Variations of the Reactor Vessel and Internals
(Figure 2.1.3-2 and Figure 2.1.3-3)

Description	Dimension or Elevation (inches)	Nominal Value (inches)	Acceptable Variation (inches)
RV inside diameter at beltline (inside cladding)	A	159.0	+1.0/-1.0
RV wall thickness at beltline (without cladding)	B	8.4	+1.0/-0.12
RV wall thickness at bottom head (without cladding)	C	6.0	+1.0/-0.12
RV inlet nozzle inside diameter at safe end	D	22.0	+0.35/-0.10
RV outlet nozzle inside diameter at safe end	E	31.0	+0.35/-0.10
Elevation from RV mating surface to centerline of inlet nozzle	F	62.5	+0.25/-0.25
Elevation from RV mating surface to centerline of outlet nozzle	G	80.0	+0.25/-0.25
Elevation from RV mating surface to centerline of direct vessel injection nozzle	H	100.0	+0.25/-0.25
Elevation from RV mating surface to inside of RV bottom head (inside cladding)	I	397.59	+1.0/-0.50
Elevation from RV mating surface to top of lower core support plate	J	327.3	+0.50/-0.50
Separation distance between bottom of upper core plate and top of lower core support with RV head in place	K	189.8	+0.20/0.20

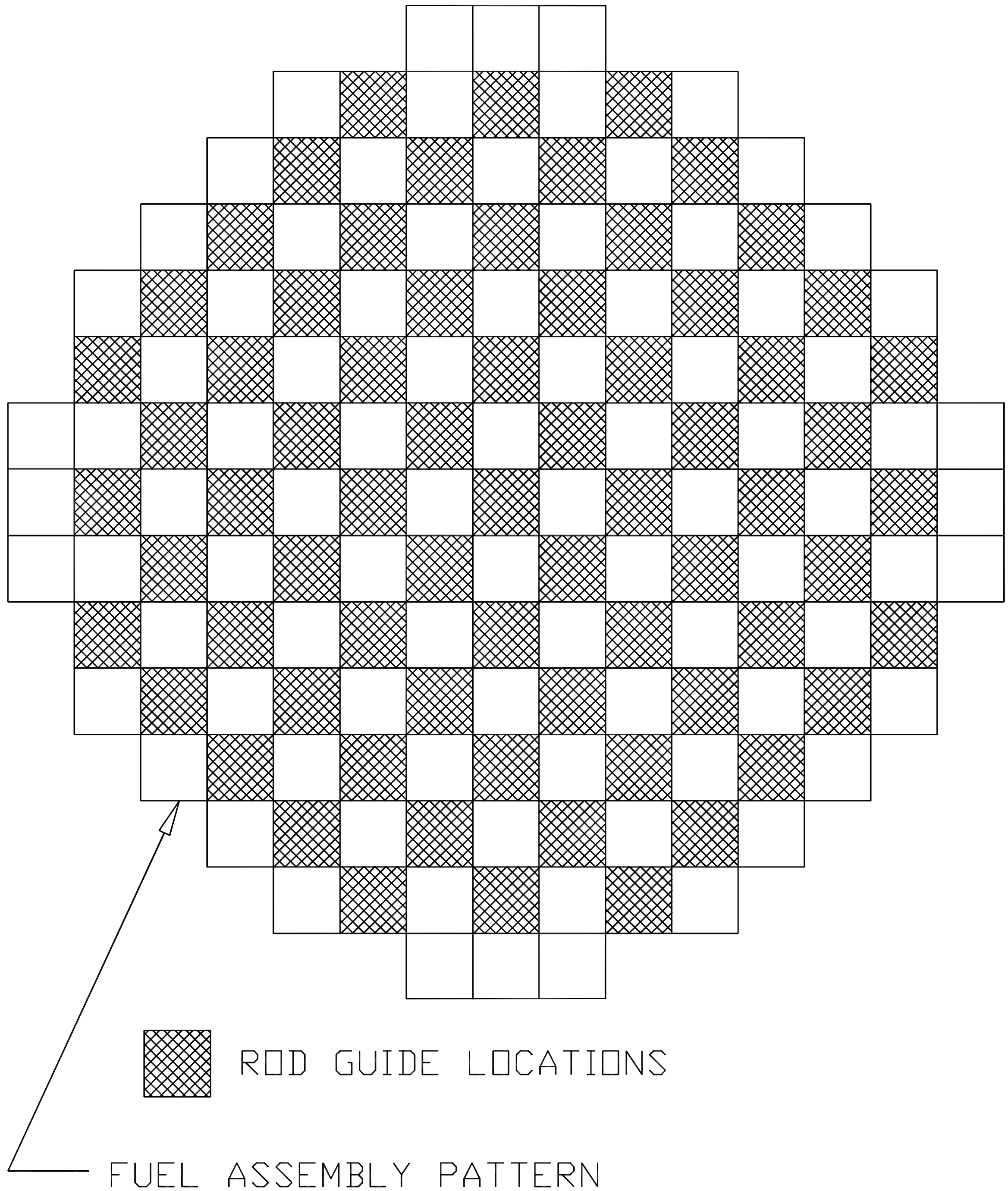


Figure 2.1.3-1
Reactor Upper Internals Rod Guide Arrangement

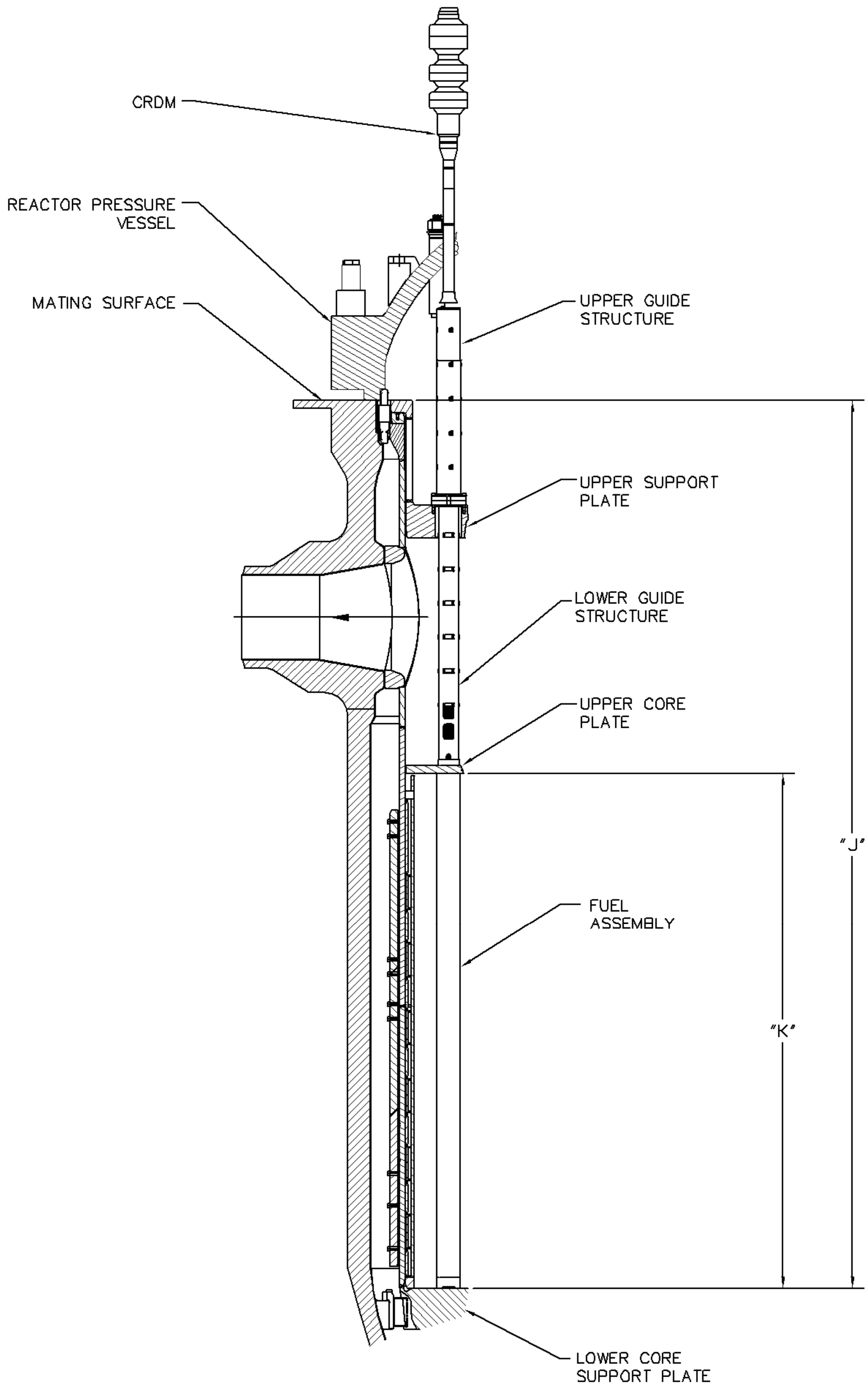


Figure 2.1.3-2
Rod Cluster Control and Drive Rod Arrangement

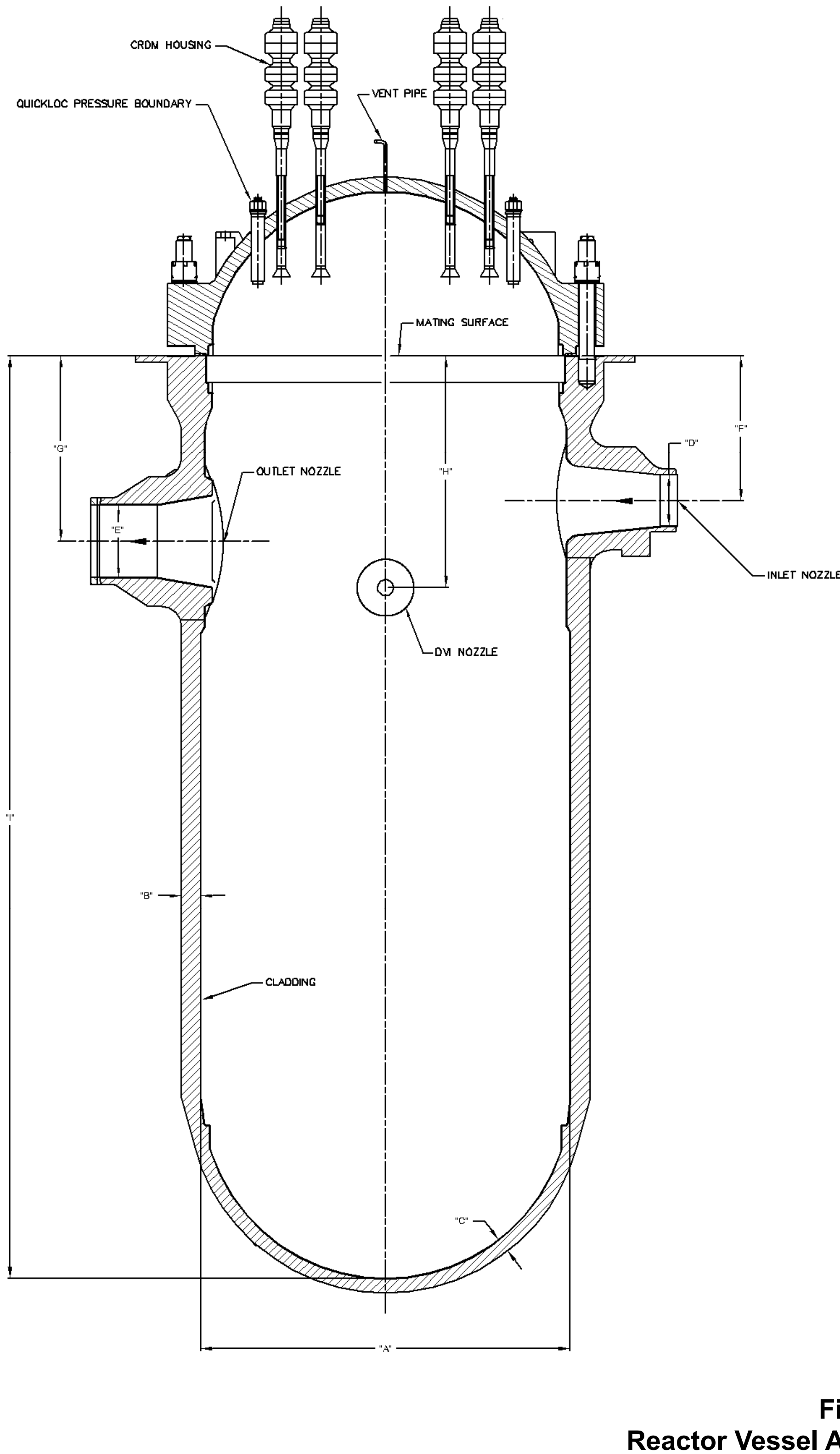


Figure 2.1.3-3
Reactor Vessel Arrangement

2.2 Nuclear Safety Systems

2.2.1 Containment System

Design Description

The containment system (CNS) is the collection of boundaries that separates the containment atmosphere from the outside environment during design basis accidents.

The CNS is as shown in Figure 2.2.1-1 and the component locations of the CNS are as shown in Table 2.2.1-4.

1. The functional arrangement of the CNS and associated systems is as described in the Design Description of this Section 2.2.1.
2. a) The components identified in Table 2.2.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.2.1-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.1-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.
5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.
6. a) The Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.1-1 are powered from their respective Class 1E division.

   c) Separation is provided between CNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

   d) The non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.

7. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.

8. Containment electrical penetration assemblies are protected against currents that are greater than the continuous ratings.

9. Safety-related displays identified in Table 2.2.1-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.1-1 to perform active functions.

    b) The valves identified in Table 2.2.1-1 as having protection and safety monitoring system (PMS) control perform an active function after receiving a signal from the PMS.

    c) The valves identified in Table 2.2.1-1 as having diverse actuation system (DAS) control perform an active function after receiving a signal from the DAS.

11. a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.1-1 assume the indicated loss of motive power position.

Table 2.2.1-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Service Air Supply Outside Containment Isolation Valve	CAS-PL-V204	Yes	Yes	No	-/-	No	-/-	None	-
Service Air Supply Inside Containment Isolation Check Valve	CAS-PL-V205	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
Instrument Air Supply Outside Containment Isolation Valve	CAS-PL-V014	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Instrument Air Supply Inside Containment Isolation Check Valve	CAS-PL-V015	Yes	Yes	No	-/-	-	-/-	Transfer Closed	-
Component Cooling Water System (CCS) Containment Isolation Motor-operated Valve (MOV) – Inlet Line Outside Reactor Containment (ORC)	CCS-PL-V200	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
CCS Containment Isolation Check Valve – Inlet Line Inside Reactor Containment (IRC)	CCS-PL-V201	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-

Note: Dash (-) indicates not applicable.

Table 2.2.1-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
CCS Containment Isolation MOV – Outlet Line IRC	CCS-PL-V207	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
CCS Containment Isolation MOV – Outlet Line ORC	CCS-PL-V208	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
CCS Containment Isolation Relief Valve – Outlet Line IRC	CCS-PL-V220	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Demineralized Water Supply Containment Isolation Valve ORC	DWS-PL-V244	Yes	Yes	No	-/-	No	-/-	None	-
Demineralized Water Supply Containment Isolation Check Valve IRC	DWS-PL-V245	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
Fuel Transfer Tube	FHS-FT-01	Yes	Yes	-	-/-	-	-/-	-	-
Fuel Transfer Tube Isolation Valve	FHS-PL-V001	Yes	Yes	-	-/-	-	-/-	Transfer Closed	-
Fire Water Containment Supply Isolation Valve – Outside	FPS-PL-V050	Yes	Yes	No	-/-	No	-/-	None	-
Fire Water Containment Isolation Supply Check Valve – Inside	FPS-PL-V052	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-

Table 2.2.1-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Spent Fuel Pool Cooling System (SFS) Discharge Line Containment Isolation Check Valve – IRC	SFS-PL-V037	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
SFS Discharge Line Containment Isolation MOV – ORC	SFS-PL-V038	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
SFS Suction Line Containment Isolation MOV – IRC	SFS-PL-V034	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
SFS Suction Line Containment Isolation MOV – ORC	SFS-PL-V035	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
SFS Suction Line Containment Isolation Relief Valve – IRC	SFS-PL-V067	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Containment Purge Inlet Containment Isolation Valve – ORC	VFS-PL-V003	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Containment Purge Inlet Containment Isolation Valve – IRC	VFS-PL-V004	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Integrated Leak Rate Testing Vent Discharge Containment Isolation Valve – ORC	VFS-PL-V008	Yes	Yes	No	-/-	No	-/-	None	-
Containment Purge Discharge Containment Isolation Valve – IRC	VFS-PL-V009	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Containment Purge Discharge Containment Isolation Valve – ORC	VFS-PL-V010	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed

Table 2.2.1-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Vacuum Relief Containment Isolation A MOV – ORC	VFS-PL-V800A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	As Is
Vacuum Relief Containment Isolation B MOV – ORC	VFS-PL-V800B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed/ Transfer Open	As Is
Vacuum Relief Containment Isolation Check Valve A – IRC	VFS-PL-V803A	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Vacuum Relief Containment Isolation Check Valve B – IRC	VFS-PL-V803B	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Fan Coolers Return Containment Isolation Valve – IRC	VWS-PL-V082	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Fan Coolers Return Containment Isolation Valve – ORC	VWS-PL-V086	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Fan Coolers Return Containment Isolation Relief Valve – IRC	VWS-PL-V080	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-

Table 2.2.1-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Fan Coolers Supply Containment Isolation Valve – ORC	VWS-PL-V058	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Fan Coolers Supply Containment Isolation Check Valve – IRC	VWS-PL-V062	Yes	Yes	No	-/-	No	-/-	Transfer Closed	-
Reactor Coolant Drain Tank (RCDT) Gas Outlet Containment Isolation Valve – IRC	WLS-PL-V067	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
RCDT Gas Outlet Containment Isolation Valve – ORC	WLS-PL-V068	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Sump Discharge Containment Isolation Valve – IRC	WLS-PL-V055	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Sump Discharge Containment Isolation Valve – ORC	WLS-PL-V057	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Closed	Closed
Sump Discharge Containment Isolation Relief Valve – IRC	WLS-PL-V058	Yes	Yes	No	-/-	No	-/-	Transfer Closed/ Transfer Open	-
Spare Penetration	CNS-PY-C01	Yes	Yes	-	-/-	-	-/-	-	-
Spare Penetration	CNS-PY-C02	Yes	Yes	-	-/-	-	-/-	-	-
Spare Penetration	CNS-PY-C03	Yes	Yes	-	-/-	-	-/-	-	-
Main Equipment Hatch	CNS-MY-Y01	Yes	Yes	-	-/-	-	-/-	-	-
Maintenance Hatch	CNS-MY-Y02	Yes	Yes	-	-/-	-	-/-	-	-

Table 2.2.1-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Personnel Hatch	CNS-MY-Y03	Yes	Yes	-	-/-	-	-/-	-	-
Personnel Hatch	CNS-MY-Y04	Yes	Yes	-	-/-	-	-/-	-	-
Containment Vessel	CNS-MV-01	Yes	Yes	-	-/-	-	-/-	-	-
Electrical Penetration P03	DAS-EY-P03Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P01	ECS-EY-P01X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P02	ECS-EY-P02X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P06	ECS-EY-P06Y	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P07	ECS-EY-P07X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P09	ECS-EY-P09W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P10	ECS-EY-P10W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P11	IDSA-EY-P11Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P12	IDSA-EY-P12Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P13	IDSA-EY-P13Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P14	IDSD-EY-P14Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P15	IDSD-EY-P15Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P16	IDSD-EY-P16Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P17	ECS-EY-P17X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P18	ECS-EY-P18X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P19	ECS-EY-P19Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P20	ECS-EY-P20Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P21	EDS-EY-P21Z	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P22	ECS-EY-P22X	Yes	Yes	-	No/Yes	-	-/-	-	-

Table 2.2.1-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Electrical Penetration P23	ECS-EY-P23X	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P24	ECS-EY-P24	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P25	ECS-EY-P25W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P26	ECS-EY-P26W	Yes	Yes	-	No/Yes	-	-/-	-	-
Electrical Penetration P27	IDSC-EY-P27Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P28	IDSC-EY-P28Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P29	IDSC-EY-P29Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P30	IDSB-EY-P30Z	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P31	IDSB-EY-P31Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Electrical Penetration P32	IDSB-EY-P32Y	Yes	Yes	-	Yes/Yes	-	-/-	-	-
Instrument Penetration P46	PCS-PY-C01	Yes	Yes	-	-/-	-	-/-	-	-
Instrument Penetration P47	PCS-PY-C02	Yes	Yes	-	-/-	-	-/-	-	-
Instrument Penetration P48	PCS-PY-C03	Yes	Yes	-	-/-	-	-/-	-	-
Instrument Penetration P49	PCS-PY-C04	Yes	Yes	-	-/-	-	-/-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.1-2		
Line Name	**Line Number**	**ASME Code Section III**
Instrument Air In	CAS-PL-L015	Yes
Service Air In	CAS-PL-L204	Yes
Component Cooling Water Supply to Containment	CCS-PL-L201	Yes
Component Cooling Water Outlet from Containment	CCS-PL-L207	Yes
Demineralized Water In	DWS-PL-L245, L230	Yes
Fire Protection Supply to Containment	FPS-PL-L107	Yes
Containment Atmosphere Return Line	PSS-PL-L038	Yes
Common Primary Sample Line A/B	PSS-PL-T005A/B	Yes
Containment Atmosphere Sample Line	PSS-PL-T031	Yes
Spent Fuel Pool Cooling Discharge	SFS-PL-L017	Yes
Spent Fuel Pool Cooling Suction from Containment	SFS-PL-L038	Yes
Containment Purge Inlet to Containment	VFS-PL-L104, L105, L106	Yes
Containment Purge Discharge from Containment	VFS-PL-L203, L204, L205, L800, L801A/B, L803, L804, L805A/B, L810A/B, L832	Yes
Fan Cooler Supply Line to Containment	VWS-PL-L032	Yes
Fan Cooler Return Line from Containment	VWS-PL-L055	Yes
RCDT Gas Out	WLS-PL-L022	Yes
Waste Sump Out	WLS-PL-L073	Yes

Table 2.2.1-3
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
90	2.2.01.01	1. The functional arrangement of the CNS and associated systems is as described in the Design Description of this Section 2.2.1.	Inspection of the as-built system will be performed.	The as-built CNS conforms with the functional arrangement as described in the Design Description of this Section 2.2.1.
91	2.2.01.02a	2.a) The components identified in Table 2.2.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.2.1-1 as ASME Code Section III.

Table 2.2.1-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
92	2.2.01.02b	2.b) The piping identified in Table 2.2.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.2.1-2 as ASME Code Section III.
93	2.2.01.03a	3.a) Pressure boundary welds in components identified in Table 2.2.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
94	2.2.01.03b	3.b) Pressure boundary welds in piping identified in Table 2.2.1-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
95	2.2.01.04a.i	4.a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	i) A hydrostatic or pressure test will be performed on the components required by the ASME Code Section III to be tested.	i) A report exists and concludes that the results of the pressure test of the components identified in Table 2.2.1-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
96	2.2.01.04a.ii	4.a) The components identified in Table 2.2.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	ii) Impact testing will be performed on the containment and pressure-retaining penetration materials in accordance with the ASME Code Section III, Subsection NE, to confirm the fracture toughness of the materials.	ii) A report exists and concludes that the containment and pressure-retaining penetration materials conform with fracture toughness requirements of the ASME Code Section III.
97	2.2.01.04b	4.b) The piping identified in Table 2.2.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic or pressure test will be performed on the piping required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the piping identified in Table 2.2.1-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
98	2.2.01.05.i	5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table	i) The seismic Category I equipment identified in Table 2.2.1-1 is located on the Nuclear Island.

Table 2.2.1-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
			2.2.1-1 are located on the Nuclear Island.	
99	2.2.01.05.ii	5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function.
100	2.2.01.05.iii	5. The seismic Category I equipment identified in Table 2.2.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
101	2.2.01.06a.i	6.a) The Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
102	2.2.01.06a.ii	6.a) The Class 1E equipment identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.2.1-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
103	2.2.01.06b	6.b) The Class 1E components identified in Table 2.2.1-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.2.1-1 when the assigned Class 1E division

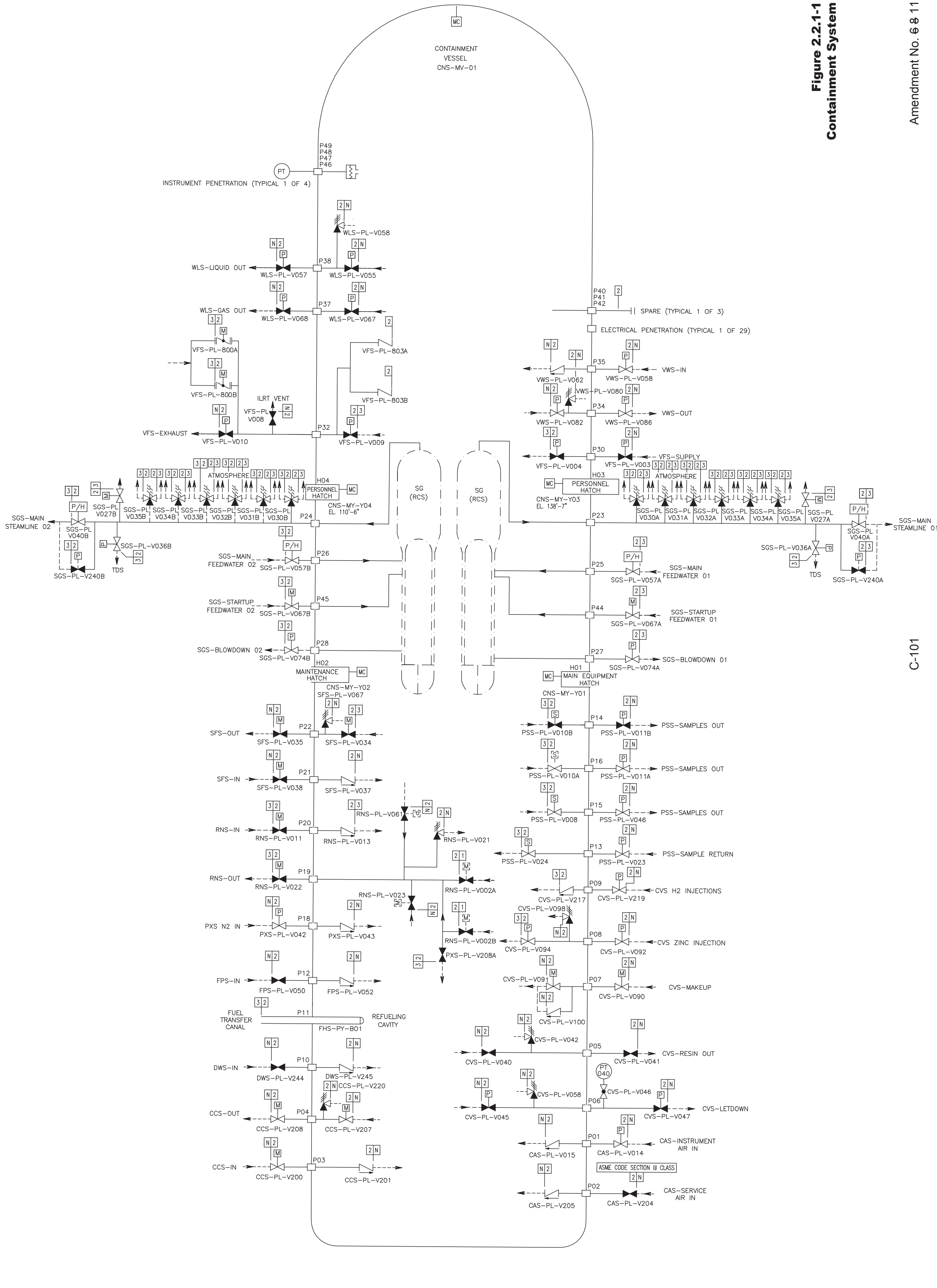
Table 2.2.1-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				assigned Class 1E division is provided the test signal.
104	2.2.01.06c	6.c) Separation is provided between CNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
105	2.2.01.06d.i	6.d) The non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on non-Class 1E electrical penetrations located in a harsh environment.	i) A report exists and concludes that the non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.
106	2.2.01.06d.ii	6.d) The non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of containment pressure boundary integrity.	ii) Inspection will be performed of the as-built non-Class 1E electrical penetrations located in a harsh environment.	ii) A report exists and concludes that the as-built non-Class 1E electrical penetrations identified in Table 2.2.1-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
107	2.2.01.07.i	7. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.	i) A containment integrated leak rate test will be performed.	i) The leakage rate from containment for the integrated leak rate test is less than L_a.
108	2.2.01.07.ii	7. The CNS provides the safety-related function of containment isolation for containment boundary integrity and provides a barrier against the release of fission products to the atmosphere.	ii) Testing will be performed to demonstrate that remotely operated containment isolation valves close within the required response times.	ii) The containment purge isolation valves (VFS-PL-V003, -V004, -V009, and -V010) close within 10 seconds, containment vacuum relief isolation valves (VFS-PL-V800A and -V800B) close within 30 seconds, SGS valves SGS-PL-V040A/B and SGS-PL-V057A/B are covered in subsection 2.2.4, Table 2.2.4-4 (item 11.b.ii)

Table 2.2.1-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				and all other containment isolation valves close within 60 seconds upon receipt of an actuation signal.
109	2.2.01.08	8. Containment electrical penetration assemblies are protected against currents that are greater than the continuous ratings.	An analysis for the as-built containment electrical penetration assemblies will be performed to demonstrate (1) that the maximum current of the circuits does not exceed the continuous rating of the containment electrical penetration assembly, or (2) that the circuits have redundant protection devices in series and that the redundant current protection devices are coordinated with the containment electrical penetration assembly's rated short circuit thermal capacity data and prevent current from exceeding the continuous current rating of the containment electrical penetration assembly.	Analysis exists for the as-built containment electrical penetration assemblies and concludes that the penetrations are protected against currents which are greater than their continuous ratings.
110	2.2.01.09	9. Safety-related displays identified in Table 2.2.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.1-1 can be retrieved in the MCR.
111	2.2.01.10a	10.a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.1-1 to perform active functions.	Stroke testing will be performed on remotely operated valves identified in Table 2.2.1-1 using the controls in the MCR.	Controls in the MCR operate to cause remotely operated valves identified in Table 2.2.1-1 to perform active safety functions.
112	2.2.01.10b	10.b) The valves identified in Table 2.2.1-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	Testing will be performed on remotely operated valves listed in Table 2.2.1-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.2.1-1 as having PMS control perform the active function identified in the table after receiving a signal from PMS.
113	2.2.01.10c	10.c) The valves identified in Table 2.2.1-1 as having DAS control perform an active safety function after receiving a signal	Testing will be performed on remotely operated valves listed in Table 2.2.1-1 using	The remotely operated valves identified in Table 2.2.1-1 as having

Table 2.2.1-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
		from DAS.	real or simulated signals into the DAS.	DAS control perform the active function identified in the table after receiving a signal from DAS.
114	2.2.01.11a.i	11.a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of each valve to operate under design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.2.1-1 under design conditions.
115	2.2.01.11a.ii	11.a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.
116	2.2.01.11a.iii	11.a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under preoperational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.2.1-1 under pre-operational test conditions.
117	2.2.01.11a.iv	11.a) The motor-operated and check valves identified in Table 2.2.1-1 perform an active safety-related function to change position as indicated in the table.	iv) Exercise testing of the check valves with active safety functions identified in Table 2.2.1-1 will be performed under preoperational test pressure, temperature and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.2.1-1.
118	2.2.01.11b	11.b) After loss of motive power, the remotely operated valves identified in Table 2.2.1-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.1-1 assumes the indicated loss of motive power position.

Table 2.2.1-4		
Component Name	**Tag No.**	**Component Location**
Containment Vessel	CNS-MV-01	Shield Building

Figure 2.2.1-1
Containment System



NOTE: PENETRATION CONNECTIONS; TEST, DRAIN OR VENT ARE NOT WITHIN THE SCOPE OF THIS FIGURE.

2.2.2 Passive Containment Cooling System

Design Description

The passive containment cooling system (PCS) removes heat from the containment during design basis events.

The PCS is as shown in Figure 2.2.2-1 and the component locations of the PCS are as shown in Table 2.2.2-4.

1. The functional arrangement of the PCS is as described in the Design Description of this Section 2.2.2.
2. a) The components identified in Table 2.2.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.2.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in the pipelines identified in Table 2.2.2-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.2.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
5. a) The seismic Category I components identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the pipelines identified in Table 2.2.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

   c) The passive containment cooling ancillary water storage tank (PCCAWST) can withstand a seismic event.
6. a) The Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.2-1 are powered from their respective Class 1E division.

   c) Separation is provided between PCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.
7. The PCS performs the following safety-related functions:

a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.

b) The PCS wets the outside surface of the containment vessel. The inside and outside of the containment vessel above the operating deck are coated with an inorganic zinc coating.

c) The PCS provides air flow over the outside of the containment vessel by a natural circulation air flow path from the air inlets to the air discharge structure.

d) The PCS drains the excess water from the outside of the containment vessel through the two upper annulus drains.

e) The PCS provides a flow path for long-term water makeup to the passive containment cooling water storage tank (PCCWST).

f) The PCS provides a flow path for long-term water makeup from the PCCWST to the spent fuel pool.

8. The PCS performs the following nonsafety-related functions:

   a) The PCCAWST contains an inventory of cooling water sufficient for PCS containment cooling from hour 72 through day 7.

   b) The PCS delivers water from the PCCAWST to the PCCWST and spent fuel pool simultaneously.

   c) The PCCWST includes a water inventory for the fire protection system.

9. Safety-related displays identified in Table 2.2.2-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.2-1 to perform active functions.

    b) The valves identified in Table 2.2.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

    c) The valves identified in Table 2.2.2-1 as having diverse actuation system (DAS) control perform an active safety function after receiving a signal from the DAS.

11. a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.2.2-1 assume the indicated loss of motive power position.

Table 2.2.2-1									
Component Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
PCCWST	PCS-MT-01	No	Yes	-	-	-	-	-	-
Water Distribution Bucket	PCS-MT-03	No	Yes	-	-	-	-	-	-
Water Distribution Wiers	PCS-MT-04	No	Yes	-	-	-	-	-	-
PCCWST Isolation Valve	PCS-PL-V001A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Open	Open
PCCWST Isolation Valve	PCS-PL-V001B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Open	Open
PCCWST Isolation Valve MOV	PCS-PL-V001C	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/Yes	Transfer Open	As Is
PCCWST Isolation Block MOV	PCS-PL-V002A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Open	As Is
PCCWST Isolation Block MOV	PCS-PL-V002B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Open	As Is
PCCWST Isolation Block MOV	PCS-PL-V002C	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Open	As Is
PCS Recirculation Return Isolation Valve	PCS-PL-V023	Yes	Yes	-	-/No	No	-	Transfer Close	-
PCCWST Supply to Fire Protection System Isolation Valve	PCS-PL-V005	Yes	Yes	-	-/No	No	-	Transfer Close	-

Note: Dash (-) indicates not applicable.

Table 2.2.2-1 (cont.)									
Component Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
PCS Makeup to SFS Isolation Valve	PCS-PL-V009	Yes	Yes	-	-/No	No	-	Transfer Open	-
Water Makeup Isolation Valve	PCS-PL-V044	Yes	Yes	-	-/No	No	-	Transfer Open	-
Water Bucket Makeup Line Drain Valve	PCS-PL-V015	Yes	Yes	-	-/No	No	-	Transfer Close	-
Water Bucket Makeup Line Isolation Valve	PCS-PL-V020	Yes	Yes	-	-/No	No	-	Transfer Open	-
PCCWST Long-Term Makeup Line Check Valve	PCS-PL-V039	Yes	Yes	-	-/No	No	-	Transfer Open	-
PCCWST Long-Term Makeup Drain Isolation	PCS-PL-V042	Yes	Yes	-	-/No	No	-	Transfer Close	-
PCS Discharge to SFS Pool Isolation Valve	PCS-PL-V045	Yes	Yes	-	-/No	No	-	Transfer Open	-
Recirc Header Discharge to PCCWST Isolation Valve	PCS-PL-V046	Yes	Yes	-	-/No	No	-	Transfer Close	-
PCCWST Drain Isolation Valve	PCS-PL-V049	Yes	Yes	-	-/No	No	-	Transfer Close	-
Recirc Header Discharge to SFS Pool Isolation Valve	PCS-PL-V050	Yes	Yes	-	-/No	No	-	Transfer Open/Clos e	-
PCCWST Discharge to SFS Pool Isolation Valve	PCS-PL-V051	Yes	Yes	-	-/No	No	-	Transfer Open/Clos e	-
PCS Water Delivery Flow Sensor	PCS-001	No	Yes	-	Yes/No	Yes	-	-	-

Table 2.2.2-1 (cont.)									
Component Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
PCS Water Delivery Flow Sensor	PCS-002	No	Yes	-	Yes/No	Yes	-	-	-
PCS Water Delivery Flow Sensor	PCS-003	No	Yes	-	Yes/No	Yes	-	-	-
PCS Water Delivery Flow Sensor	PCS-004	No	Yes	-	Yes/No	Yes	-	-	-
Containment Pressure Sensor	PCS-005	No	Yes	-	Yes/Yes	Yes	-	-	-
Containment Pressure Sensor	PCS-006	No	Yes	-	Yes/Yes	Yes	-	-	-
Containment Pressure Sensor	PCS-007	No	Yes	-	Yes/Yes	Yes	-	-	-
Containment Pressure Sensor	PCS-008	No	Yes	-	Yes/Yes	Yes	-	-	-
PCCWST Water Level Sensor	PCS-010	No	Yes	-	Yes/No	Yes	-	-	-
PCCWST Water Level Sensor	PCS-011	No	Yes	-	Yes/No	Yes	-	-	-
High-range Containment Pressure Sensor	PCS-012	No	Yes	-	Yes/Yes	Yes	-	-	-
High-range Containment Pressure Sensor	PCS-013	No	Yes	-	Yes/Yes	Yes	-	-	-
High-range Containment Pressure Sensor	PCS-014	No	Yes	-	Yes/Yes	Yes	-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.2-2			
Pipeline Name	**Line Number**	**ASME Code Section III**	**Functional Capability Required**
PCCWST Discharge Lines	PCS-PL-L001A/B/C/D	Yes	Yes
PCCWST Discharge Cross-connect Line	PCS-PL-L002	Yes	Yes
PCCWST Discharge Header Lines	PCS-PL-L003A/B PCS-PL-L005	Yes	Yes
Post-72-hour Supply Line Connection	PCS-PL-L051 PCS-PL-L054 PCS-PL-L065	Yes	Yes
Post-72-hour Containment Cooling Makeup From Supply Line Connections	PCS-PL-L004 PCS-PL-L007 PCS-PL-L008 PCS-PL-L023 PCS-PL-L050	Yes	Yes
Post-72-hour SFS Makeup From PCCWST	PCS-PL-L011 PCS-PL-L017 PCS-PL-L018 PCS-PL-L030* PCS-PL-L039* PCS-PL-L041 PCS-PL-L049* PCS-PL-L073	Yes	Yes
Post-72-hour SFS Makeup From Supply Line Connection	PCS-PL-L025 PCS-PL-L029 PCS-PL-L030* PCS-PL-L039* PCS-PL-L048 PCS-PL-L049* PCS-PL-L052	Yes	Yes
Note: * Lines PCS-PL-L049, L039, and L030 comprise a common makeup line from both sources.			

Table 2.2.2-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
119	2.2.02.01	1. The functional arrangement of the PCS is as described in the Design Description of this Section 2.2.2.	Inspection of the as-built system will be performed.	The as-built PCS conforms to the functional arrangement as described in the Design Description of this Section 2.2.2.

Table 2.2.2-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
120	2.2.02.02a	2.a) The components identified in Table 2.2.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.2.2-1 as ASME Code Section III.
121	2.2.02.02b	2.b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.2.2-2 as ASME Code Section III.
122	2.2.02.03a	3.a) Pressure boundary welds in components identified in Table 2.2.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
123	2.2.02.03b	3.b) Pressure boundary welds in the pipelines identified in Table 2.2.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
124	2.2.02.04a	4.a) The components identified in Table 2.2.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.2.2-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
125	2.2.02.04b	4.b) The pipelines identified in Table 2.2.2-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.2.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
126	2.2.02.05a.i	5.a) The seismic Category I components identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I components and valves identified in Table 2.2.2-1 are located on the Nuclear Island.	i) The seismic Category I components identified in Table 2.2.2-1 are located on the Nuclear Island.

Table 2.2.2-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
127	2.2.02.05a.ii	5.a) The seismic Category I components identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I components will be performed.	ii) A report exists and concludes that the seismic Category I components can withstand seismic design basis loads without loss of safety function.
128	2.2.02.05a.iii	5.a) The seismic Category I components identified in Table 2.2.2-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built components including anchorage are seismically bounded by the tested or analyzed conditions.	iii) The report exists and concludes that the as-built components including anchorage are seismically bounded by the tested or analyzed conditions.
129	2.2.02.05b	5.b) Each of the pipelines identified in Table 2.2.2-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report concluding that the as-built pipelines meet the requirements for functional capability.	A report exists and concludes that each of the as-built pipelines identified in Table 2.2.2-2 for which functional capability is required meets the requirements for functional capability.
130	2.2.02.05c	5.c) The PCCAWST can withstand a seismic event.	Inspection will be performed for the existence of a report verifying that the as-built PCCAWST and its anchorage are designed using seismic Category II methods and criteria.	A report exists and concludes that the as-built PCCAWST and its anchorage are designed using seismic Category II methods and criteria.
131	2.2.02.06a.i	6.a) The Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests or a combination of type tests and analyses will be performed on Class 1E components located in a harsh environment.	i) A report exists and concludes that the Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
132	2.2.02.06a.ii	6.a) The Class 1E components identified in Table 2.2.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E components and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E components and the associated wiring, cables, and terminations identified in Table 2.2.2-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

Table 2.2.2-3
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
133	2.2.02.06b	6.b) The Class 1E components identified in Table 2.2.2-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E components identified in Table 2.2.2-1 when the assigned Class 1E division is provided the test signal.
134	2.2.02.06c	6.c) Separation is provided between PCS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
135	2.2.02.07a.i	7.a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.	i) Testing will be performed to measure the PCCWST delivery rate from each one of the three parallel flow paths.	i) When tested, each one of the three flow paths delivers water at greater than or equal to: – 469.1 gpm at a PCCWST water level of 27.4 ft + 0.2, - 0.0 ft above the tank floor – 226.6 gpm when the PCCWST water level uncovers the first (i.e. tallest) standpipe – 176.3 gpm when the PCCWST water level uncovers the second tallest standpipe – 144.2 gpm when the PCCWST water level uncovers the third tallest standpipe
136	2.2.02.07a.ii	7.a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.	ii) Testing and or analysis will be performed to demonstrate the PCCWST inventory provides 72 hours of adequate water flow.	ii) When tested and/or analyzed with all flow paths delivering and an initial water level at 27.4 + 0.2, - 0.00 ft, the PCCWST water inventory provides greater than or equal to 72 hours of flow, and the flow rate at 72 hours is greater than or equal to 100.7 gpm.
137	2.2.02.07a.iii	7.a) The PCS delivers water from the PCCWST to the outside, top of the containment vessel.	iii) Inspection will be performed to determine the PCCWST standpipes elevations.	iii) The elevations of the standpipes above the tank floor are: – 16.8 ft ± 0.2 ft – 20.3 ft ± 0.2 ft – 24.1 ft ± 0.2 ft

Table 2.2.2-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
138	2.2.02.07b.i	7.b) The PCS wets the outside surface of the containment vessel. The inside and the outside of the containment vessel above the operating deck are coated with an inorganic zinc material.	i) Testing will be performed to measure the outside wetted surface of the containment vessel with one of the three parallel flow paths delivering water to the top of the containment vessel.	i) A report exists and concludes that when the water in the PCCWST uncovers the standpipes at the following levels, the water delivered by one of the three parallel flow paths to the containment shell provides coverage measured at the spring line that is equal to or greater than the stated coverages. - 24.1 ± 0.2 ft above the tank floor; at least 90% of the perimeter is wetted. - 20.3 ± 0.2 ft above the tank floor; at least 72.9% of the perimeter is wetted. - 16.8 ± 0.2 ft above the tank floor; at least 59.6% of the perimeter is wetted.
139	2.2.02.07b.ii	7.b) The PCS wets the outside surface of the containment vessel. The inside and the outside of the containment vessel above the operating deck are coated with an inorganic zinc material.	ii) Inspection of the containment vessel exterior coating will be conducted.	ii) A report exists and concludes that the containment vessel exterior surface is coated with an inorganic zinc coating above elevation 135'-3".
140	2.2.02.07b.iii	7.b) The PCS wets the outside surface of the containment vessel. The inside and the outside of the containment vessel above the operating deck are coated with an inorganic zinc material.	iii) Inspection of the containment vessel interior coating will be conducted.	iii) A report exists and concludes that the containment vessel interior surface is coated with an inorganic zinc coating above 7' above the operating deck.
141	2.2.02.07c	7.c) The PCS provides air flow over the outside of the containment vessel by a natural circulation air flow path from the air inlets to the air discharge structure.	Inspections of the air flow path segments will be performed.	Flow paths exist at each of the following locations: – Air inlets – Base of the outer annulus – Base of the inner annulus – Discharge structure
142	2.2.02.07d	7.d) The PCS drains the excess water from the outside of the containment vessel through the two upper annulus drains.	Testing will be performed to verify the upper annulus drain flow performance.	With a water level within the upper annulus 10" ± 1" above the annulus drain inlet, the flow rate through each drain is greater than or equal to 525 gpm.

Table 2.2.2-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
143	2.2.02.07e.i	7.e) The PCS provides a flow path for long-term water makeup to the PCCWST.	i) See item 1 in this table.	i) See item 1 in this table.
144	2.2.02.07e.ii	7.e) The PCS provides a flow path for long-term water makeup to the PCCWST.	ii) Testing will be performed to measure the delivery rate from the long-term makeup connection to the PCCWST.	ii) With a water supply connected to the PCS long-term makeup connection, each PCS recirculation pump delivers greater than or equal to 100 gpm when tested separately.
145	2.2.02.07f.i	7.f) The PCS provides a flow path for long-term water makeup from the PCCWST to the spent fuel pool.	i) Testing will be performed to measure the delivery rate from the PCCWST to the spent fuel pool.	i) With the PCCWST water level at 27.4 ft + 0.2, - 0.0 ft above the bottom of the tank, the flow path from the PCCWST to the spent fuel pool delivers greater than or equal to 118 gpm.
146	2.2.02.07f.ii	7.f) The PCS provides a flow path for long-term water makeup from the PCCWST to the spent fuel pool.	ii) Inspection of the PCCWST will be performed.	ii) The volume of the PCCWST is greater than 756,700 gallons.
147	2.2.02.08a	8.a) The PCCAWST contains an inventory of cooling water sufficient for PCS containment cooling from hour 72 through day 7.	Inspection of the PCCAWST will be performed.	The volume of the PCCAWST is greater than 780,000 gallons.
148	2.2.02.08b	8.b) The PCS delivers water from the PCCAWST to the PCCWST and spent fuel pool simultaneously.	Testing will be performed to measure the delivery rate from the PCCAWST to the PCCWST and spent fuel pool simultaneously.	With PCCAWST aligned to the suction of the recirculation pumps, each pump delivers greater than or equal to 100 gpm to the PCCWST and 35 gpm to the spent fuel pool simultaneously when each pump is tested separately.
149	2.2.02.08c	8.c) The PCCWST includes a water inventory for the fire protection system.	See ITAAC Table 2.3.4-2, items 1 and 2.	See ITAAC Table 2.3.4-2, items 1 and 2.
150	2.2.02.09	9. Safety-related displays identified in Table 2.2.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.2-1 can be retrieved in the MCR.
151	2.2.02.10a	10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.2-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.2.2-1 using the controls in the MCR.	Controls in the MCR operate to cause remotely operated valves identified in Table 2.2.2-1 to perform active functions.

Table 2.2.2-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
152	2.2.02.10b	10.b) The valves identified in Table 2.2.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	Testing will be performed on the remotely operated valves in Table 2.2.2-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.2.2-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
153	2.2.02.10c	10.c) The valves identified in Table 2.2.2-1 as having DAS control perform an active safety function after receiving a signal from the DAS.	Testing will be performed on the remotely operated valves listed in Table 2.2.2-1 using real or simulated signals into the DAS.	The remotely operated valves identified in Table 2.2.2-1 as having DAS control perform the active function identified in the table after receiving a signal from the DAS.
154	2.2.02.11a.i	11.a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.2.2-1 under design conditions.
155	2.2.02.11a.ii	11.a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the capability of the as-built motor-operated valves bound the tested conditions.	ii) A report exists and concludes that the capability of the as-built motor-operated valves bound the tested conditions.
156	2.2.02.11a.iii	11.a) The motor-operated valves identified in Table 2.2.2-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under preoperational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.2.2-1 under preoperational test conditions.
157	2.2.02.11b	11.b) After loss of motive power, the remotely operated valves identified in Table 2.2.2-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.2-1 assumes the indicated loss of motive power position.

Table 2.2.2-4		
Component Name	**Tag No.**	**Component Location**
PCCWST	PCS-MT-01	Shield Building
PCCAWST	PCS-MT-05	Yard
Recirculation Pump A	PCS-MP-01A	Auxiliary Building
Recirculation Pump B	PCS-MP-01B	Auxiliary Building

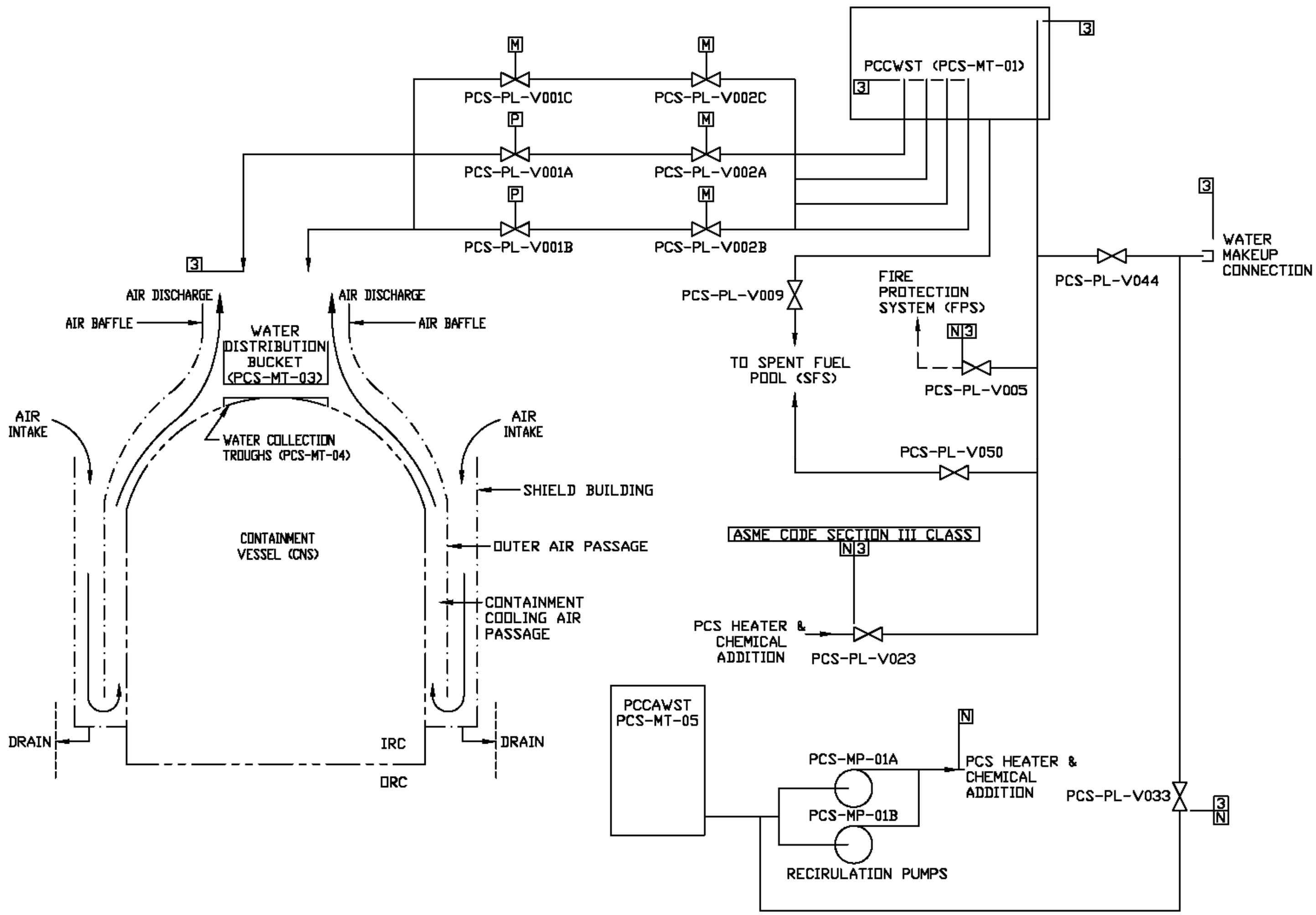


Figure 2.2.2-1
Passive Containment Cooling System

2.2.3 Passive Core Cooling System

Design Description

The passive core cooling system (PXS) provides emergency core cooling during design basis events.

The PXS is as shown in Figure 2.2.3-1 and the component locations of the PXS are as shown in Table 2.2.3-5.

1. The functional arrangement of the PXS is as described in the Design Description of this Section 2.2.3.
2. a) The components identified in Table 2.2.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.2.3-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.2.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.
5. a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.2.3-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.
6. Each of the as-built lines identified in Table 2.2.3-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.
7. a) The Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.2.3-1 are powered from their respective Class 1E division.

   c) Separation is provided between PXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.
8. The PXS provides the following safety-related functions:

a) The PXS provides containment isolation of the PXS lines penetrating the containment.

b) The PRHR HX provides core decay heat removal during design basis events.

c) The CMTs, accumulators, in-containment refueling water storage tank (IRWST) and containment recirculation provide reactor coolant system (RCS) makeup, boration, and safety injection during design basis events.

d) The PXS provides pH adjustment of water flooding the containment following design basis accidents.

9. The PXS has the following features:

a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.

b) The accumulator discharge check valves (PXS-PL-V028A/B and V029A/B) are of a different check valve type than the CMT discharge check valves (PXS-PL-V016A/B and V017A/B).

c) The equipment listed in Table 2.2.3-6 has sufficient thermal lag to withstand the effects of identified hydrogen burns associated with severe accidents.

10. Safety-related displays of the parameters identified in Table 2.2.3-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).

b) The valves identified in Table 2.2.3-1 as having protection and safety monitoring system (PMS) control perform their active function after receiving a signal from the PMS.

c) The valves identified in Table 2.2.3-1 as having diverse actuation system (DAS) control perform their active function after receiving a signal from the DAS.

12. a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.

b) After loss of motive power, the remotely operated valves identified in Table 2.2.3-1 assume the indicated loss of motive power position.

13. Displays of the parameters identified in Table 2.2.3-3 can be retrieved in the MCR.

Table 2.2.3-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank A	PXS-MT-01A	Yes	Yes	-	- / -	-	- / -	-	-
Accumulator Tank B	PXS-MT-01B	Yes	Yes	-	- / -	-	- / -	-	-
Core Makeup Tank (CMT) A	PXS-MT-02A	Yes	Yes	-	- / -	-	- / -	-	-
CMT B	PXS-MT-02B	Yes	Yes	-	- / -	-	- / -	-	-
IRWST	PXS-MT-03	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen A	PXS-MY-Y01A	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen B	PXS-MY-Y01B	No	Yes	-	- / -	-	- / -	-	-
IRWST Screen C	PXS-MY-Y01C	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen A	PXS-MY-Y02A	No	Yes	-	- / -	-	- / -	-	-
Containment Recirculation Screen B	PXS-MY-Y02B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3A	PXS-MY-Y03A	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 3B	PXS-MY-Y03B	No	Yes	-	- / -	-	- / -	-	-
pH Adjustment Basket 4A	PXS-MY-Y04A	No	Yes		- / -		- / -		
pH Adjustment Basket 4B	PXS-MY-Y04B	No	Yes		- / -		- / -		
Downspout Screen 1A	PXS-MY-Y81	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1B	PXS-MY-Y82	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1C	PXS-MY-Y83	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 1D	PXS-MY-Y84	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2A	PXS-MY-Y85	No	Yes	-	- / -	-	- / -	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
Downspout Screen 2B	PXS-MY-Y86	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2C	PXS-MY-Y87	No	Yes	-	- / -	-	- / -	-	-
Downspout Screen 2D	PXS-MY-Y88	No	Yes	-	- / -	-	- / -	-	-
CMT A Inlet Isolation Motor-operated Valve	PXS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is
CMT B Inlet Isolation Motor-operated Valve	PXS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/No	None	As Is

Table 2.2.3-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
CMT A Discharge Isolation Valve	PXS-PL-V014A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V014B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Isolation Valve	PXS-PL-V015A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT B Discharge Isolation Valve	PXS-PL-V015B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
CMT A Discharge Check Valve	PXS-PL-V016A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V016B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT A Discharge Check Valve	PXS-PL-V017A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
CMT B Discharge Check Valve	PXS-PL-V017B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Accumulator A Pressure Relief Valve	PXS-PL-V022A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-

Table 2.2.3-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
Accumulator B Pressure Relief Valve	PXS-PL-V022B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Accumulator A Discharge Isolation Valve	PXS-PL-V027A	Yes	Yes	Yes	- / -	No	- /No	None	As Is
Accumulator B Discharge Isolation Valve	PXS-PL-V027B	Yes	Yes	Yes	- / -	No	- /No	None	As Is
Accumulator A Discharge Check Valve	PXS-PL-V028A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator B Discharge Check Valve	PXS-PL-V028B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator A Discharge Check Valve	PXS-PL-V029A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Accumulator B Discharge Check Valve	PXS-PL-V029B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Close	-
Nitrogen Supply Containment Isolation Valve	PXS-PL-V042	Yes	Yes	Yes	Yes/No	Yes (position)	Yes/No	Transfer Closed	Close
Nitrogen Supply Containment Isolation Check Valve	PXS-PL-V043	Yes	Yes	No	- / -	No	- / -	Transfer Closed	-
PRHR HX Inlet Isolation Motor-operated Valve	PXS-PL-V101	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is

Table 2.2.3-1

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. Harsh Envir.	Safety-Related Display	Control PMS/ DAS	Active Function	Loss of Motive Power Position
PRHR HX Control Valve	PXS-PL-V108A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
PRHR HX Control Valve	PXS-PL-V108B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	Open
Containment Recirculation A Isolation Motor-operated Valve	PXS-PL-V117A	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is
Containment Recirculation B Isolation Motor-operated Valve	PXS-PL-V117B	Yes	Yes	Yes	Yes/Yes	Yes (position)	Yes/No	None	As Is
Containment Recirculation A Squib Valve	PXS-PL-V118A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation B Squib Valve	PXS-PL-V118B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation A Check Valve	PXS-PL-V119A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Containment Recirculation B Check Valve	PXS-PL-V119B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
Containment Recirculation A Squib Valve	PXS-PL-V120A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
Containment Recirculation B Squib Valve	PXS-PL-V120B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is

Table 2.2.3-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
B Squib Valve						(Position)	s		
IRWST Injection A Check Valve	PXS-PL-V122A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection B Check Valve	PXS-PL-V122B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection A Squib Valve	PXS-PL-V123A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection B Squib Valve	PXS-PL-V123B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection A Check Valve	PXS-PL-V124A	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection B Check Valve	PXS-PL-V124B	Yes	Yes	No	- / -	No	- / -	Transfer Open/ Transfer Closed	-
IRWST Injection A Squib Valve	PXS-PL-V125A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Injection B Squib Valve	PXS-PL-V125B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Open	As Is
IRWST Gutter Isolation Valve	PXS-PL-V130A	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Closed	Closed

Table 2.2.3-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
IRWST Gutter Isolation Valve	PXS-PL-V130B	Yes	Yes	Yes	Yes/Yes	Yes (Position)	Yes/Yes	Transfer Closed	Closed
CMT A Level Sensor	PXS-011A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011C	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT A Level Sensor	PXS-011D	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-012A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-012D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013A	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013B	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013C	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT A Level Sensor	PXS-013D	-	Yes	-	Yes/Yes	Yes	-/-	-	-
CMT B Level Sensor	PXS-014A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014C	-	Yes	-	Yes/Yes	Yes	- / -	-	-
CMT B Level Sensor	PXS-014D	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-045	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-046	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-047	-	Yes	-	Yes/Yes	Yes	- / -	-	-
IRWST Level Sensor	PXS-048	-	Yes	-	Yes/Yes	Yes	- / -	-	-

Table 2.2.3-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. Harsh Envir.**	**Safety-Related Display**	**Control PMS/ DAS**	**Active Function**	**Loss of Motive Power Position**
PRHR HX Flow Sensor	PXS-049A	-	Yes	-	Yes/Yes	Yes	- / -	-	-
PRHR HX Flow Sensor	PXS-049B	-	Yes	-	Yes/Yes	Yes	- / -	-	-
Containment Flood-up Level Sensor	PXS-050	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Containment Flood-up Level Sensor	PXS-051	-	Yes	-	Yes/Yes	Yes	-/-	-	-
Containment Flood-up Level Sensor	PXS-052	-	Yes	-	Yes/Yes	Yes	-/-	-	-
RNS Suction Leak Test Valve	PXS-PL-V208A	Yes	Yes	No	- / -	No	-/-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
PRHR HX inlet line from hot leg and outlet line to steam generator channel head	RCS-L134, PXS-L102, PXS-L103, PXS-L104A, PXS-L104B, PXS-L105, RCS-L113	Yes	Yes	Yes
	PXS-L107	Yes	Yes	No
CMT A inlet line from cold leg C and outlet line to reactor vessel direct vessel injection (DVI) nozzle A	RCS-L118A, PXS-L007A, PXS-L015A, PXS-L016A, PXS-L017A, PXS-L018A, PXS-L020A, PXS-L021A	Yes	Yes	Yes
	PXS-L070A	Yes	Yes	No
CMT B inlet line from cold leg D and outlet line to reactor vessel DVI nozzle B	RCS-L118B, PXS-L007B, PXS-L015B, PXS-L016B, PXS-L017B, PXS-L018B, PXS-L020B, PXS-L021B	Yes	Yes	Yes
	PXS-L070B	Yes	Yes	No
RNS A discharge line to PXS from RNS check valve RNS-PL-V017A to DVI line A	PXS-L019A	Yes	Yes	Yes
RNS B discharge line to PXS from RNS check valve RNS-PL-V017B to DVI line B	PXS-L019B	Yes	Yes	Yes
Accumulator A discharge line to DVI line A	PXS-L025A, PXS-L027A, PXS-L029A	Yes	Yes	Yes
Accumulator B discharge line to DVI line B	PXS-L025B, PXS-L027B, PXS-L029B	Yes	Yes	Yes
IRWST injection line A to DVI line A	PXS-L123A, PXS-L125A, PXS-L127A	Yes	Yes	Yes
	PXS-L124A, PXS-L118A, PXS-L117A, PXS-L116A, PXS-L112A	Yes	No	Yes
	PXS-L133A, PXS-L134A	Yes	Yes	No

Table 2.2.3-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
IRWST injection line B to DVI line B	PXS-L123B, PXS-L125B, PXS-L127B	Yes	Yes	Yes
	PXS-L124B, PXS-L118B, PXS-L117B, PXS-L116B, PXS-L114, PXS-L112B, PXS-L120	Yes	No	Yes
	PXS-L133B, PXS-L134B	Yes	Yes	No
IRWST screen cross-connect line	PXS-L180A, PXS-L180B	Yes	No	Yes
Containment recirculation line A	PXS-L113A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L100, PXS-L101, PXS-L106, PXS-L113B, PXS-L131B, PXS-L132B	Yes	No	Yes
IRWST gutter drain line	PXS-L142A, PXS-L142B	Yes	No	Yes
	PXS-L141A, PXS-L141B	Yes	No	No
Downspout drain lines from polar crane girder and internal stiffener to collection box A	PXS-L301A, PXS-L302A, PXS-L303A, PXS-L304A, PXS-L305A, PXS-L306A, PXS-L307A, PXS-L308A, PXS-L309A, PXS-L310A	Yes	No	Yes
Downspout drain lines from polar crane girder and internal stiffener to collection box B	PXS-L301B, PXS-L302B, PXS-L303B, PXS-L304B, PXS-L305B, PXS-L306B, PXS-L307B, PXS-L308B, PXS-L309B, PXS-L310B	Yes	No	Yes

Table 2.2.3-3			
Equipment	**Tag No.**	**Display**	**Control Function**
CMT A Discharge Isolation Valve (Position)	PXS-PL-V014A	Yes (Position)	-
CMT B Discharge Isolation Valve (Position)	PXS-PL-V014B	Yes (Position)	-
CMT A Discharge Isolation Valve (Position)	PXS-PL-V015A	Yes (Position)	-
CMT B Discharge Isolation Valve (Position)	PXS-PL-V015B	Yes (Position)	-
Accumulator A Nitrogen Vent Valve (Position)	PXS-PL-V021A	Yes (Position)	-
Accumulator B Nitrogen Vent Valve (Position)	PXS-PL-V021B	Yes (Position)	-
Accumulator A Discharge Isolation Valve (Position)	PXS-PL-V027A	Yes (Position)	-
Accumulator B Discharge Isolation Valve (Position)	PXS-PL-V027B	Yes (Position)	-
PRHR HX Control Valve (Position)	PXS-PL-V108A	Yes (Position)	-
PRHR HX Control Valve (Position)	PXS-PL-V108B	Yes (Position)	-
Containment Recirculation A Isolation Valve	PXS-PL-V117A	Yes (Position)	-
Containment Recirculation B Isolation Valve	PXS-PL-V117B	Yes (Position)	-
Containment Recirculation A Isolation Valve (Position)	PXS-PL-V118A	Yes (Position)	-
Containment Recirculation B Isolation Valve (Position)	PXS-PL-V118B	Yes (Position)	-
Containment Recirculation A Isolation Valve (Position)	PXS-PL-V120A	Yes (Position)	-
Containment Recirculation B Isolation Valve (Position)	PXS-PL-V120B	Yes (Position)	-
IRWST Line A Isolation Valve (Position)	PXS-PL-V121A	Yes (Position)	-
IRWST Line B Isolation Valve (Position)	PXS-PL-V121B	Yes (Position)	-
IRWST Injection A Isolation Squib (Position)	PXS-PL-V123A	Yes (Position)	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-3 (cont.)			
Equipment	**Tag No.**	**Display**	**Control Function**
IRWST Injection B Isolation Squib (Position)	PXS-PL-V123B	Yes (Position)	-
IRWST Injection A Isolation Squib (Position)	PXS-PL-V125A	Yes (Position)	-
IRWST Injection B Isolation Squib (Position)	PXS-PL-V125B	Yes (Position)	-
IRWST Gutter Bypass Isolation Valve (Position)	PXS-PL-V130A	Yes (Position)	-
IRWST Gutter Bypass Isolation Valve (Position)	PXS-PL-V130B	Yes (Position)	-
Accumulator A Level Sensor	PXS-JE-L021	Yes	-
Accumulator B Level Sensor	PXS-JE-L022	Yes	-
Accumulator A Level Sensor	PXS-JE-L023	Yes	-
Accumulator B Level Sensor	PXS-JE-L024	Yes	-
PRHR HX Inlet Temperature Sensor	PXS-JE-T064	Yes	-
IRWST Surface Temperature Sensor	PXS-JE-T041	Yes	-
IRWST Surface Temperature Sensor	PXS-JE-T042	Yes	-
IRWST Bottom Temperature Sensor	PXS-JE-T043	Yes	-
IRWST Bottom Temperature Sensor	PXS-JE-T044	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
158	2.2.03.01	1. The functional arrangement of the PXS is as described in the Design Description of this Section 2.2.3.	Inspection of the as-built system will be performed.	The as-built PXS conforms with the functional arrangement as described in the Design Description of this Section 2.2.3.
159	2.2.03.02a	2.a) The components identified in Table 2.2.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.2.3-1 as ASME Code Section III.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
160	2.2.03.02b	2.b) The piping identified in Table 2.2.3-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.2.3-2 as ASME Code Section III.
161	2.2.03.03a	3.a) Pressure boundary welds in components identified in Table 2.2.3-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
162	2.2.03.03b	3.b) Pressure boundary welds in piping identified in Table 2.2.3-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
163	2.2.03.04a	4.a) The components identified in Table 2.2.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.2.3-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
164	2.2.03.04b	4.b) The piping identified in Table 2.2.3-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.2.3-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
165	2.2.03.05a.i	5.a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.3-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.2.3-1 is located on the Nuclear Island.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
166	2.2.03.05a.ii	5.a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function. For the PXS containment recirculation and IRWST screens, a report exists and concludes that the screens can withstand seismic dynamic loads and also post-accident operating loads, including head loss and debris weights.
167	2.2.03.05a.iii	5.a) The seismic Category I equipment identified in Table 2.2.3-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions. For the PXS containment recirculation and IRWST screens, a report exists and concludes that the as-built screens including their anchorage are bounded by the seismic loads and also post-accident operating loads, including head loss and debris weights.
168	2.2.03.05b	5.b) Each of the lines identified in Table 2.2.3-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.2.3-2 for which functional capability is required meets the requirements for functional capability.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
169	2.2.03.06	6. Each of the as-built lines identified in Table 2.2.3-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from dynamic effects of a pipe break. Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built PXS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.
170	2.2.03.07a.i	7.a) The Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
171	2.2.03.07a.ii	7.a) The Class 1E equipment identified in Table 2.2.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.2.3-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
172	2.2.03.07b	7.b) The Class 1E components identified in Table 2.2.3-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.2.3-1 when the assigned Class 1E division is provided the test signal.
173	2.2.03.07c	7.c) Separation is provided between PXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
174	2.2.03.08a	8.a) The PXS provides containment isolation of the PXS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
175	2.2.03.08b.01	8.b) The PXS provides core decay heat removal during design basis events.	1. A heat removal performance test and analysis of the PRHR HX will be performed to determine the heat transfer from the HX. For the test, the reactor coolant hot leg temperature will be initially at ≥ 540°F with the reactor coolant pumps stopped. The IRWST water level for the test will be above the top of the HX. The IRWST water temperature is not specified for the test. The test will continue until the hot leg temperature decreases below 420°F.	1. A report exists and concludes that the PRHR HX heat transfer rate with the design basis number of PRHR HX tubes plugged is: $\geq 1.78 \times 10^{8}$ Btu/hr with 520°F HL Temp and 80°F IRWST temperatures. $\geq 1.11 \times 10^{8}$ Btu/hr with 420°F HL Temp and 80°F IRWST temperatures.
176	2.2.03.08b.02	8.b) The PXS provides core decay heat removal during design basis events.	2. Inspection of the elevation of the PRHR HX will be conducted.	2. The elevation of the centerline of the HX's upper channel head is greater than the HL centerline by at least 26.3 ft.
177	2.2.03.08c.i.01	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	i) A low-pressure injection test and analysis for each CMT, each accumulator, each IRWST injection line, and each containment recirculation line will be conducted. Each test is initiated by opening isolation valve(s) in the line being tested. Test fixtures may be used to simulate squib valves. 1. CMTs: Each CMT will be initially filled with water. All valves in these lines will be open during the test.	i) The injection line flow resistance from each source is as follows: 1. CMTs: The calculated flow resistance between each CMT and the reactor vessel is $\geq 1.81 \times 10^{-5}$ ft/gpm^2 and $\leq 2.25 \times 10^{-5}$ ft/gpm^2.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
178	2.2.03.08c.i.02	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	i) A low-pressure injection test and analysis for each CMT, each accumulator, each IRWST injection line, and each containment recirculation line will be conducted. Each test is initiated by opening isolation valve(s) in the line being tested. Test fixtures may be used to simulate squib valves. 2. Accumulators: Each accumulator will be partially filled with water and pressurized with nitrogen. All valves in these lines will be open during the test. Sufficient flow will be provided to fully open the check valves.	i) The injection line flow resistance from each source is as follows: 2. Accumulators: The calculated flow resistance between each accumulator and the reactor vessel is $\geq 1.47 \times 10^{-5}$ ft/gpm^2 and $\leq 1.83 \times 10^{-5}$ ft/gpm^2.
179	2.2.03.08c.i.03	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	i) A low-pressure injection test and analysis for each CMT, each accumulator, each IRWST injection line, and each containment recirculation line will be conducted. Each test is initiated by opening isolation valve(s) in the line being tested. Test fixtures may be used to simulate squib valves. 3. IRWST Injection: The IRWST will be partially filled with water. All valves in these lines will be open during the test. Sufficient flow will be provided to fully open the check valves.	i) The injection line flow resistance from each source is as follows: 3. IRWST Injection: The calculated flow resistance for each IRWST injection line between the IRWST and the reactor vessel is: Line A: $\geq 5.53 \times 10^{-6}$ ft/gpm^2 and $\leq 9.20 \times 10^{-6}$ ft/gpm^2 and Line B: $\geq 6.21 \times 10^{-6}$ ft/gpm^2 and $\leq 1.03 \times 10^{-5}$ ft/gpm^2.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
180	2.2.03.08c.i.04	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	i) A low-pressure injection test and analysis for each CMT, each accumulator, each IRWST injection line, and each containment recirculation line will be conducted. Each test is initiated by opening isolation valve(s) in the line being tested. Test fixtures may be used to simulate squib valves. 4. Containment Recirculation: A temporary water supply will be connected to the recirculation lines. All valves in these lines will be open during the test. Sufficient flow will be provided to fully open the check valves.	i) The injection line flow resistance from each source is as follows: 4. Containment Recirculation: The calculated flow resistance for each containment recirculation line between the containment and the reactor vessel is: Line A: $\leq 1.11 \times 10^{-5}$ ft/gpm^2 and Line B: $\leq 1.04 \times 10^{-5}$ ft/gpm^2.
181	2.2.03.08c.ii	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	ii) A low-pressure test and analysis will be conducted for each CMT to determine piping flow resistance from the cold leg to the CMT. The test will be performed by filling the CMT via the cold leg balance line by operating the normal residual heat removal pumps.	ii) The flow resistance from the cold leg to the CMT is $\leq 7.21 \times 10^{-6}$ ft/gpm^2.
182	2.2.03.08c.iii	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	iii) Inspections of the routing of the following pipe lines will be conducted: – CMT inlet line, cold leg to high point – PRHR HX inlet line, hot leg to high point	iii) These lines have no downward sloping sections between the connection to the RCS and the high point of the line.
183	2.2.03.08c.iv.01	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	iv) Inspections of the elevation of the following pipe lines will be conducted: 1. IRWST injection lines; IRWST connection to DVI nozzles	iv) The maximum elevation of the top inside surface of these lines is less than the elevation of: 1. IRWST bottom inside surface
184	2.2.03.08c.iv.02	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	iv) Inspections of the elevation of the following pipe lines will be conducted: 2. Containment recirculation lines; containment to IRWST lines	iv) The maximum elevation of the top inside surface of these lines is less than the elevation of: 2. IRWST bottom inside surface

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
185	2.2.03.08c.iv.03	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	iv) Inspections of the elevation of the following pipe lines will be conducted: 3. CMT discharge lines to DVI connection	iv) The maximum elevation of the top inside surface of these lines is less than the elevation of: 3. CMT bottom inside surface
186	2.2.03.08c.iv.04	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	iv) Inspections of the elevation of the following pipe lines will be conducted: 4. PRHR HX outlet line to SG connection	iv) The maximum elevation of the top inside surface of these lines is less than the elevation of: 4. PRHR HX lower channel head top inside surface
187	2.2.03.08c.v.01	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	v) Inspections of the elevation of the following tanks will be conducted: 1. CMTs	v) The elevation of the bottom inside tank surface is higher than the direct vessel injection nozzle centerline by the following: 1. CMTs ≥ 7.5 ft
188	2.2.03.08c.v.02	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	v) Inspections of the elevation of the following tanks will be conducted: 2. IRWST	v) The elevation of the bottom inside tank surface is higher than the direct vessel injection nozzle centerline by the following: 2. IRWST ≥ 3.4 ft
189	2.2.03.08c.vi.01	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	vi) Inspections of each of the following tanks will be conducted: 1. CMTs	vi) The calculated volume of each of the following tanks is as follows: 1. CMTs ≥ 2487 ft^3
190	2.2.03.08c.vi.02	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	vi) Inspections of each of the following tanks will be conducted: 2. Accumulators	vi) The calculated volume of each of the following tanks is as follows: 2. Accumulators ≥ 2000 ft^3
191	2.2.03.08c.vi.03	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	vi) Inspections of each of the following tanks will be conducted: 3.– IRWST	vi) The calculated volume of each of the following tanks is as follows: 3. IRWST ≥ 73,100 ft^3 between the tank outlet connection and the tank overflow

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
192	2.2.03.08c.vii	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	vii) Inspection of the as-built components will be conducted for the plate located above the containment recirculation screens.	vii) The plate located above the containment recirculation screens is no more than 1 ft, 3 in above the top of the face of the screens and extends at least 8 ft, 3 in perpendicular to the front and at least 7 ft to the side of the face of the screens.
193	2.2.03.08c.viii	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	viii) Inspections of the IRWST and containment recirculation screens will be conducted. The inspections will include measurements of the pockets and the number of pockets used in each screen. The pocket frontal face area is based on a width times a height. The width is the distance between pocket centerlines for pockets located beside each other. The height is the distance between pocket centerlines for pockets located above each other. The pocket screen area is the total area of perforated plate inside each pocket; this area will be determined by inspection of the screen manufacturing drawings.	viii) The screens utilize pockets with a frontal face area of $\geq$ 6.2 in^2 and a screen surface area $\geq$ 140 in^2 per pocket. IRWST Screens A and B each have a sufficient number of pockets to provide a frontal face area $\geq$ 25 ft^2, a screen surface area $\geq$ 575 ft^2, and a screen mesh size of $\leq$ 0.0625 inch. IRWST Screen C has a sufficient number of pockets to provide a frontal face area $\geq$ 50 ft^2, a screen surface area $\geq$ 1150 ft^2, and a screen mesh size $\leq$ 0.0625 inch. Each containment recirculation screen has a sufficient number of pockets to provide a frontal face area $\geq$ 105 ft^2, a screen surface area $\geq$ 2500 ft^2, and a screen mesh size $\leq$ 0.0625 inch. A debris curb exists in front of the containment recirculation screens which is $\geq$ 2 ft above the loop compartment floor. The bottoms of the IRWST screens are located $\geq$ 6 in above the bottom of the IRWST.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
194	2.2.03.08c.ix	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	ix) Inspections will be conducted of the insulation used inside the containment on the ASME Class 1 lines, reactor vessel, reactor coolant pumps, pressurizer and steam generators.	ix) The type of insulation used on these lines and equipment is a metal reflective type or a suitable equivalent. If an insulation other than metal reflective insulation is used, a report must exist and conclude that the insulation is a suitable equivalent.
			Inspections will be conducted of other insulation used inside the containment within the zone of influence (ZOI).	The type of insulation used on these lines and equipment is a metal reflective type or a suitable equivalent. If an insulation other than metal reflective insulation is used, a report must exist and conclude that the insulation is a suitable equivalent.
			Inspection will be conducted of other insulation below the maximum flood level of a design basis loss-of-coolant accident (LOCA).	The type of insulation used on these lines is metal reflective insulation, jacketed fiberglass, or a suitable equivalent. If an insulation other than metal reflective or jacketed fiberglass insulation is used, a report must exist and conclude that the insulation is a suitable equivalent.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
195	2.2.03.08c.x	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	x) Inspections will be conducted of the as-built nonsafety-related coatings or of plant records of the nonsafety-related coatings used inside containment on walls, floors, ceilings, and structural steel except in the CVS room. Inspections will be conducted of the as-built non-safety-related coatings or of plant records of the non-safety-related coatings used on components below the maximum flood level of a design basis LOCA or located above the maximum flood level and not inside cabinets or enclosures.	x) A report exists and concludes that the coatings used on these surfaces have a dry film density of ≥ 100 lb/ft^3. If a coating is used that has a lower dry film density, a report must exist and conclude that the coating will not transport. A report exists and concludes that inorganic zinc coatings used on these surfaces are Safety – Service Level I.
			Inspections will be conducted on caulking, tags, and signs used inside containment below the maximum flood level of a design basis LOCA or located above the maximum flood level and not inside cabinets or enclosures.	A report exists and concludes that tags and signs used in these locations are made of steel or another metal with a density ≥ 100 lb/ft^3. In addition, a report exists and concludes that caulking used in these locations or coatings used on these signs or tags have a dry film density of ≥ 100 lb/ft^3. If a material is used that has a lower density, a report must exist and conclude that there is insufficient water flow to transport lightweight caulking, signs, or tags.
			Inspections will be conducted of ventilation filters and fiber-producing fire barriers used inside containment within the ZOI or below the maximum flood level of a design basis LOCA.	A report exists and concludes that the ventilation filters and fire barriers in these locations have a density of ≥ 100 lb/ft^3.
196	2.2.03.08c.xi	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	xi) Inspection of the as-built CMT inlet diffuser will be conducted.	xi) The CMT inlet diffuser has a flow area ≥ 165 in^2.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
197	2.2.03.08c.xii	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	xii) Inspections will be conducted of the CMT level sensors (PXS-11A/B/D/C, -12A/B/C/D, - 13A/B/C/D, -14A/B/C/D) upper level tap lines.	xii) Each upper level tap line has a downward slope of ≥ 2.4 degrees from the centerline of the connection to the CMT to the centerline of the connection to the standpipe.
198	2.2.03.08c.xiii	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	xiii) Inspections will be conducted of the surfaces in the vicinity of the containment recirculation screens. The surfaces in the vicinity of the containment recirculation screens are the surfaces located above the bottom of the recirculation screens up to and including the bottom surface of the plate discussed in Table 2.2.3-4, item 8.c.vii, out at least 8 ft, 3 in perpendicular to the front and at least 7 feet to the side of the face of the screens.	xiii) These surfaces are stainless steel.
199	2.2.03.08c.xiv	8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	xiv) Inspection will be conducted of the excore (source range, intermediate range, and power range) detectors.	xiv) A report exists and concludes that the aluminum surfaces of the excore detectors are encased in a watertight stainless steel or titanium housing.
200	2.2.03.08d	8.d) The PXS provides pH adjustment of water flooding the containment following design basis accidents.	Inspections of the pH adjustment baskets will be conducted.	pH adjustment baskets exist, with a total calculated volume ≥ 560 ft^3. The pH baskets are located below plant elevation 107 ft, 2 in.
201	2.2.03.09a.i	9.a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.	i) A flow test and analysis for each IRWST drain line to the containment will be conducted. The test is initiated by opening isolation valves in each line. Test fixtures may be used to simulate squib valves.	i) The calculated flow resistance for each IRWST drain line between the IRWST and the containment is ≤ 4.07×10^{-6} ft/gpm^2.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
202	2.2.03.09a.ii	9.a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.	ii) Inspections of the as-built reactor vessel insulation will be performed.	ii) The combined total flow area of the water inlets is not less than 6 ft^2. The combined total flow area of the steam outlet(s) is not less than 12 ft^2. A report exists and concludes that the minimum flow area between the vessel insulation and reactor vessel for the flow path that vents steam is not less than 12 ft^2 considering the maximum deflection of the vessel insulation with a static pressure of 12.95 ft of water.
203	2.2.03.09a.iii	9.a) The PXS provides a function to cool the outside of the reactor vessel during a severe accident.	iii) Inspections will be conducted of the flow path(s) from the loop compartments to the reactor vessel cavity.	iii) A flow path with a flow area not less than 6 ft^2 exists from the loop compartment to the reactor vessel cavity.
204	2.2.03.09b	9.b) The accumulator discharge check valves (PXS-PL-V028A/B and V029A/B) are of a different check valve type than the CMT discharge check valves (PXS-PL-V016A/B and V017A/B).	An inspection of the accumulator and CMT discharge check valves is performed.	The accumulator discharge check valves are of a different check valve type than the CMT discharge check valves.
205	2.2.03.09c	9.c) The equipment listed in Table 2.2.3-6 has sufficient thermal lag to withstand the effects of identified hydrogen burns associated with severe accidents.	Type tests, analyses, or a combination of type tests and analyses will be performed to determine the thermal lag of this equipment.	A report exists and concludes that the thermal lag of this equipment is greater than the value required.
206	2.2.03.10	10. Safety-related displays of the parameters identified in Table 2.2.3-1 can be retrieved in the MCR.	Inspection will be performed for the retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.3-1 can be retrieved in the MCR.
207	2.2.03.11a.i	11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).	i) Testing will be performed on the squib valves identified in Table 2.2.3-1 using controls in the MCR, without stroking the valve.	i) Controls in the MCR operate to cause a signal at the squib valve electrical leads that is capable of actuating the squib valve.
208	2.2.03.11a.ii	11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.3-1 to perform their active function(s).	ii) Stroke testing will be performed on remotely operated valves other than squib valves identified in Table 2.2.3-1 using the controls in the MCR.	ii) Controls in the MCR operate to cause remotely operated valves other than squib valves to perform their active functions.

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
209	2.2.03.11b.i	11.b) The valves identified in Table 2.2.3-1 as having PMS control perform their active function after receiving a signal from the PMS.	i) Testing will be performed on the squib valves identified in Table 2.2.3-1 using real or simulated signals into the PMS without stroking the valve.	i) Squib valves receive an electrical signal at the valve electrical leads that is capable of actuating the valve after a signal is input to the PMS.
210	2.2.03.11b.ii	11.b) The valves identified in Table 2.2.3-1 as having PMS control perform their active function after receiving a signal from the PMS.	ii) Testing will be performed on the remotely operated valves other than squib valves identified in Table 2.2.3-1 using real or simulated signals into the PMS.	ii) Remotely operated valves other than squib valves perform the active function identified in the table after a signal is input to the PMS.
211	2.2.03.11b.iii	11.b) The valves identified in Table 2.2.3-1 as having PMS control perform their active function after receiving a signal from the PMS.	iii) Testing will be performed to demonstrate that remotely operated PXS isolation valves PXS-V014A/B, V015A/B, V108A/B open within the required response times.	iii) These valves open within 20 seconds after receipt of an actuation signal.
212	2.2.03.11c.i	11.c) The valves identified in Table 2.2.3-1 as having DAS control perform their active function after receiving a signal from the DAS.	i) Testing will be performed on the squib valves identified in Table 2.2.3-1 using real or simulated signals into the DAS without stroking the valve.	i) Squib valves receive an electrical signal at the valve electrical leads that is capable of actuating the valve after a signal is input to the DAS.
213	2.2.03.11c.ii	11.c) The valves identified in Table 2.2.3-1 as having DAS control perform their active function after receiving a signal from the DAS.	ii) Testing will be performed on the remotely operated valves other than squib valves identified in Table 2.2.3-1 using real or simulated signals into the DAS.	ii) Remotely operated valves other than squib valves perform the active function identified in Table 2.2.3-1 after a signal is input to the DAS.
214	2.2.03.12a.i	12.a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of squib valves will be performed that demonstrate the capability of the valve to operate under its design condition.	i) A test report exists and concludes that each squib valve changes position as indicated in Table 2.2.3-1 under design conditions.
215	2.2.03.12a.ii	12.a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the as-built squib valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built squib valves are bounded by the tests or type tests.
216	2.2.03.12a.iv	12.a) The squib valves and check valves identified in Table 2.2.3-1 perform an active safety-related function to change position as indicated in the table.	iv) Exercise testing of the check valves with active safety functions identified in Table 2.2.3-1 will be performed under preoperational test pressure, temperature, and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.2.3-1

Table 2.2.3-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
217	2.2.03.12b	12.b) After loss of motive power, the remotely operated valves identified in Table 2.2.3-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.3-1 assumes the indicated loss of motive power position.
218	2.2.03.13	13. Displays of the parameters identified in Table 2.2.3-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.2.3-3 in the MCR.	Displays identified in Table 2.2.3-3 can be retrieved in the MCR.

Table 2.2.3-5		
Component Name	**Tag No.**	**Component Location**
Passive Residual Heat Removal Heat Exchanger (PRHR HX)	PXS-ME-01	Containment Building
Accumulator Tank A	PXS-MT-01A	Containment Building
Accumulator Tank B	PXS-MT-01B	Containment Building
Core Makeup Tank (CMT) A	PXS-MT-02A	Containment Building
CMT B	PXS-MT-02B	Containment Building
IRWST	PXS-MT-03	Containment Building
IRWST Screen A	PXS-MY-Y01A	Containment Building
IRWST Screen B	PXS-MY-Y01B	Containment Building
IRWST Screen C	PXS-MY-Y01C	Containment Building
Containment Recirculation Screen A	PXS-MY-Y02A	Containment Building
Containment Recirculation Screen B	PXS-MY-Y02B	Containment Building
pH Adjustment Basket 3A	PXS-MY-Y03A	Containment Building
pH Adjustment Basket 3B	PXS-MY-Y03B	Containment Building
pH Adjustment Basket 4A	PXS-MY-Y04A	Containment Building
pH Adjustment Basket 4B	PXS-MY-Y04B	Containment Building

Table 2.2.3-6		
Equipment	**Tag No.**	**Function**
Hot Leg Sample Isolation Valves	PSS-PL-V001A/B	Transfer open
Liquid Sample Line Containment Isolation Valves IRC	PSS-PL-V010A/B	Transfer open
Containment Pressure Sensors	PCS-012, 013, 014	Sense pressure
RCS Wide Range Pressure Sensors	RCS-140A, B, C, D	Sense pressure
SG1 Wide Range Level Sensors	SGS-011, 012, 015, 016	Sense level
SG2 Wide Range Level Sensors	SGS-013, 014, 017, 018	Sense level
Hydrogen Monitors	VLS-001, 002, 003	Sense concentration
Hydrogen Igniters	VLS-EH-01 through 66	Ignite hydrogen
Containment Electrical Penetrations	P01, P02, P03, P06, P07, P09, P10, P11, P12, P13, P14, P15, P16, P17, P18, P19, P20, P21, P22, P23, P24, P25, P26, P27, P28, P29, P30, P31, P32	Maintain containment boundary

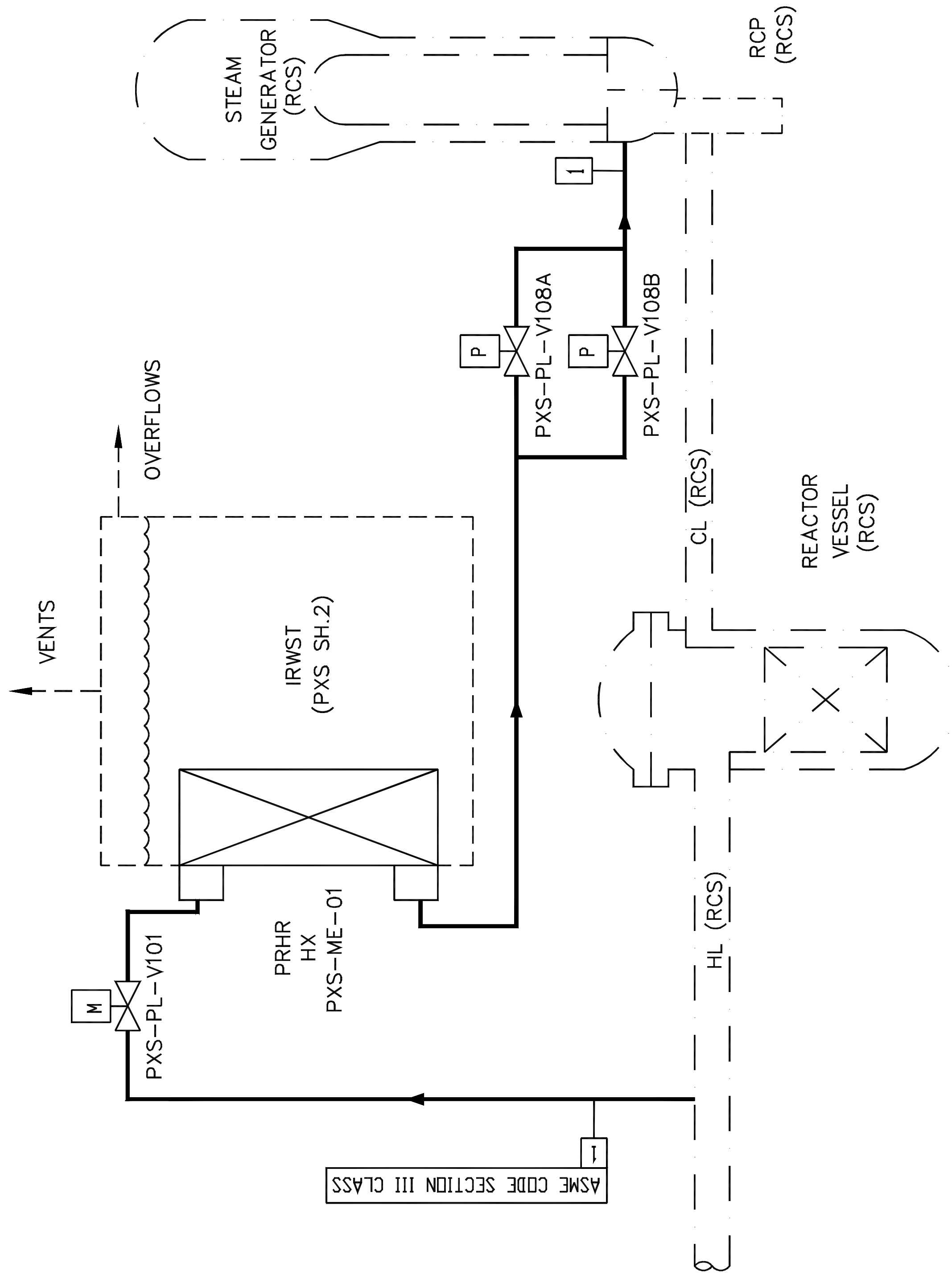


Figure 2.2.3-1 (Sheet 1 of 2)
Passive Core Cooling System

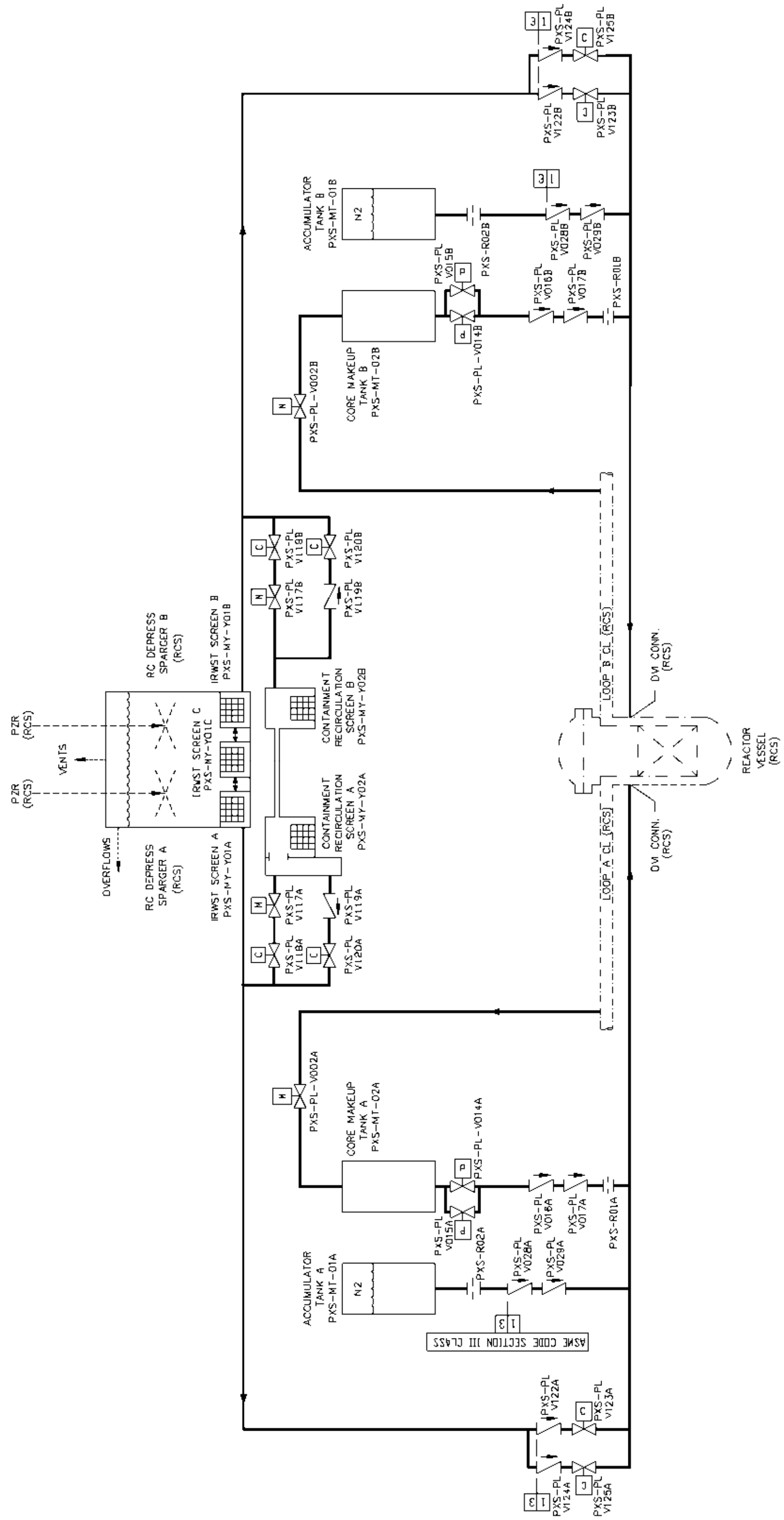


Figure 2.2.3-1 (Sheet 2 of 2)
Passive Core Cooling System

2.2.4 Steam Generator System

Design Description

The steam generator system (SGS) and portions of the main and startup feedwater system (FWS) transport and control feedwater from the condensate system to the steam generators during normal operation. The SGS and portions of the main steam system (MSS) and turbine system (MTS) transport and control steam from the steam generators to the turbine generator during normal operations. These systems also isolate the steam generators from the turbine generator and the condensate system during design basis accidents.

The SGS is as shown in Figure 2.2.4-1, sheets 1 and 2, and portions of the FWS, MSS, and MTS are as shown in Figure 2.2.4-1, sheet 3, and the locations of the components in these systems is as shown in Table 2.2.4-5.

1. The functional arrangement of the SGS and portions of the FWS, MSS, and MTS are as described in the Design Description of this Section 2.2.4.
2. a) The components identified in Table 2.2.4-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.2.4-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.2.4-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.2.4-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.2.4-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.2.4-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.
5. a) The seismic Category I equipment identified in Table 2.2.4-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.2.4-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.
6. Each of the as-built lines identified in Table 2.2.4-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.
7. a) The Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

b) The Class 1E components identified in Table 2.2.4-1 are powered from their respective Class 1E division.

c) Separation is provided between SGS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The SGS provides the following safety-related functions:

a) The SGS provides a heat sink for the reactor coolant system (RCS) and provides overpressure protection.

b) During design basis events, the SGS limits steam generator blowdown and feedwater flow to the steam generator.

c) The SGS preserves containment integrity by isolation of the SGS lines penetrating the containment. The inside containment isolation function (isolating the RCS and containment atmosphere from the environment) is provided by the steam generator, tubes, and SGS lines inside containment while isolation outside containment is provided by manual and automatic valves.

9. The SGS provides the following nonsafety-related functions:

a) Components within the main steam system, main and startup feedwater system, and the main turbine system identified in Table 2.2.4-3 provide backup isolation of the SGS to limit steam generator blowdown and feedwater flow to the steam generator.

b) During shutdown operations, the SGS removes decay heat by delivery of startup feedwater to the steam generator and venting of steam from the steam generators to the atmosphere.

10. Safety-related displays identified in Table 2.2.4-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.4-1 to perform active functions.

b) The valves identified in Table 2.2.4-1 as having PMS control perform an active safety function after receiving a signal from PMS.

12. a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.

b) After loss of motive power, the remotely operated valves identified in Table 2.2.4-1 assume the indicated loss of motive power position.

Table 2.2.4-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Main Steam Safety Valve SG01	SGS-PL-V030A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL-V030B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL-V031A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL-V031B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL-V032A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL-V032B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-

Note: Dash (-) indicates not applicable.

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Main Steam Safety Valve SG01	SGS-PL-V033A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL–V033B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL–V034A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL–V034B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG01	SGS-PL–V035A	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Main Steam Safety Valve SG02	SGS-PL–V035B	Yes	Yes	-	-/-	No	-	Transfer Open/ Transfer Closed	-
Power-operated Relief Valve Block Motor-operated Valve Steam Generator 01	SGS-PL-V027A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Power-operated Relief Valve Block Motor-operated Valve Steam Generator 02	SGS-PL-V027B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Steam Line Condensate Drain Isolation Valve	SGS-PL-V036A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Line Condensate Drain Isolation Valve	SGS-PL-V036B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Steam Line Isolation Valve	SGS-PL-V040A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Main Steam Line Isolation Valve	SGS-PL-V040B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Steam Line Condensate Drain Control Valve	SGS-PL-V086A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Line Condensate Drain Control Valve	SGS-PL-V086B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Feedwater Isolation Valve	SGS-PL-V057A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Main Feedwater Isolation Valve	SGS-PL-V057B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Startup Feedwater Isolation Motor-operated Valve	SGS-PL-V067A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Startup Feedwater Isolation Motor-operated Valve	SGS-PL-V067B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
Steam Generator Blowdown Isolation Valve	SGS-PL-V074A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator Blowdown Isolation Valve	SGS-PL-V074B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator Blowdown Isolation Valve	SGS-PL-V075A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator Blowdown Isolation Valve	SGS-PL-V075B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Power-operated Relief Valve	SGS-PL-V233A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Power-operated Relief Valve	SGS-PL-V233B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Main Steam Isolation Valve Bypass Isolation	SGS-PL-V240A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Steam Isolation Valve Bypass Isolation	SGS-PL-V240B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Feedwater Control Valve	SGS-PL-V250A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Main Feedwater Control Valve	SGS-PL-V250B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Startup Feedwater Control Valve	SGS-PL-V255A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Startup Feedwater Control Valve	SGS-PL-V255B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
Steam Generator 1 Narrow Range Level Sensor	SGS-001	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Narrow Range Level Sensor	SGS–002	No	Yes	-	Yes/Yes	Yes	-	-	-

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Steam Generator 1 Narrow Range Level Sensor	SGS–003	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Narrow Range Level Sensor	SGS–004	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS–005	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS–006	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS–007	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Narrow Range Level Sensor	SGS-008	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-011	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-012	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Wide Range Level Sensor	SGS-013	No	Yes	-	Yes/Yes	Yes	-	-	-

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Steam Generator 2 Wide Range Level Sensor	SGS-014	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-015	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 1 Wide Range Level Sensor	SGS-016	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Wide Range Level Sensor	SGS-017	No	Yes	-	Yes/Yes	Yes	-	-	-
Steam Generator 2 Wide Range Level Sensor	SGS-018	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-030	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-031	No	Yes	-	Yes/No	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-032	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 1 Pressure Sensor	SGS-033	No	Yes	-	Yes/No	Yes	-	-	-

Table 2.2.4-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-034	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-035	No	Yes	-	Yes/No	Yes	-	-	-
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-036	No	Yes	-	Yes/Yes	Yes	-	-	-
Main Steam Line Steam Generator 2 Pressure Sensor	SGS-037	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 1 Startup Feedwater Flow Sensor	SGS-55A	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 1 Startup Feedwater Flow Sensor	SGS-55B	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 2 Startup Feedwater Flow Sensor	SGS-56A	No	Yes	-	Yes/No	Yes	-	-	-
Steam Generator 2 Startup Feedwater Flow Sensor	SGS-56B	No	Yes	-	Yes/No	Yes	-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.4-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
Main Feedwater Line	SGS-PL-L002A, L002B	Yes	No	No
Main Feedwater Line	SGS-PL-L003A, L003B	Yes	No	No
Startup Feedwater Line	SGS-PL-L004A, L004B	Yes	No	No
Startup Feedwater Line	SGS-PL-L005A, L005B	Yes	No	No
Main Steam Line (within containment)	SGS-PL-L006A, L006B	Yes	Yes	Yes
Main Steam Line (outside of containment)	SGS-PL-L006A, L006B	Yes	No	Yes
Main Steam Line	SGS-PL-L007A, L007B	Yes	No	No
Safety Valve Inlet Line	SGS-PL-L015A, L015B, L015C, L015D, L015E, L015F, L015G, L015H, L015J, L015K, L015L, L015M	Yes	No	Yes
Safety Valve Discharge Line	SGS-PL-L018A, L018B, L018C, L018D, L018E, L018F, L018G, L018H, L018J, L018K, L018L, L018M	Yes	No	Yes
Power-operated Relief Block Valve Inlet Line	SGS-PL-L024A, L024B	Yes	No	No
Power-operated Relief Valve Inlet Line	SGS-PL-L014A, L014B	Yes	No	No
Main Steam Isolation Valve Bypass Inlet Line	SGS-PL-L022A, L022B	Yes	No	No
Main Steam Isolation Valve Bypass Outlet Line	SGS-PL-L023A, L023B	Yes	No	No
Main Steam Condensate Drain Line	SGS-PL-L021A, L021B	Yes	No	No

Table 2.2.4-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
Steam Generator Blowdown Line	SGS-PL-L009A, L009B	Yes	No	No
Steam Generator Blowdown Line	SGS-PL-L027A, L027B	Yes	No	No
Steam Generator Blowdown Line	SGS-PL-L010A, L010B	Yes	No	No

Note: Dash (-) indicates not applicable.

Table 2.2.4-3		
Equipment Name	**Tag No.**	**Control Function**
Turbine Stop Valve	MTS-PL-V001A	Close
Turbine Stop Valve	MTS-PL-V001B	Close
Turbine Control Valve	MTS-PL-V002A	Close
Turbine Control Valve	MTS-PL-V002B	Close
Turbine Stop Valve	MTS-PL-V003A	Close
Turbine Stop Valve	MTS-PL-V003B	Close
Turbine Control Valve	MTS-PL-V004A	Close
Turbine Control Valve	MTS-PL-V004B	Close
Turbine Bypass Control Valve	MSS-PL-V001	Close
Turbine Bypass Control Valve	MSS-PL-V002	Close
Turbine Bypass Control Valve	MSS-PL-V003	Close
Turbine Bypass Control Valve	MSS-PL-V004	Close
Turbine Bypass Control Valve	MSS-PL-V005	Close
Turbine Bypass Control Valve	MSS-PL-V006	Close
Moisture Separator Reheater 2nd Stage Steam Isolation Valve	MSS-PL-V015A	Close
Moisture Separator Reheater 2nd Stage Steam Isolation Valve	MSS-PL-V015B	Close
Main Feedwater Pump	FWS-MP-02A	Trip
Main Feedwater Pump	FWS-MP-02B	Trip
Main Feedwater Pump	FWS-MP-02C	Trip
Startup Feedwater Pump	FWS-MP-03A	Trip
Startup Feedwater Pump	FWS-MP-03B	Trip

Table 2.2.4-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
219	2.2.04.01	1. The functional arrangement of the SGS and portions of the FWS, MSS, and MTS are as described in the Design Description of this Section 2.2.4.	Inspection of the as-built system will be performed.	The as-built SGS and portions of the FWS, MSS, and MTS conform with the functional arrangement as defined in the Design Description of this Section 2.2.4.
220	2.2.04.02a	2.a) The components identified in Table 2.2.4-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.2.4-1 as ASME Code Section III.
221	2.2.04.02b	2.b) The piping identified in Table 2.2.4-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.2.4-2 as ASME Code Section III.
222	2.2.04.03a	3.a) Pressure boundary welds in components identified in Table 2.2.4-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
223	2.2.04.03b	3.b) Pressure boundary welds in piping identified in Table 2.2.4-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
224	2.2.04.04a	4.a) The components identified in Table 2.2.4-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.2.4-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
225	2.2.04.04b	4.b) The piping identified in Table 2.2.4-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.2.4-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

Table 2.2.4-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
226	2.2.04.05a.i	5.a) The seismic Category I equipment identified in Table 2.2.4-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.2.4-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.2.4-1 is located on the Nuclear Island.
227	2.2.04.05a.ii	5.a) The seismic Category I equipment identified in Table 2.2.4-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
228	2.2.04.05a.iii	5.a) The seismic Category I equipment identified in Table 2.2.4-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
229	2.2.04.05b	5.b) Each of the lines identified in Table 2.2.4-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report concluding that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.2.4-2 for which functional capability is required meets the requirements for functional capability.
230	2.2.04.06	6. Each of the as-built lines identified in Table 2.2.4-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from effects of a pipe break. Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built SGS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.

Table 2.2.4-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
231	2.2.04.07a.i	7.a) The Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
232	2.2.04.07a.ii	7.a) The Class 1E equipment identified in Table 2.2.4-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.2.4-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
233	2.2.04.07b	7.b) The Class 1E components identified in Table 2.2.4-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.2.4-1 when the assigned Class 1E division is provided the test signal.
234	2.2.04.07c	7.c) Separation is provided between SGS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
235	2.2.04.08a.i	8.a) The SGS provides a heat sink for the RCS and provides overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	i) Inspections will be conducted to confirm that the value of the vendor code plate rating of the steam generator safety valves is greater than or equal to system relief requirements.	i) The sum of the rated capacities recorded on the valve vendor code plates of the steam generator safety valves exceeds 8,240,000 lb/hr per steam generator.
236	2.2.04.08a.ii	8.a) The SGS provides a heat sink for the RCS and provides overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	ii) Testing and analyses in accordance with ASME Code Section III will be performed to determine set pressure.	ii) A report exists to indicate the set pressure of the valves is less than 1305 psig.

Table 2.2.4-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
237	2.2.04.08b.i	8.b) During design basis events, the SGS limits steam generator blowdown and feedwater flow to the steam generator.	i) Testing will be performed to confirm isolation of the main feedwater, startup feedwater, blowdown, and main steam lines. See item 11 in this table.	See item 11 in this table.
238	2.2.04.08b.ii	8.b) During design basis events, the SGS limits steam generator blowdown and feedwater flow to the steam generator.	ii) Inspection will be performed for the existence of a report confirming that the area of the flow limiting orifice within the SG main steam outlet nozzle will limit releases to the containment.	ii) A report exists to indicate the installed flow limiting orifice within the SG main steam line discharge nozzle does not exceed 1.4 sq. ft.
239	2.2.04.08c	8.c) The SGS preserves containment integrity by isolation of the SGS lines penetrating the containment.	See ITAAC Table 2.2.1-3, item 7.	See ITAAC Table 2.2.1-3, item 7.
240	2.2.04.09a.i	9.a) Components within the main steam system, main and startup feedwater system, and the main turbine system identified in Table 2.2.4-3 provide backup isolation of the SGS to limit steam generator blowdown and feedwater flow to the steam generator.	i) Testing will be performed to confirm closure of the valves identified in Table 2.2.4-3.	i) The valves identified in Table 2.2.4-3 close after a signal is generated by the PMS.
241	2.2.04.09a.ii	9.a) Components within the main steam system, main and startup feedwater system, and the main turbine system identified in Table 2.2.4-3 provide backup isolation of the SGS to limit steam generator blowdown and feedwater flow to the steam generator.	ii) Testing will be performed to confirm the trip of the pumps identified in Table 2.2.4-3.	ii) The pumps identified in Table 2.2.4-3 trip after a signal is generated by the PMS.
242	2.2.04.09b.i	9.b) During shutdown operations, the SGS removes decay heat by delivery of startup feedwater to the steam generator and venting of steam from the steam generators to the atmosphere.	i) Tests will be performed to demonstrate the ability of the startup feedwater system to provide feedwater to the steam generators.	i) See ITAAC Table 2.4.1-2, Item 2.
243	2.2.04.09b.ii	9.b) During shutdown operations, the SGS removes decay heat by delivery of startup feedwater to the steam generator and venting of steam from the steam generators to the atmosphere.	ii) Type tests and/or analyses will be performed to demonstrate the ability of the power-operated relief valves to discharge steam from the steam generators to the atmosphere.	ii) A report exists and concludes that each power-operated relief valve will relieve greater than 300,000 lb/hr at 1106 psia ±10 psi.
244	2.2.04.10	10. Safety-related displays identified in Table 2.2.4-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.4-1 can be retrieved in the MCR.

Table 2.2.4-4
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
245	2.2.04.11a	11.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.2.4-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves listed in Table 2.2.4-1 using controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves to perform active safety functions.
246	2.2.04.11b.i	11.b) The valves identified in Table 2.2.4-1 as having PMS control perform an active safety function after receiving a signal from PMS.	i) Testing will be performed on the remotely operated valves listed in Table 2.2.4-1 using real or simulated signals into the PMS.	i) The remotely-operated valves identified in Table 2.2.4-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
247	2.2.04.11b.ii	11.b) The valves identified in Table 2.2.4-1 as having PMS control perform an active safety function after receiving a signal from PMS.	ii) Testing will be performed to demonstrate that remotely operated SGS isolation valves SGS-V027A/B, V040A/B, V057A/B, V250A/B close within the required response times.	ii) These valves close within the following times after receipt of an actuation signal: V027A/B < 44 sec V040A/B, V057A/B < 5 sec V250A/B < 5 sec
248	2.2.04.12a.i	12.a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.2.4-1 under design conditions.
249	2.2.04.12a.ii	12.a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.
250	2.2.04.12a.iii	12.a) The motor-operated valves identified in Table 2.2.4-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.2.4-1 under pre-operational test conditions.
251	2.2.04.12b	12.b) After loss of motive power, the remotely operated valves identified in Table 2.2.4-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.4-1 assumes the indicated loss of motive power position. Motive power to SGS-PL-V040A/B and SGS-PL-V057A/B is electric power to the actuator from plant services.

Table 2.2.4-5		
Component Name	**Tag No.**	**Component Location**
Main Steam Line Isolation Valve	SGS-PL-V040A	Auxiliary Building
Main Steam Line Isolation Valve	SGS-PL-V040B	Auxiliary Building
Main Feedwater Isolation Valve	SGS-PL-V057A	Auxiliary Building
Main Feedwater Isolation Valve	SGS-PL-V057B	Auxiliary Building
Main Feedwater Control Valve	SGS-PL-V250A	Auxiliary Building
Main Feedwater Control Valve	SGS-PL-V250B	Auxiliary Building
Turbine Stop Valves	MTS-PL-V001A MTS-PL-V001B MTS-PL-V003A MTS-PL-V003B	Turbine Building
Turbine Control Valves	MTS-PL-V002A MTS-PL-V002B MTS-PL-V004A MTS-PL-V004B	Turbine Building
Main Feedwater Pumps	FWS-MP-02A FWS-MP-02B FWS-MP-02C	Turbine Building
Feedwater Booster Pumps	FWS-MP-01A FWS-MP-01B FWS-MP-01C	Turbine Building

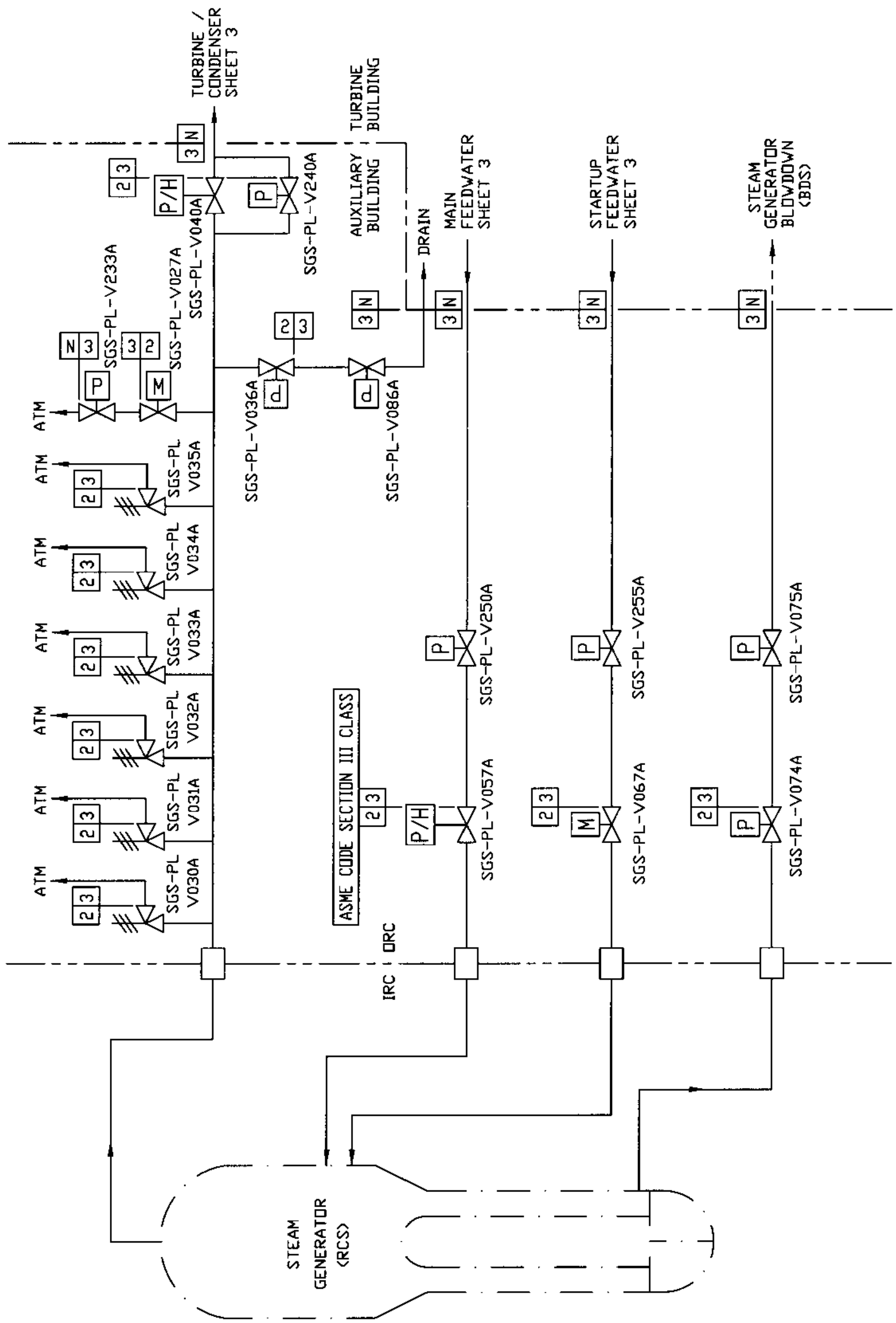


Figure 2.2.4-1 (Sheet 1 of 3)
Steam Generator System

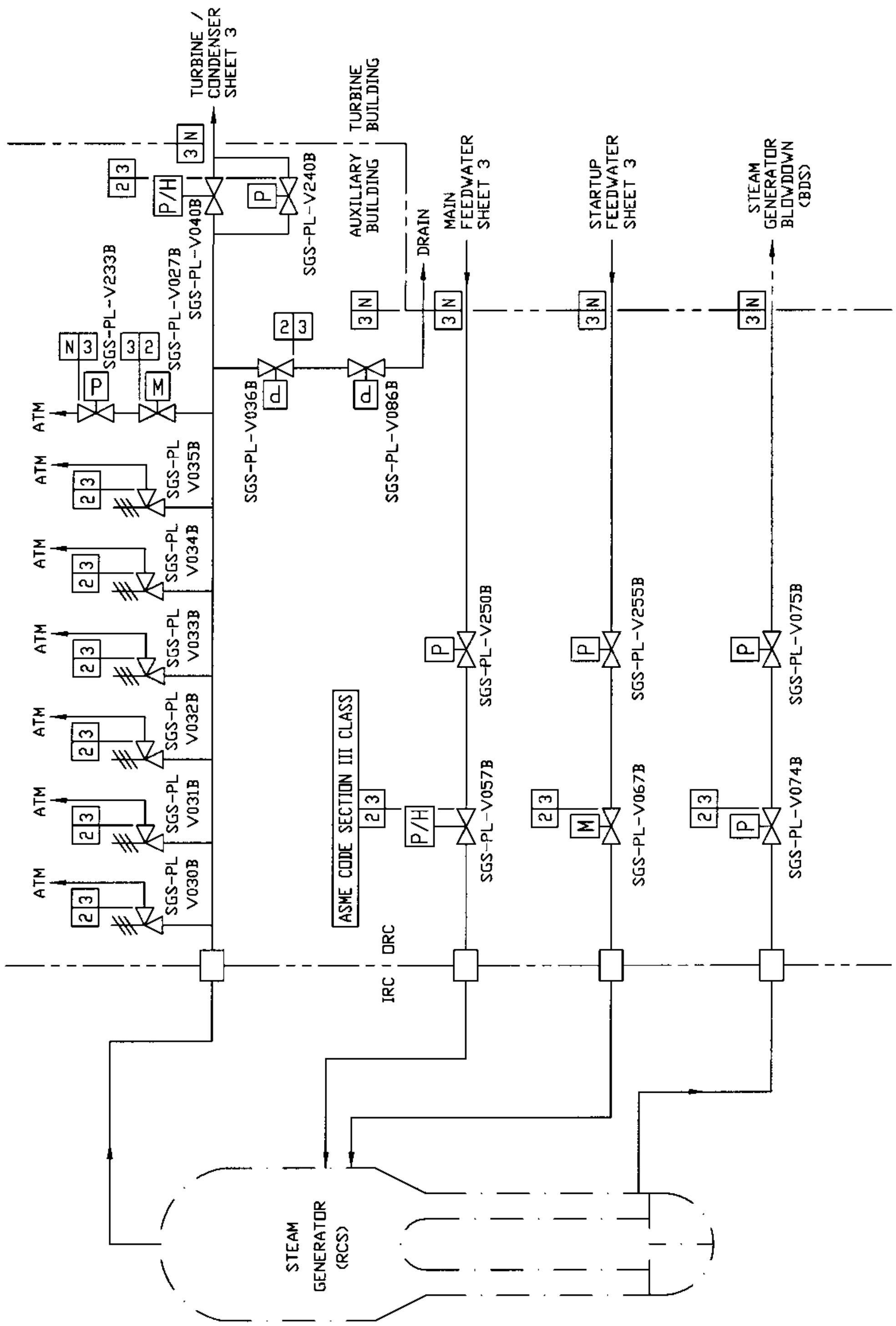


Figure 2.2.4-1 (Sheet 2 of 3)
Steam Generator System

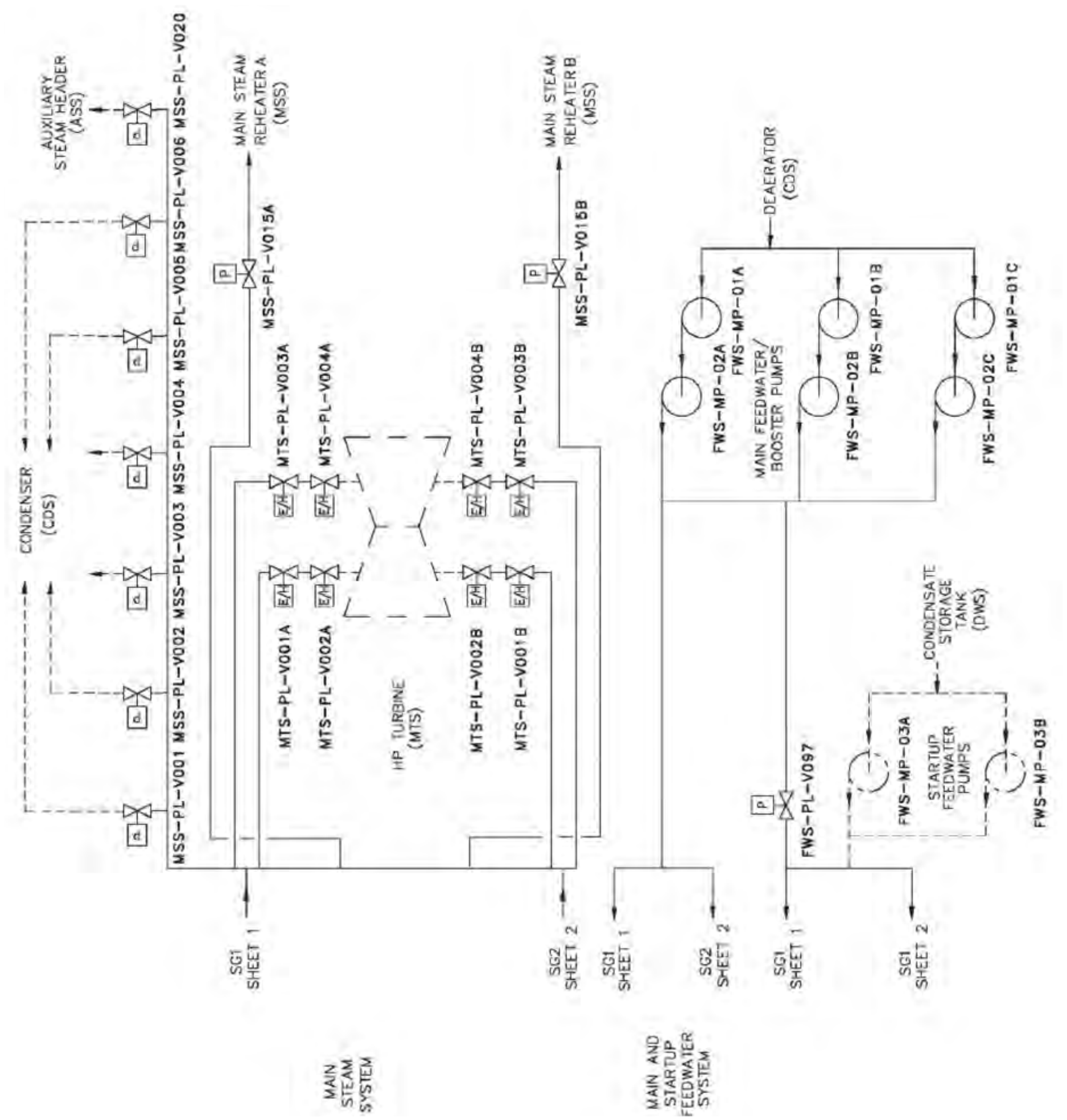


Figure 2.2.4-1 (Sheet 3 of 3)
Steam Generator System

2.2.5 Main Control Room Emergency Habitability System

Design Description

The main control room emergency habitability system (VES) provides a supply of breathable air for the main control room (MCR) occupants and maintains the MCR at a positive pressure with respect to the surrounding areas whenever ac power is not available to operate the nuclear island nonradioactive ventilation system (VBS) or high radioactivity is detected in the MCR air supply. (See Section 3.5 for Radiation Monitoring). The VES also limits the heatup of the MCR, the 1E instrumentation and control (I&C) equipment rooms, and the Class 1E dc equipment rooms by using the heat capacity of surrounding structures.

The VES is as shown in Figure 2.2.5-1 and the component locations of the VES are as shown in Table 2.2.5-6.

1. The functional arrangement of the VES is as described in the Design Description of this Section 2.2.5.
2. a) The components identified in Table 2.2.5-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
   b) The piping identified in Table 2.2.5-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.2.5-1 as ASME Code Section III meet ASME Code Section III requirements.
   b) Pressure boundary welds in piping identified in Table 2.2.5-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.2.5-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
   b) The piping identified in Table 2.2.5-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.
5. a) The seismic Category I equipment identified in Table 2.2.5-1 can withstand seismic design basis loads without loss of safety function.
   b) Each of the lines identified in Table 2.2.5-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.
6. a) The Class 1E components identified in Table 2.2.5-1 are powered from their respective Class 1E division.
   b) Separation is provided between VES Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.
7. The VES provides the following safety-related functions:
   a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.

b) The VES maintains the MCR pressure boundary at a positive pressure with respect to the surrounding areas. There is a discharge of air through the MCR vestibule.

c) The heat loads within the MCR, the I&C equipment rooms, and the Class 1E dc equipment rooms are within design basis assumptions to limit the heatup of the rooms identified in Table 2.2.5-4.

d) The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.

8. Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.

9. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.2.5-1 to perform their active functions.

   b) The valves identified in Table 2.2.5-1 as having protection and safety monitoring system (PMS) control perform their active safety function after receiving a signal from the PMS.

10. After loss of motive power, the remotely operated valves identified in Table 2.2.5-1 assume the indicated loss of motive power position.

11. Displays of the parameters identified in Table 2.2.5-3 can be retrieved in the MCR.

12. The background noise level in the MCR does not exceed 65 dB(A) at the operator workstations when the VES is operating.

Table 2.2.5-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Emergency Air Storage Tank 01	VES-MT-01	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 02	VES-MT-02	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 03	VES-MT-03	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 04	VES-MT-04	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 05	VES-MT-05	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 06	VES-MT-06	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 07	VES-MT-07	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 08	VES-MT-08	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 09	VES-MT-09	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 10	VES-MT-10	No	Yes	-	-/-	-	-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.5-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Emergency Air Storage Tank 11	VES-MT-11	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 12	VES-MT-12	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 13	VES-MT-13	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 14	VES-MT-14	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 15	VES-MT-15	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 16	VES-MT-16	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 17	VES-MT-17	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 18	VES-MT-18	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 19	VES-MT-19	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 20	VES-MT-20	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 21	VES-MT-21	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 22	VES-MT-22	No	Yes	-	-/-	-	-	-	-

Table 2.2.5-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Emergency Air Storage Tank 23	VES-MT-23	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 24	VES-MT-24	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 25	VES-MT-25	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 26	VES-MT-26	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 27	VES-MT-27	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 28	VES-MT-28	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 29	VES-MT-29	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 30	VES-MT-30	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 31	VES-MT-31	No	Yes	-	-/-	-	-	-	-
Emergency Air Storage Tank 32	VES-MT-32	No	Yes	-	-/-	-	-	-	-
Air Delivery Alternate Isolation Valve	VES-PL-V001	Yes	Yes	No	-/-	No	-	Transfer Open	-
Eductor Flow Path Isolation Valve	VES-PL-V045	Yes	Yes	No	-/-	No	-	Transfer Close	-

Table 2.2.5-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Eductor Bypass Isolation Valve	VES-PL-V046	Yes	Yes	No	-/-	No	-	Transfer Open	-
Pressure Regulating Valve A	VES-PL-V002A	Yes	Yes	No	-/-	No	-	Throttle Flow	-
Pressure Regulating Valve B	VES-PL-V002B	Yes	Yes	No	-/-	No	-	Throttle Flow	-
MCR Air Delivery Isolation Valve A	VES-PL-V005A	Yes	Yes	Yes	Yes/No	No	Yes	Transfer Open	Open
MCR Air Delivery Isolation Valve B	VES-PL-V005B	Yes	Yes	Yes	Yes/No	No	Yes	Transfer Open	Open
MCR Pressure Relief Isolation Valve A	VES-PL-V022A	Yes	Yes	Yes	Yes/No	No	Yes	Transfer Open	Open
MCR Pressure Relief Isolation Valve B	VES-PL-V022B	Yes	Yes	Yes	Yes/No	No	Yes	Transfer Open	Open
Air Tank Safety Relief Valve A	VES-PL-V040A	Yes	Yes	No	-/-	No	-	Transfer Open / Transfer Close	-
Air Tank Safety Relief Valve B	VES-PL-V040B	Yes	Yes	No	-/-	No	-	Transfer Open / Transfer Close	-

Table 2.2.5-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Air Tank Safety Relief Valve C	VES-PL-V040C	Yes	Yes	No	-/-	No	-	Transfer Open / Transfer Close	-
Air Tank Safety Relief Valve D	VES-PL-V040D	Yes	Yes	No	-/-	No	-	Transfer Open / Transfer Close	-
Main Air Flow Path Isolation Valve	VES-PL-V044	Yes	Yes	No	-/-	No	-	Transfer Close	-

Table 2.2.5-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
MCR Air Filtration Line Eductor	VES-PY-N01	Yes	Yes	-	-	-	-	-	-
MCR Air Filtration Line Charcoal Filter	VES-MY-F01	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line HEPA Filter	VES-MY-F02	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Postfilter	VES-MY-F03	No	Yes	-	-	-	-	-	-
MCR Gravity Relief Dampers	VES-MD-D001A	No	Yes	-	-	-	-	-	-
MCR Gravity Relief Dampers	VES-MD-D001B	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Supply Damper	VES-MD-D002	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Supply Damper	VES-MD-D003	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Silencer	VES-MY-Y01	No	Yes	-	-	-	-	-	-
MCR Air Filtration Line Silencer	VES-MY-Y02	No	Yes	-	-	-	-	-	-
MCR Air Delivery Line Flow Sensor	VES-003A	No	Yes	-	Yes/No	Yes	-	-	-
MCR Air Delivery Line Flow Sensor	VES-003B	No	Yes	-	Yes/No	Yes	-	-	-

Table 2.2.5-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
MCR Differential Pressure Sensor A	VES-004A	No	Yes	-	Yes/No	Yes	-	-	-
MCR Differential Pressure Sensor B	VES-004B	No	Yes	-	Yes/No	Yes	-	-	-

Note: Dash (-) indicates not applicable.

Table 2.2.5-2			
Line Name	**Line Number**	**ASME Code Section III**	**Functional Capability Required**
MCR Relief Line	VES-PL-022A	Yes	Yes
MCR Relief Line	VES-PL-022B	Yes	Yes

Table 2.2.5-3		
Equipment	**Tag No.**	**Display**
Air Storage Tank Pressure	VES-001A	Yes
Air Storage Tank Pressure	VES-001B	Yes

Table 2.2.5-4			
Room Name	**Room Numbers**	**Heat Load 0 to 24 Hours (Btu/s)**	**Heat Load 24 to 72 Hours (Btu/s)**
MCR Envelope	12401	12.8 (hour 0 through 3) 5.1 (hour 4 through 24)	3.9
I&C Rooms	12301, 12305	8.8	0
I&C Rooms	12302, 12304	13.0	4.2
dc Equipment Rooms	12201, 12205	3.7 (hour 0 through 1) 2.4 (hour 2 through 24)	0
dc Equipment Rooms	12203, 12207	5.8 (hour 0 through 1) 4.5 (hour 2 through 24)	2.0

Table 2.2.5-5 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
252	2.2.05.01	1. The functional arrangement of the VES is as described in the Design Description of this Section 2.2.5.	Inspection of the as-built system will be performed.	The as-built VES conforms with the functional arrangement described in the Design Description of this Section 2.2.5.
253	2.2.05.02a	2.a) The components identified in Table 2.2.5-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.2.5-1 as ASME Code Section III.

Table 2.2.5-5 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
254	2.2.05.02b	2.b) The piping identified in Table 2.2.5-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.2.5-2 as ASME Code Section III.
255	2.2.05.03a	3.a) Pressure boundary welds in components identified in Table 2.2.5-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
256	2.2.05.03b	3.b) Pressure boundary welds in piping identified in Table 2.2.5-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
257	2.2.05.04a	4.a) The components identified in Table 2.2.5-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.2.5-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
258	2.2.05.04b	4.b) The piping identified in Table 2.2.5-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.2.5-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
259	2.2.05.05a.i	5.a) The seismic Category I equipment identified in Table 2.2.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.2.5-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.2.5-1 is located on the Nuclear Island.
260	2.2.05.05a.ii	5.a) The seismic Category I equipment identified in Table 2.2.5-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.

Table 2.2.5-5 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
261	2.2.05.05a.iii	5.a) The seismic Category I equipment identified in Table 2.2.5-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
262	2.2.05.05b	5.b) Each of the lines identified in Table 2.2.5-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.2.5-2 for which functional capability is required meets the requirements for functional capability.
263	2.2.05.06a	6.a) The Class 1E components identified in Table 2.2.5-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.2.5-1 when the assigned Class 1E division is provided the test signal.
264	2.2.05.06b	6.b) Separation is provided between VES Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
265	2.2.05.07a.i	7.a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.	i) Testing will be performed to confirm that the required amount of air flow is delivered to the MCR.	i) The air flow rate from the VES is at least 60 scfm and not more than 70 scfm.
266	2.2.05.07a.ii	7.a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.	ii) Analysis of storage capacity will be performed based on manufacturers data.	ii) The calculated storage capacity is greater than or equal to 327,574 scf.
267	2.2.05.07a.iii	7.a) The VES provides a 72-hour supply of breathable quality air for the occupants of the MCR.	iii) MCR air samples will be taken during VES testing and analyzed for quality.	iii) The MCR air is of breathable quality.
268	2.2.05.07b.i	7.b) The VES maintains the MCR pressure boundary at a positive pressure with respect to the surrounding areas.	i) Testing will be performed with VES flow rate between 60 and 70 scfm to confirm that the MCR is capable of maintaining the required pressurization of the pressure boundary.	i) The MCR pressure boundary is pressurized to greater than or equal to 1/8-in. water gauge with respect to the surrounding area.
269	2.2.05.07b.ii	7.b) The VES maintains the MCR pressure boundary at a positive pressure with respect to the surrounding areas.	ii) Air leakage into the MCR will be measured during VES testing using a tracer gas.	ii) Air leakage into the MCR is less than or equal to 10 cfm.

Table 2.2.5-5 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
270	2.2.05.07c	7.c) The heat loads within the MCR, the I&C equipment rooms, and the Class 1E dc equipment rooms are within design basis assumptions to limit the heatup of the rooms identified in Table 2.2.5-4.	An analysis will be performed to determine that the heat loads from as-built equipment within the rooms identified in Table 2.2.5-4 are less than or equal to the design basis assumptions.	A report exists and concludes that: the heat loads within rooms identified in Table 2.2.5-4 are less than or equal to the specified values or that an analysis report exists that concludes: – The temperature and humidity in the MCR remain within limits for reliable human performance for the 72-hour period. – The maximum temperature for the 72-hour period for the I&C rooms is less than or equal to 120°F. – The maximum temperature for the 72-hour period for the Class 1E dc equipment rooms is less than or equal to 120°F.
271	2.2.05.07d	7.d) The system provides a passive recirculation flow of MCR air to maintain main control room dose rates below an acceptable level during VES operation.	Testing will be performed to confirm that the required amount of air flow circulates through the MCR passive filtration system.	The air flow rate at the outlet of the MCR passive filtration system is at least 600 cfm greater than the flow measured by VES-003A/B.
272	2.2.05.08	8. Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.2.5-1 can be retrieved in the MCR.
273	2.2.05.09a	9.a) Controls exist in the MCR to cause remotely operated valves identified in Table 2.2.5-1 to perform their active functions.	Stroke testing will be performed on remotely operated valves identified in Table 2.2.5-1 using the controls in the MCR.	Controls in the MCR operate to cause remotely operated valves identified in Table 2.2.5-1 to perform their active safety functions.
274	2.2.05.09b	9.b) The valves identified in Table 2.2.5-1 as having PMS control perform their active safety function after receiving a signal from the PMS.	Testing will be performed on remotely operated valves listed in Table 2.2.5-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.2.5-1 as having PMS control perform the active safety function identified in the table after receiving a signal from the PMS.

Table 2.2.5-5 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
275	2.2.05.10	10. After loss of motive power, the remotely operated valves identified in Table 2.2.5-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.2.5-1 assumes the indicated loss of motive power position.
276	2.2.05.11	11. Displays of the parameters identified in Table 2.2.5-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.2.5-3 can be retrieved in the MCR.
277	2.2.05.12	12. The background noise level in the MCR does not exceed 65 dB(A) at the operator workstations when VES is operating.	The as-built VES will be operated, and background noise levels in the MCR will be measured at the operator work stations with the plant not operating.	The background noise level in the MCR does not exceed 65 dB(A) at the operator work stations when the VES is operating.

Table 2.2.5-6		
Component Name	**Tag Number**	**Component Location**
Emergency Air Storage Tank 01	VES-MT-01	Auxiliary Building
Emergency Air Storage Tank 02	VES-MT-02	Auxiliary Building
Emergency Air Storage Tank 03	VES-MT-03	Auxiliary Building
Emergency Air Storage Tank 04	VES-MT-04	Auxiliary Building
Emergency Air Storage Tank 05	VES-MT-05	Auxiliary Building
Emergency Air Storage Tank 06	VES-MT-06	Auxiliary Building
Emergency Air Storage Tank 07	VES-MT-07	Auxiliary Building
Emergency Air Storage Tank 08	VES-MT-08	Auxiliary Building
Emergency Air Storage Tank 09	VES-MT-09	Auxiliary Building
Emergency Air Storage Tank 10	VES-MT-10	Auxiliary Building
Emergency Air Storage Tank 11	VES-MT-11	Auxiliary Building
Emergency Air Storage Tank 12	VES-MT-12	Auxiliary Building
Emergency Air Storage Tank 13	VES-MT-13	Auxiliary Building
Emergency Air Storage Tank 14	VES-MT-14	Auxiliary Building
Emergency Air Storage Tank 15	VES-MT-15	Auxiliary Building
Emergency Air Storage Tank 16	VES-MT-16	Auxiliary Building

Table 2.2.5-6		
Component Name	**Tag Number**	**Component Location**
Emergency Air Storage Tank 17	VES-MT-17	Auxiliary Building
Emergency Air Storage Tank 18	VES-MT-18	Auxiliary Building
Emergency Air Storage Tank 19	VES-MT-19	Auxiliary Building
Emergency Air Storage Tank 20	VES-MT-20	Auxiliary Building
Emergency Air Storage Tank 21	VES-MT-21	Auxiliary Building
Emergency Air Storage Tank 22	VES-MT-22	Auxiliary Building
Emergency Air Storage Tank 23	VES-MT-23	Auxiliary Building
Emergency Air Storage Tank 24	VES-MT-24	Auxiliary Building
Emergency Air Storage Tank 25	VES-MT-25	Auxiliary Building
Emergency Air Storage Tank 26	VES-MT-26	Auxiliary Building
Emergency Air Storage Tank 27	VES-MT-27	Auxiliary Building
Emergency Air Storage Tank 28	VES-MT-28	Auxiliary Building
Emergency Air Storage Tank 29	VES-MT-29	Auxiliary Building
Emergency Air Storage Tank 30	VES-MT-30	Auxiliary Building
Emergency Air Storage Tank 31	VES-MT-31	Auxiliary Building
Emergency Air Storage Tank 32	VES-MT-32	Auxiliary Building

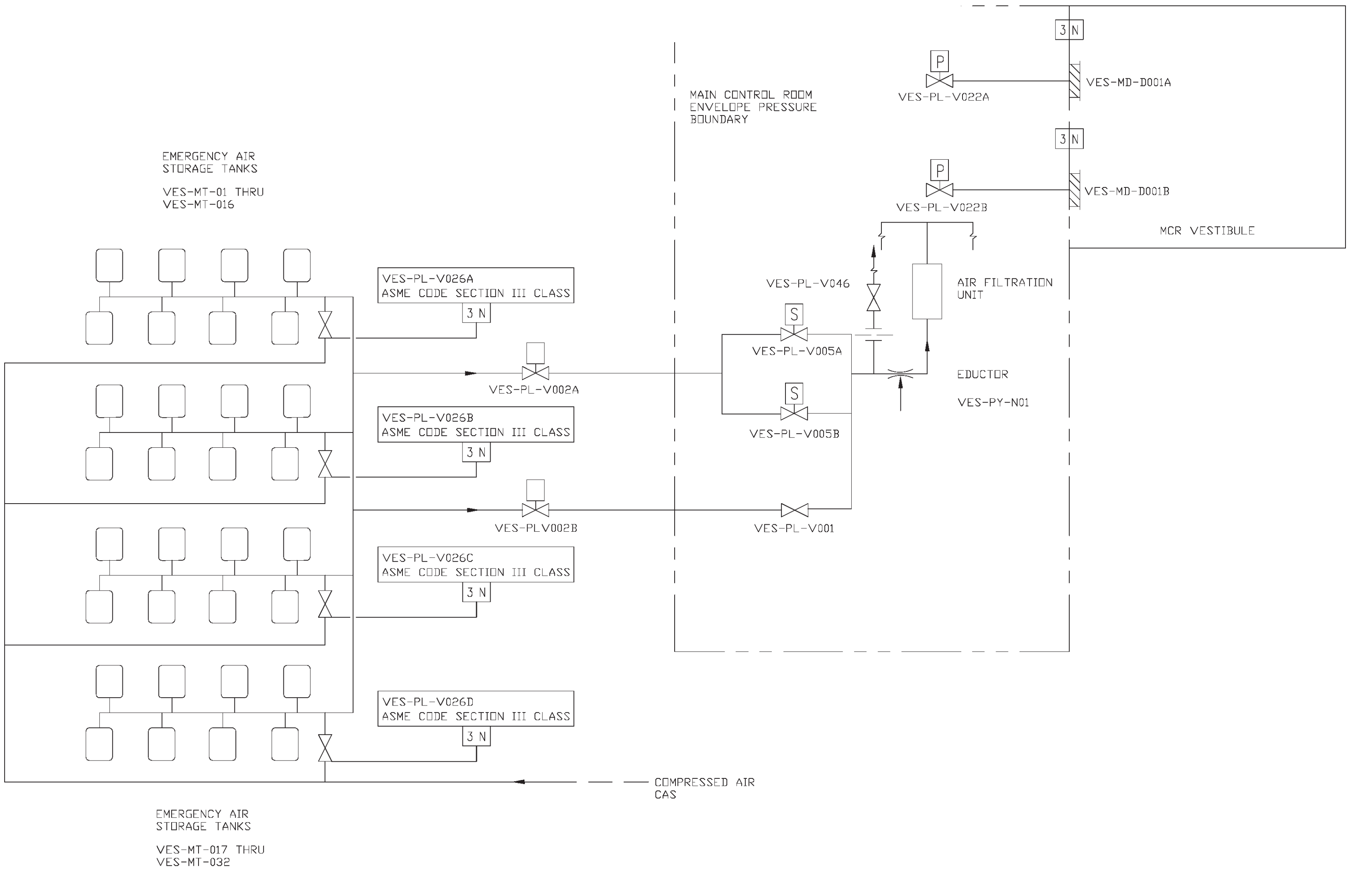


Figure 2.2.5-1
Main Control Room Emergency Habitability System

2.3 Auxiliary Systems

2.3.1 Component Cooling Water System

Design Description

The component cooling water system (CCS) removes heat from various plant components and transfers this heat to the service water system (SWS) during normal modes of plant operation including power generation, shutdown and refueling. The CCS has two pumps and two heat exchangers.

The CCS is as shown in Figure 2.3.1-1 and the CCS component locations are as shown in Table 2.3.1-3.

1. The functional arrangement of the CCS is as described in the Design Description of this Section 2.3.1.
2. The CCS preserves containment integrity by isolation of the CCS lines penetrating the containment.
3. The CCS provides the nonsafety-related functions of transferring heat from the normal residual heat removal system (RNS) during shutdown and the spent fuel pool cooling system during all modes of operation to the SWS.
4. Controls exist in the main control room (MCR) to cause the pumps identified in Table 2.3.1-1 to perform the listed functions.
5. Displays of the parameters identified in Table 2.3.1-1 can be retrieved in the MCR.

Table 2.3.1-1

Equipment Name	Tag No.	Display	Control Function
CCS Pump A	CCS-MP-01A	Yes (Run Status)	Start
CCS Pump B	CCS-MP-01B	Yes (Run Status)	Start
CCS Discharge Header Flow Sensor	CCS-101	Yes	-
CCS to Normal Residual Heat Removal System Heat Exchanger (RNS HX) A Flow Sensor	CCS-301	Yes	-
CCS to RNS HX B Flow Sensor	CCS-302	Yes	-
CCS to Spent Fuel Pool Cooling System (SFS) HX A Flow Sensor	CCS-341	Yes	-
CCS to SFS HX B Flow Sensor	CCS-342	Yes	-
CCS Surge Tank Level Sensor A	CCS-130	Yes	-
CCS Surge Tank Level Sensor B	CCS-131	Yes	-

Table 2.3.1-1			
Equipment Name	**Tag No.**	**Display**	**Control Function**
CCS Heat Exchanger Inlet Temperature Sensor	CCS-121	Yes	-
CCS Heat Exchanger Outlet Temperature Sensor	CCS-122	Yes	-
CCS Flow to Reactor Coolant Pump (RCP) 1A Valve (Position Indicator)	CCS-PL-V256A	Yes	-
CCS Flow to RCP 1B Valve (Position Indicator)	CCS-PL-V256B	Yes	-
CCS Flow to RCP 2A Valve (Position Indicator)	CCS-PL-V256C	Yes	-
CCS Flow to RCP 2B Valve (Position Indicator)	CCS-PL-V256D	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.1-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
278	2.3.01.01	1. The functional arrangement of the CCS is as described in the Design Description of this Section 2.3.1.	Inspection of the as-built system will be performed.	The as-built CCS conforms with the functional arrangement described in the Design Description of this Section 2.3.1.
279	2.3.01.02	2. The CCS preserves containment integrity by isolation of the CCS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
280	2.3.01.03.i	3. The CCS provides the nonsafety-related functions of transferring heat from the RNS during shutdown and the spent fuel pool cooling system during all modes of operation to the SWS.	i) Inspection will be performed for the existence of a report that determines the heat transfer capability of the CCS heat exchangers.	i) A report exists and concludes that the UA of each CCS heat exchanger is greater than or equal to 14.0 million Btu/hr-°F.
281	2.3.01.03.ii	3. The CCS provides the nonsafety-related functions of transferring heat from the RNS during shutdown and the spent fuel pool cooling system during all modes of operation to the SWS.	ii) Testing will be performed to confirm that the CCS can provide cooling water to the RNS HXs while providing cooling water to the SFS HXs.	ii) Each pump of the CCS can provide at least 2685 gpm of cooling water to one RNS HX and at least 1200 gpm of cooling water to one SFS HX while providing at least 4415 gpm to other users of cooling water.

Table 2.3.1-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
282	2.3.01.04	4. Controls exist in the MCR to cause the pumps identified in Table 2.3.1-1 to perform the listed functions.	Testing will be performed to actuate the pumps identified in Table 2.3.1-1 using controls in the MCR.	Controls in the MCR operate to cause pumps listed in Table 2.3.1-1 to perform the listed functions.
283	2.3.01.05	5. Displays of the parameters identified in Table 2.3.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	Displays identified in Table 2.3.1-1 can be retrieved in the MCR.

Table 2.3.1-3		
Component Name	**Tag No.**	**Component Location**
CCS Pump A	CCS-MP-01A	Turbine Building
CCS Pump B	CCS-MP-01B	Turbine Building
CCS Heat Exchanger A	CCS-ME-01A	Turbine Building
CCS Heat Exchanger B	CCS-ME-01B	Turbine Building

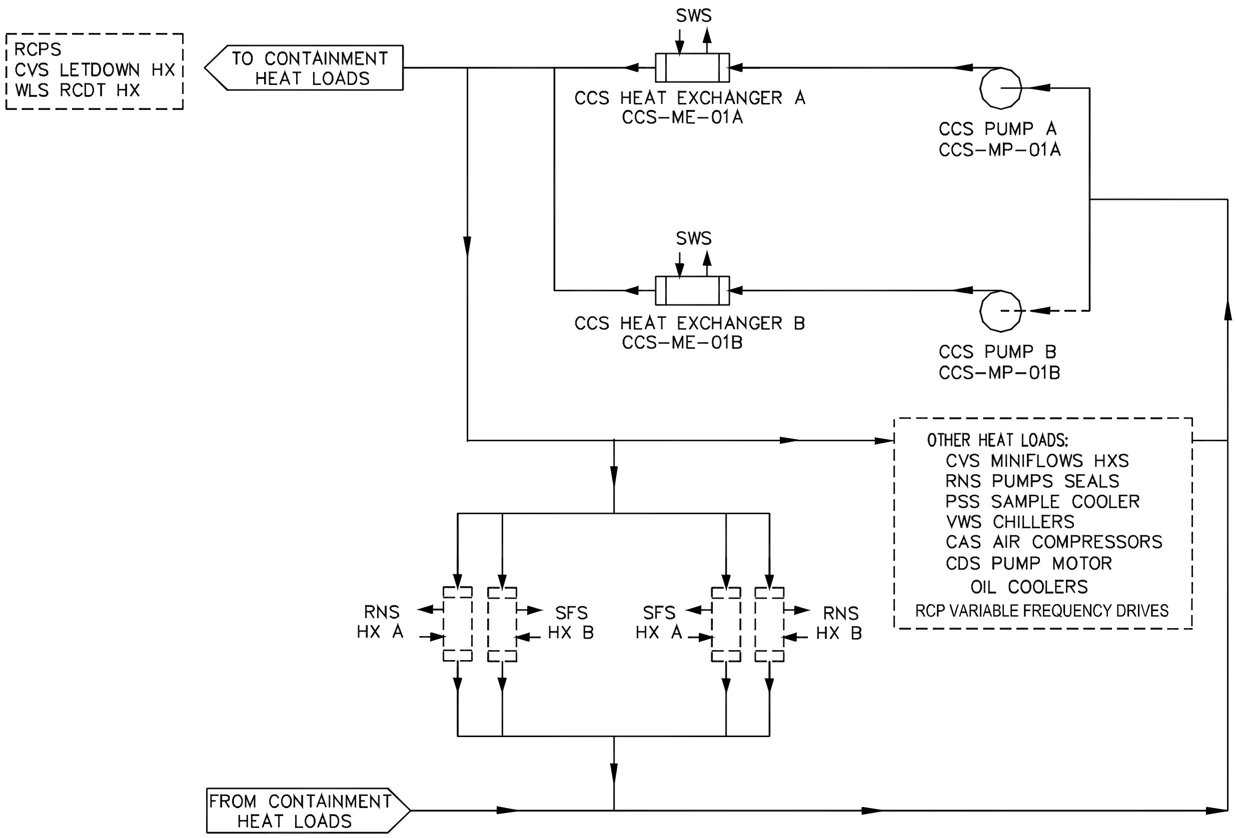


Figure 2.3.1-1
Component Cooling Water System

2.3.2 Chemical and Volume Control System

Design Description

The chemical and volume control system (CVS) provides reactor coolant system (RCS) purification, RCS inventory control and makeup, chemical shim and chemical control, oxygen control, and auxiliary pressurizer spray. The CVS performs these functions during normal modes of operation including power generation and shutdown.

The CVS is as shown in Figure 2.3.2-1 and the component locations of the CVS are as shown in Table 2.3.2-5.

1. The functional arrangement of the CVS is as described in the Design Description of this Section 2.3.2.

2. a) The components identified in Table 2.3.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.3.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.3.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.3.2-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.3.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.3.2-1 are powered from their respective Class 1E division.

   c) Separation is provided between CVS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

7. The CVS provides the following safety-related functions:

   a) The CVS preserves containment integrity by isolation of the CVS lines penetrating the containment.

b) The CVS provides termination of an inadvertent RCS boron dilution by isolating demineralized water from the RCS.

c) The CVS provides isolation of makeup to the RCS.

8. The CVS provides the following nonsafety-related functions:

    a) The CVS provides makeup water to the RCS.

    b) The CVS provides the pressurizer auxiliary spray.

9. Safety-related displays in Table 2.3.2-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.

    b) The valves identified in Table 2.3.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

11. a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.

    b) After a loss of motive power, the remotely operated valves identified in Table 2.3.2-1 assume the indicated loss of motive power position.

12. a) Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function.

    b) The pumps identified in Table 2.3.2-3 start after receiving a signal from the PLS.

13. Displays of the parameters identified in Table 2.3.2-3 can be retrieved in the MCR.

14. The nonsafety-related piping located inside containment and designated as reactor coolant pressure boundary, as identified in Table 2.3.2-2 (pipe lines with "No" in the ASME Code column), has been designed to withstand a seismic design basis event and maintain structural integrity.

Table 2.3.2-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
RCS Purification Motor-operated Isolation Valve	CVS-PL-V001	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Purification Motor-operated Isolation Valve	CVS-PL-V002	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Purification Motor-operated Isolation Valve	CVS-PL-V003	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Resin Flush Line Containment Isolation Valve	CVS-PL-V040	Yes	Yes	No	- / -	-	-	-	-
CVS Resin Flush Line Containment Isolation Valve	CVS-PL-V041	Yes	Yes	No	- / -	-	-	-	-
CVS Demineralizer Resin Flush Line Containment Isolation Thermal Relief Valve	CVS-PL-V042	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Letdown Containment Isolation Valve	CVS-PL-V045	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Letdown Containment Isolation Valve	CVS-PL-V047	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

Table 2.3.2-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
CVS Letdown Line Containment Isolation Thermal Relief Valve	CVS-PL-V058	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Makeup Return Line Bypass Check Valve	CVS-PL-V067	Yes	Yes	No	-/-	-	-	Transfer Open/ Transfer Closed	-
CVS Purification Return Line Pressure Boundary Check Valve	CVS-PL-V080	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Purification Return Line Pressure Boundary Isolation Check Valve	CVS-PL-V081	Yes	Yes	No	- / -	No	-	Transfer Closed	-
CVS Purification Return Line Pressure Boundary Check Valve	CVS-PL-V082	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Auxiliary Pressurizer Spray Line Pressure Boundary Valve	CVS-PL-V084	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Auxiliary Pressurizer Spray Line Pressure Boundary Check Valve	CVS-PL-V085	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Makeup Line Containment Isolation Motor-operated Valve	CVS-PL-V090	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Makeup Line Containment Isolation Motor-operated Valve	CVS-PL-V091	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is

Table 2.3.2-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
CVS Zinc Injection Containment Isolation Valve ORC	CVS-PL-V092	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Zinc Injection Containment Isolation Valve IRC	CVS-PL-V094	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Zinc Addition Line Ctmt Isol Thermal Relief Valve	CVS-PL-V098	Yes	Yes	No	-/-	-	-	Transfer Open/ Transfer Closed	-
CVS Makeup Line Containment Isolation Thermal Relief Valve	CVS-PL-V100	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Demineralized Water Isolation Valve	CVS-PL-V136A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Demineralized Water Isolation Valve	CVS-PL-V136B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Hydrogen Injection Containment Isolation Check Valve IRC	CVS-PL-V217	Yes	Yes	No	-/-	-	-	Transfer Closed	-
CVS Hydrogen Injection Containment Isolation Valve ORC	CVS-PL-V219	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

Table 2.3.2-2		
Line Name	**Line Number**	**ASME Code Section III**
CVS Purification Line	L001 L040	Yes Yes
CVS Resin Flush Containment Penetration Line	L026	Yes
CVS Purification Line Return	L038	Yes
CVS Pressurizer Auxiliary Spray Connection	L070 L071	Yes Yes
CVS Letdown Containment Penetration Line	L051	Yes
CVS Makeup Containment Penetration Line	L053	Yes
CVS Hydrogen Injection Containment Penetration Line	L213 L214 L217	Yes
CVS Zinc Injection Containment Penetration Line	L061	Yes
CVS Supply Line to Regenerative Heat Exchanger	L002	No
CVS Return Line from Regenerative Heat Exchanger	L018 L036	No Yes
CVS Line from Regenerative Heat Exchanger to Letdown Heat Exchanger	L003	No
CVS Lines from Letdown Heat Exchanger to Demin. Tanks	L004 L005 L072	No No No
CVS Lines from Demin Tanks to RC Filters and Connected Lines	L006(1) L007(1) L010(1) L011(1) L012 L015(1) L016(1) L020 L021 L022 L023(1) L024(1) L029 L037	No No No No No No No No No No No No No No

Table 2.3.2-2 (cont.)		
Line Name	**Line Number**	**ASME Code Section III**
CVS Lines from RC Filters to Regenerative Heat Exchanger	L030 L031 L034 L050	No No No No
CVS Resin Fill Lines to Demin. Tanks	L008[(1)] L013[(1)] L025[(1)]	No No No

Note:

1. Special seismic requirements include only the portion of piping normally exposed to RCS pressure. Piping beyond the first normally closed isolation valve is evaluated as seismic Category II piping extending to either an interface anchor, a rigid support following a six-way anchor, or the last seismic support of a rigidly supported region of the piping system as necessary to satisfy analysis requirements for piping connected to seismic Category I piping systems.

Table 2.3.2-3			
Equipment	**Tag No.**	**Display**	**Control Function**
CVS Makeup Pump A	CVS-MP-01A	Yes (Run Status)	Start
CVS Makeup Pump B	CVS-MP-01B	Yes (Run Status)	Start
Purification Flow Sensor	CVS-001	Yes	-
Purification Return Flow Sensor	CVS-025	Yes	-
CVS Purification Return Line (Position Indicator)	CVS-PL-V081	Yes	-
Auxiliary Spray Line Isolation Valve (Position Indicator)	CVS-PL-V084	Yes	-
Boric Acid Storage Tank Level Sensor	CVS-109	Yes	-
Boric Acid Flow Sensor	CVS-115	Yes	-
Makeup Blend Valve (Position Indicator)	CVS-PL-V115	Yes	-
CVS Demineralized Water Isolation Valve (Position Indicator)	CVS-PL-V136A	Yes	-
CVS Demineralized Water Isolation Valve (Position Indicator)	CVS-PL-V136B	Yes	-
Makeup Pump Discharge Flow Sensor	CVS-157	Yes	-
Makeup Flow Control Valve (Position Indicator)	CVS-PL-V157	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
284	2.3.02.01	1. The functional arrangement of the CVS is as described in the Design Description of this Section 2.3.2.	Inspection of the as-built system will be performed.	The as-built CVS conforms with the functional arrangement as described in the Design Description of this Section 2.3.2.
285	2.3.02.02a	2.a) The components identified in Table 2.3.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.3.2-1 as ASME Code Section III.
286	2.3.02.02b	2.b) The piping identified in Table 2.3.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.3.2-2 as ASME Code Section III.
287	2.3.02.03a	3.a) Pressure boundary welds in components identified in Table 2.3.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
288	2.3.02.03b	3.b) Pressure boundary welds in piping identified in Table 2.3.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
289	2.3.02.04a	4.a) The components identified in Table 2.3.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.2-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
290	2.3.02.04b	4.b) The piping identified in Table 2.3.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.3.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.

Table 2.3.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
291	2.3.02.05.i	5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.2-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.2-1 is located on the Nuclear Island.
292	2.3.02.05.ii	5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function.
293	2.3.02.05.iii	5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
294	2.3.02.06a.i	6.a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
295	2.3.02.06a.ii	6.a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.3.2-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

Table 2.3.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
296	2.3.02.06b	6.b) The Class 1E components identified in Table 2.3.2-1 are powered from their respective Class 1E division.	Testing will be performed on the CVS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.2-1 when the assigned Class 1E division is provided the test signal.
297	2.3.02.06c	6.c) Separation is provided between CVS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
298	2.3.02.07a	7.a) The CVS preserves containment integrity by isolation of the CVS lines penetrating the containment.	See ITAAC Table 2.2.1-3, item 7.	See ITAAC Table 2.2.1-3, item 7.
299	2.3.02.07b	7.b) The CVS provides termination of an inadvertent RCS boron dilution by isolating demineralized water from the RCS.	See item 10b in this table.	See item 10b in this table.
300	2.3.02.07c	7.c) The CVS provides isolation of makeup to the RCS.	See item 10b in this table.	See item 10b in this table.
301	2.3.02.08a.i	8.a) The CVS provides makeup water to the RCS.	i) Testing will be performed by aligning a flow path from each CVS makeup pump, actuating makeup flow to the RCS at pressure greater than or equal to 2000 psia, and measuring the flow rate in the makeup pump discharge line with each pump suction aligned to the boric acid storage tank.	i) Each CVS makeup pump provides a flow rate of greater than or equal to 100 gpm.
302	2.3.02.08a.ii	8.a) The CVS provides makeup water to the RCS.	ii) Inspection of the boric acid storage tank volume will be performed.	ii) The volume in the boric acid storage tank is at least 70,000 gallons between the tank suction point and the tank overflow.
303	2.3.02.08a.iii	8.a) The CVS provides makeup water to the RCS.	iii) Testing will be performed to measure the delivery rate from the DWS to the RCS. Both CVS makeup pumps will be operating and the RCS pressure will be below 6 psig.	iii) The total CVS makeup flow to the RCS is less than or equal to 175 gpm.

Table 2.3.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
304	2.3.02.08b	8.b) The CVS provides the pressurizer auxiliary spray.	Testing will be performed by aligning a flow path from each CVS makeup pump to the pressurizer auxiliary spray and measuring the flow rate in the makeup pump discharge line with each pump suction aligned to the boric acid storage tank and with RCS pressure greater than or equal to 2000 psia.	Each CVS makeup pump provides spray flow to the pressurizer.
305	2.3.02.09	9. Safety-related displays identified in Table 2.3.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.2-1 can be retrieved in the MCR.
306	2.3.02.10a	10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.2-1 using the controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.
307	2.3.02.10b.i	10.b) The valves identified in Table 2.3.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	i) Testing will be performed using real or simulated signals into the PMS.	i) The valves identified in Table 2.3.2-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
308	2.3.02.10b.ii	10.b) The valves identified in Table 2.3.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	ii) Testing will be performed to demonstrate that the remotely operated CVS isolation valves CVS-V090, V091, V136A/B close within the required response time.	ii) These valves close within the following times after receipt of an actuation signal: V090, V091 < 30 sec V136A/B < 20 sec
309	2.3.02.11a.i	11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.3.2-1 under design conditions.
310	2.3.02.11a.ii	11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tested conditions.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tests or type tests.

Table 2.3.2-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
311	2.3.02.11a.iii	11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.3.2-1 under pre-operational test conditions.
312	2.3.02.11a.iv	11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	iv) Exercise testing of the check valves with active safety functions identified in Table 2.3.2-1 will be performed under pre-operational test pressure, temperature and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.3.2-1.
313	2.3.02.11b	11.b) After loss of motive power, the remotely operated valves identified in Table 2.3.2-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.3.2-1 assumes the indicated loss of motive power position.
314	2.3.02.12a	12.a) Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function.	Testing will be performed to actuate the pumps identified in Table 2.3.2-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.2-3 to perform the listed function.
315	2.3.02.12b	12.b) The pumps identified in Table 2.3.2-3 start after receiving a signal from the PLS.	Testing will be performed to confirm starting of the pumps identified in Table 2.3.2-3.	The pumps identified in Table 2.3.2-3 start after a signal is generated by the PLS.
316	2.3.02.13	13. Displays of the parameters identified in Table 2.3.2-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.2-3 in the MCR.	Displays identified in Table 2.3.2-3 can be retrieved in the MCR.
317	2.3.02.14	14. The nonsafety-related piping located inside containment and designated as reactor coolant pressure boundary, as identified in Table 2.3.2-2, has been designed to withstand a seismic design basis event and maintain structural integrity.	Inspection will be conducted of the as-built components as documented in the CVS Seismic Analysis Report.	The CVS Seismic Analysis Reports exist for the non-safety related piping located inside containment and designated as reactor coolant pressure boundary as identified in Table 2.3.2-2.

Table 2.3.2-5		
Component Name	**Tag No.**	**Component Location**
CVS Makeup Pump A	CVS-MP-01A	Auxiliary Building
CVS Makeup Pump B	CVS-MP-01B	Auxiliary Building
Boric Acid Storage Tank	CVS-MT-01	Yard
Regenerative Heat Exchanger	CVS-ME-01	Containment
Letdown Heat Exchanger	CVS-ME-02	Containment
Mixed Bed Demineralizer A	CVS-MV-01A	Containment
Mixed Bed Demineralizer B	CVS-MV-01B	Containment
Cation Bed Demineralizer	CVS-MV-02	Containment
Reactor Coolant Filter A	CVS-MV-03A	Containment
Reactor Coolant Filter B	CVS-MV-03B	Containment

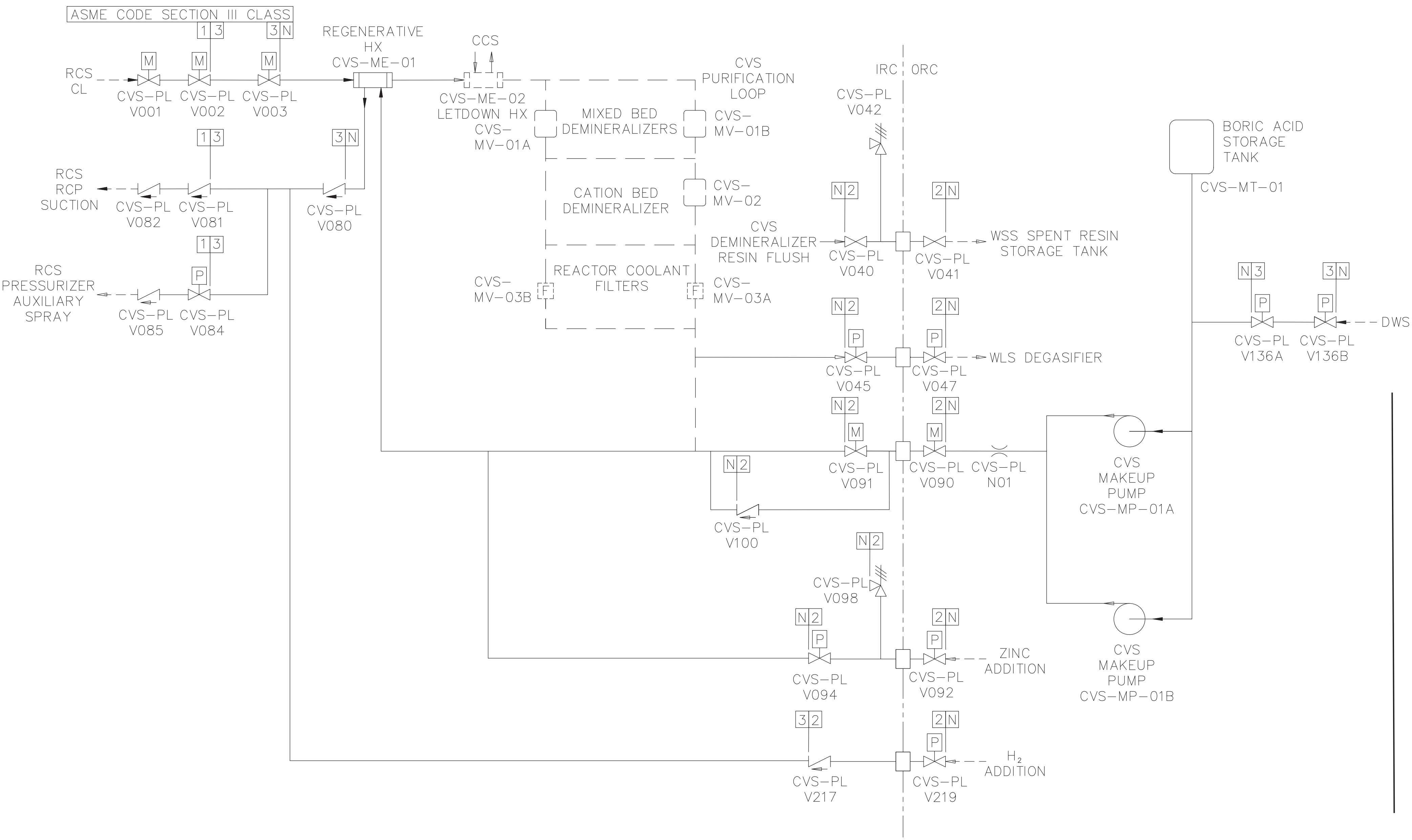


Figure 2.3.2-1
Chemical and Volume Control System

2.3.3 Standby Diesel Fuel Oil System

Design Description

The standby diesel fuel oil system (DOS) supplies diesel fuel oil for the onsite standby power system. The diesel fuel oil is supplied by two above-ground fuel oil storage tanks. The DOS also provides fuel oil for the ancillary diesel generators. A single fuel oil storage tank services both ancillary diesel generators.

The DOS is as shown in Figure 2.3.3-1 and the component locations of the DOS are as shown in Table 2.3.3-3.

1. The functional arrangement of the DOS is as described in the Design Description of this Section 2.3.3.
2. The ancillary diesel generator fuel tank can withstand a seismic event.
3. The DOS provides the following nonsafety-related functions:
   a) Each fuel oil storage tank provides for at least 7 days of continuous operation of the associated standby diesel generator.
   b) Each fuel oil day tank provides for at least four hours of continuous operation of the associated standby diesel engine generator.
   c) The fuel oil flow rate to the day tank of each standby diesel generator provides for continuous operation of the associated diesel generator.
   d) The ancillary diesel generator fuel tank is sized to supply power to long-term safety-related post-accident monitoring loads and control room lighting through a regulating transformer and one PCS recirculation pump for a period of 4 days.
4. Controls exist in the main control room (MCR) to cause the components identified in Table 2.3.3-1 to perform the listed function.
5. Displays of the parameters identified in Table 2.3.3-1 can be retrieved in the MCR.

Table 2.3.3-1

Equipment Name	Tag No.	Display	Control Function
Diesel Fuel Oil Pump 1A (Motor)	DOS-MP-01A	Yes (Run Status)	Start
Diesel Fuel Oil Pump 1B (Motor)	DOS-MP-01B	Yes (Run Status)	Start
Diesel Generator Fuel Oil Day Tank A Level	DOS-016A	Yes	-
Diesel Generator Fuel Oil Day Tank B Level	DOS-016B	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.3-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
318	2.3.03.01	1. The functional arrangement of the DOS is as described in the Design Description of this Section 2.3.3.	Inspection of the as-built system will be performed.	The as-built DOS conforms with the functional arrangement described in the Design Description of this Section 2.3.3.
319	2.3.03.02	2. The ancillary diesel generator fuel tank can withstand a seismic event.	Inspection will be performed for the existence of a report verifying that the as-built ancillary diesel generator fuel tank and its anchorage are designed using seismic Category II methods and criteria.	A report exists and concludes that the as-built ancillary diesel generator fuel tank and its anchorage are designed using seismic Category II methods and criteria.
320	2.3.03.03a	3.a) Each fuel oil storage tank provides for at least 7 days of continuous operation of the associated standby diesel generator.	Inspection of each fuel oil storage tank will be performed.	The volume of each fuel oil storage tank available to the standby diesel generator is greater than or equal to 55,000 gallons.
321	2.3.03.03b	3.b) Each fuel oil storage day tank provides for at least 4 hours of operation of the associated standby diesel generator.	Inspection of the fuel oil day tank will be performed.	The volume of each fuel oil day tank is greater than or equal to 1300 gallons.
322	2.3.03.03c	3.c) The fuel oil flow rate to the day tank of each standby diesel generator provides for continuous operation of the associated diesel generator.	Testing will be performed to determine the flow rate.	The flow rate delivered to each day tank is 8 gpm or greater.
323	2.3.03.03d	3.d) The ancillary diesel generator fuel tank is sized to supply power to long-term safety-related post accident monitoring loads and control room lighting through a regulating transformer and one PCS recirculation pump for four days.	Inspection of the ancillary diesel generator fuel tank will be performed.	The volume of the ancillary diesel generator fuel tank is greater than or equal to 650 gallons.
324	2.3.03.04	4. Controls exist in the MCR to cause the components identified in Table 2.3.3-1 to perform the listed function.	Testing will be performed on the components in Table 2.3.3-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.3.3-1 to perform the listed functions.
325	2.3.03.05	5. Displays of the parameters identified in Table 2.3.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.3-1 can be retrieved in the MCR.

Table 2.3.3-3		
Component Name	**Tag No.**	**Component Location**
Diesel Oil Transfer Package A	DOS-MS-01A	Yard
Diesel Oil Transfer Package B	DOS-MS-01B	Yard
Fuel Oil Storage Tank A	DOS-MT-01A	Yard
Fuel Oil Storage Tank B	DOS-MT-01B	Yard
Diesel Generator A Fuel Oil Day Tank	DOS-MT-02A	Diesel Building
Diesel Generator B Fuel Oil Day Tank	DOS-MT-02B	Diesel Building
Ancillary Diesel Fuel Oil Storage Tank	DOS-MT-03	Annex Building

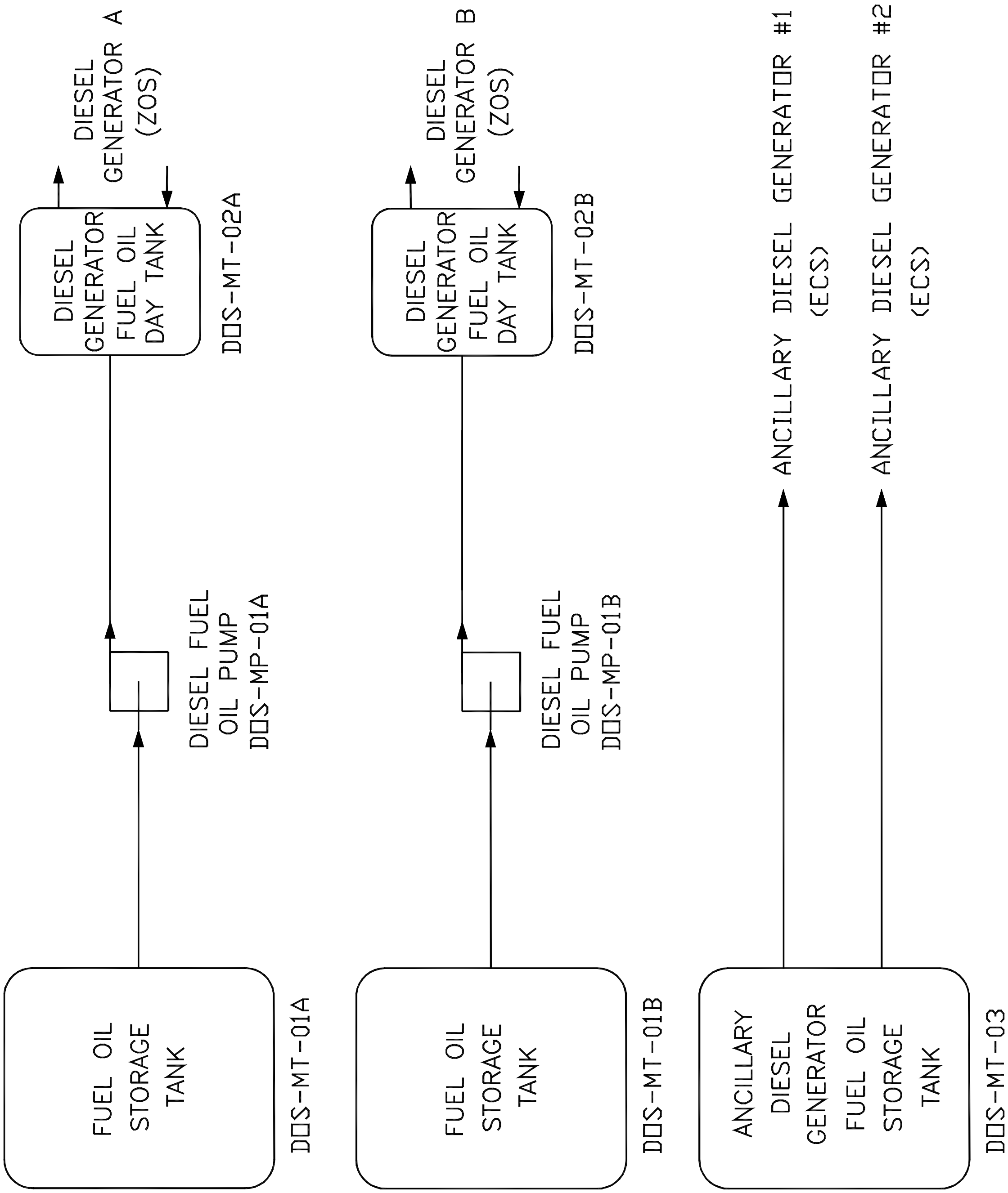


Figure 2.3.3-1
Standby Diesel Fuel Oil System

2.3.4 Fire Protection System

Design Description

The fire protection system (FPS) detects and suppresses fires in the plant. The FPS consists of water distribution systems, automatic and manual suppression systems, a fire detection and alarm system, and portable fire extinguishers. The FPS provides fire protection for the nuclear island, the annex building, the turbine building, the radwaste building and the diesel generator building.

The FPS is as shown in Figure 2.3.4-1 and the component locations of the FPS are as shown in Table 2.3.4-3.

1. The functional arrangement of the FPS is as described in the Design Description of this Section 2.3.4.
2. The FPS piping identified in Table 2.3.4-4 remains functional following a safe shutdown earthquake.
3. The FPS provides the safety-related function of preserving containment integrity by isolation of the FPS line penetrating the containment.
4. The FPS provides for manual fire fighting capability in plant areas containing safety-related equipment.
5. Displays of the parameters identified in Table 2.3.4-1 can be retrieved in the main control room (MCR).
6. The FPS provides nonsafety-related containment spray for severe accident management.
7. The FPS provides two fire water storage tanks, each capable of holding at least 300,000 gallons of water.
8. Two FPS fire pumps provide at least 2000 gpm each at a total head of at least 350 ft.
9. The fuel tank for the diesel-driven fire pump is capable of holding at least 385 gallons.
10. Individual fire detectors provide fire detection capability and can be used to initiate fire alarms in areas containing safety-related equipment.
11. The FPS seismic standpipe subsystem can be supplied from the FPS fire main by opening the normally closed cross-connect valve to the FPS plant fire main.

Table 2.3.4-1

Equipment Name	Tag No.	Display	Control Function
Motor-driven Fire Pump	FPS-MP-01A	Yes (Run Status)	Start
Diesel-driven Fire Pump	FPS-MP-01B	Yes (Run Status)	Start
Jockey Pump	FPS-MP-02	Yes (Run Status)	Start

Table 2.3.4-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
326	2.3.04.01	1. The functional arrangement of the FPS is as described in the Design Description of this Section 2.3.4.	Inspection of the as-built system will be performed.	The as-built FPS conforms with the functional arrangement described in the Design Description of this Section 2.3.4.
327	2.3.04.02.i	2. The FPS piping identified in Table 2.3.4-4 remains functional following a safe shutdown earthquake.	i) Inspection will be performed to verify that the piping identified in Table 2.3.4-4 is located on the Nuclear Island.	i) The piping identified in Table 2.3.4-4 is located on the Nuclear Island.
328	2.3.04.02.ii	2. The FPS piping identified in Table 2.3.4-4 remains functional following a safe shutdown earthquake.	ii) A reconciliation analysis using the as-designed and as-built piping information will be performed, or an analysis of the as-built piping will be performed.	ii) The as-built piping stress report exists and concludes that the piping remains functional following a safe shutdown earthquake.
329	2.3.04.03	3. The FPS provides the safety-related function of preserving containment integrity by isolation of the FPS line penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
330	2.3.04.04.i	4. The FPS provides for manual fire fighting capability in plant areas containing safety-related equipment.	i) Inspection of the passive containment cooling system (PCS) storage tank will be performed.	i) The volume of the PCS tank above the standpipe feeding the FPS and below the overflow is at least 18,000 gal.
331	2.3.04.04.ii	4. The FPS provides for manual fire fighting capability in plant areas containing safety-related equipment.	ii) Testing will be performed by measuring the water flow rate as it is simultaneously discharged from the two highest fire-hose stations and when the water for the fire is supplied from the PCS storage tank.	ii) Water is simultaneously discharged from each of the two highest fire-hose stations in plant areas containing safety-related equipment at not less than 75 gpm.
332	2.3.04.05	5. Displays of the parameters identified in Table 2.3.4-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.3.4-1 can be retrieved in the MCR.
333	2.3.04.06	6. The FPS provides nonsafety-related containment spray for severe accident management.	Inspection of the containment spray headers will be performed.	The FPS has spray headers and nozzles as follows: At least 44 nozzles at plant elevation of at least 260 feet, and 24 nozzles at plant elevation of at least 275 feet.
334	2.3.04.07	7. The FPS provides two fire water storage tanks, each capable of holding at least 300,000 gallons of water.	Inspection of each fire water storage tank will be performed.	The volume of each fire water storage tank supplying the FPS is at least 300,000 gallons.

Table 2.3.4-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
335	2.3.04.08	8. Two FPS fire pumps provide at least 2000 gpm each at a total head of at least 350 ft.	Testing and/or analysis of each fire pump will be performed.	The tests and/or analysis concludes that each fire pump provides a flow rate of at least 2000 gpm at a total head of at least 350 ft.
336	2.3.04.09	9. The fuel tank for the diesel-driven fire pump is capable of holding at least 385 gallons.	Inspection of the diesel-driven fire pump fuel tank will be performed.	The volume of the diesel driven fire pump fuel tank is at least 385 gallons.
337	2.3.04.10	10. Individual fire detectors provide fire detection capability and can be used to initiate fire alarms in areas containing safety-related equipment.	Testing will be performed on the as-built individual fire detectors in the fire areas identified in subsection 3.3, Table 3.3-3. (Individual fire detectors will be tested using simulated fire conditions.)	The tested individual fire detectors respond to simulated fire conditions.
338	2.3.04.11	11. The FPS seismic standpipe subsystem can be supplied from the FPS fire main by opening the normally closed cross-connect valve to the FPS plant fire main.	Inspection for the existence of a cross-connect valve from the FPS seismic standpipe subsystem to FPS plant fire main will be performed.	Valve FPS-PL-V101 exists and can connect the FPS seismic standpipe subsystem to the FPS plant fire main.

Table 2.3.4-3		
Component Name	**Tag No.**	**Location**
Motor-driven Fire Pump	FPS-MP-01A	Turbine Building
Diesel-driven Fire Pump	FPS-MP-01B	Yard
Jockey Pump	FPS-MP-02	Turbine Building
Primary Fire Water Tank	FPS-MT-01A	Yard
Secondary Fire Water/Clearwell Storage Tank	FPS-MT-01B	Yard
Fire Pump Diesel Fuel Day Tank	FPS-MT-02	Yard

Table 2.3.4-4 **FPS Piping Which Must Remain Functional Following a Safe Shutdown Earthquake**			
L049	L114	L142	L188
L090A	L115	L143	L189
L090B	L116	L144	L190
L091A	L117	L145	L191
L091B	L118	L146	L192
L091C	L119	L147	L193
L092A	L120	L148	L194
L092B	L121	L149	L195
L092C	L122	L150	L196
L093	L123	L151	L197
L094	L124	L152	L198
L095	L125	L153	L199
L096	L126	L154	L301
L102	L127	L155	L701
L103	L128	L156	L702
L105	L129	L159	L703
L106	L130	L180	L704
L107	L131	L181	L705
L108	L132	L182	L706
L109	L133A	L183	L707
L110	L133B	L184	L708
L111	L133C	L185	L709
L112	L140	L186	
L113	L141	L187	

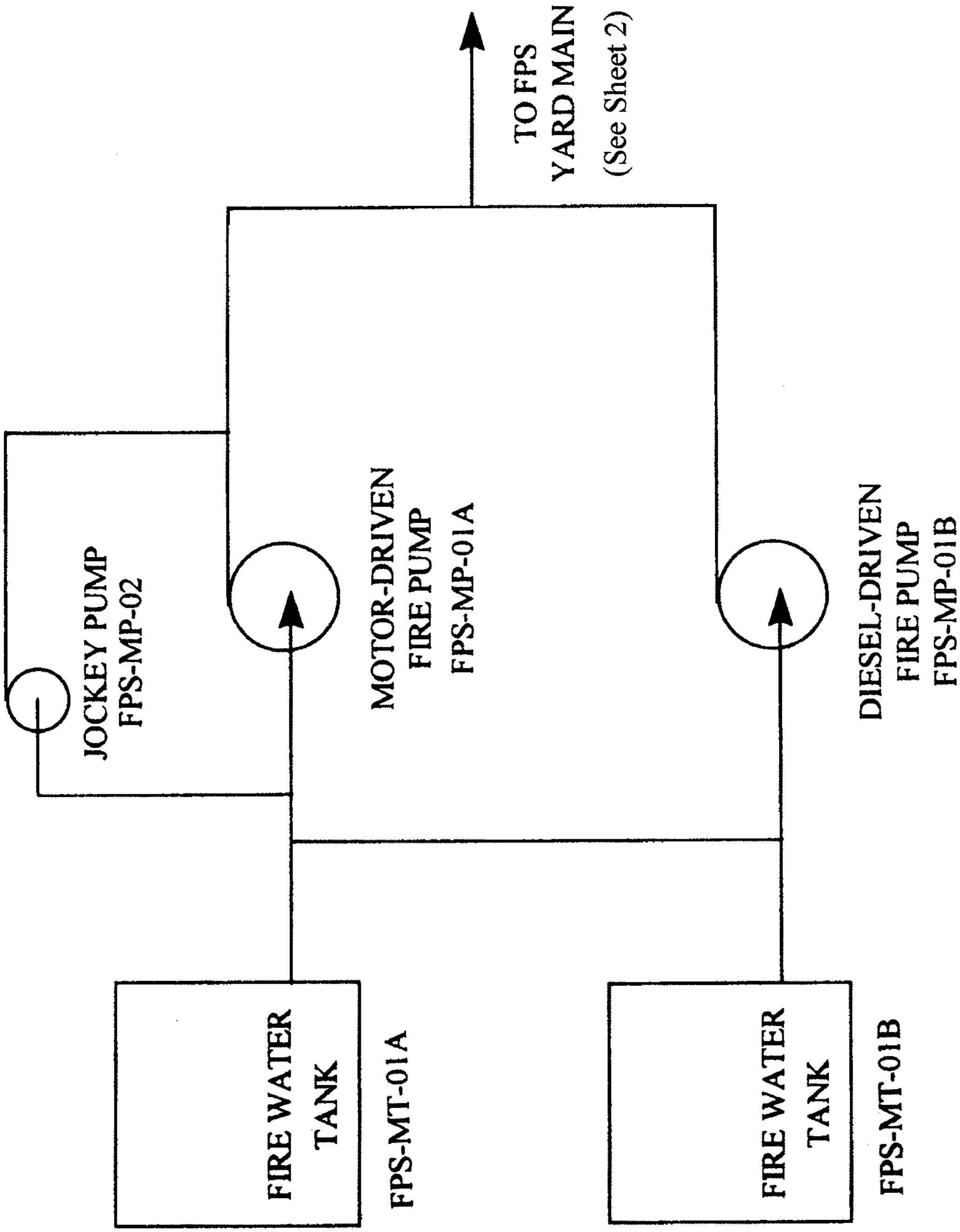


Figure 2.3.4-1 (Sheet 1 of 2)
Fire Protection System

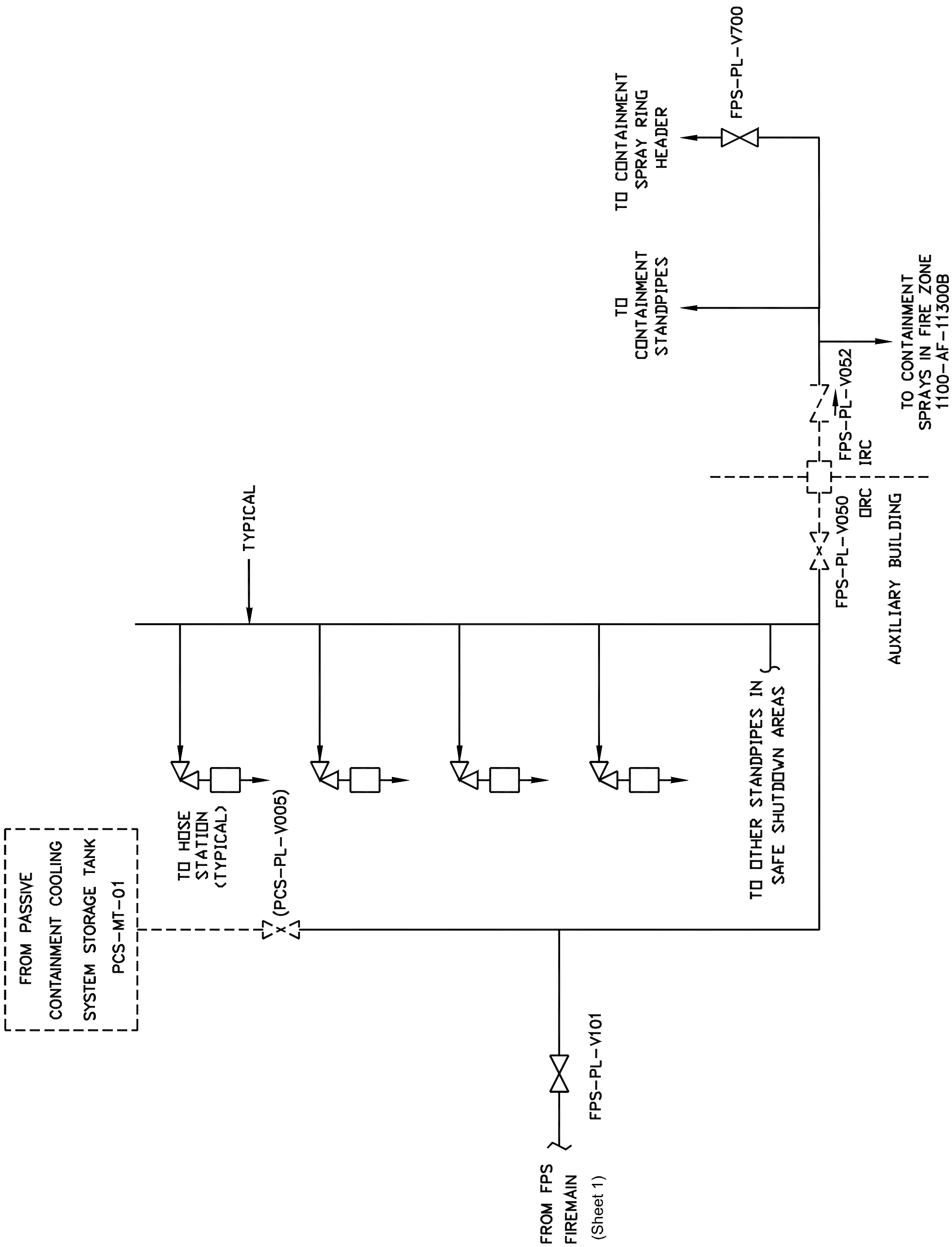


Figure 2.3.4-1 (Sheet 2 of 2)
Fire Protection System

2.3.5 Mechanical Handling System

Design Description

The mechanical handling system (MHS) provides for lifting heavy loads. The MHS equipment can be operated during shutdown and refueling.

The component locations of the MHS are as shown in Table 2.3.5-3.

1. The functional arrangement of the MHS is as described in the Design Description of this Section 2.3.5.
2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.
3. The MHS components listed below are single failure proof:
   a) Polar crane
   b) Cask handling crane
   c) Equipment hatch hoist
   d) Maintenance hatch hoist
4. The cask handling crane cannot move over the spent fuel pool.

Table 2.3.5-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety Function
Containment Polar Crane	MHS-MH-01	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Cask Handling Crane	MHS-MH-02	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Equipment Hatch Hoist	MHS-MH-05	Yes	No/No	Avoid uncontrolled lowering of heavy load.
Maintenance Hatch Hoist	MHS-MH-06	Yes	No/No	Avoid uncontrolled lowering of heavy load.

Table 2.3.5-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
339	2.3.05.01	1. The functional arrangement of the MHS is as described in the Design Description of this Section 2.3.5.	Inspection of the as-built system will be performed.	The as-built MHS conforms with the functional arrangement as described in the Design Description of this Section 2.3.5.
340	2.3.05.02.i	2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.5-1 is located on the Nuclear Island.
341	2.3.05.02.ii	2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
342	2.3.05.02.iii	2. The seismic Category I equipment identified in Table 2.3.5-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
343	2.3.05.03a.i	3.a) The polar crane is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the polar crane is single failure proof. A certificate of conformance from the vendor exists and concludes that the polar crane is single failure proof.
344	2.3.05.03a.ii	3.a) The polar crane is single failure proof.	ii) Testing of the polar crane is performed.	ii) The polar crane shall be static-load tested to 125% of the rated load.
345	2.3.05.03a.iii	3.a) The polar crane is single failure proof.	iii) Testing of the polar crane is performed.	iii) The polar crane shall lift a test load that is 100% of the rated load. Then it shall lower, stop, and hold the test load.

Table 2.3.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
346	2.3.05.03b.i	3.b) The cask handling crane is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the cask handling crane is single failure proof. A certificate of conformance from the vendor exists and concludes that the cask handling crane is single failure proof.
347	2.3.05.03b.ii	3.b) The cask handling crane is single failure proof.	ii) Testing of the cask handling crane is performed.	ii) The cask handling crane shall be static load tested to 125% of the rated load.
348	2.3.05.03b.iii	3.b) The cask handling crane is single failure proof.	iii) Testing of the cask handling crane is performed.	iii) The cask handling crane shall lift a test load that is 100% of the rated load. Then it shall lower, stop, and hold the test load.
349	2.3.05.03c.i	3.c) The equipment hatch hoist is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the equipment hatch hoist is single failure proof. A certificate of conformance from the vendor exists and concludes that the equipment hatch hoist is single failure proof.
350	2.3.05.03c.ii	3.c) The equipment hatch hoist is single failure proof.	ii) Testing of the equipment hatch hoist is performed.	ii) The equipment hatch hoist holding mechanism shall stop and hold the hatch.
351	2.3.05.03d.i	3.d) The maintenance hatch hoist is single failure proof.	i) Validation of double design factors is provided for hooks where used as load bearing components. Validation of redundant factors is provided for load bearing components such as: • Hoisting ropes • Sheaves • Equalizer assembly • Holding brakes	i) A report exists and concludes that the maintenance hatch hoist is single failure proof. A certificate of conformance from the vendor exists and concludes that the maintenance hatch hoist is single failure proof.

Table 2.3.5-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
352	2.3.05.03d.ii	3.d) The maintenance hatch hoist is single failure proof.	ii) Testing of the maintenance hatch hoist is performed.	ii) The maintenance hatch hoist holding mechanism shall stop and hold the hatch.
353	2.3.05.04	4. The cask handling crane cannot move over the spent fuel pool.	Testing of the cask handling crane is performed.	The cask handling crane does not move over the spent fuel pool.

Table 2.3.5-3		
Component Name	**Tag No.**	**Component Location**
Containment Polar Crane	MHS-MH-01	Containment
Cask Handling Crane	MHS-MH-02	Auxiliary Building
Equipment Hatch Hoist	MHS-MH-05	Containment
Maintenance Hatch Hoist	MHS-MH-06	Containment

2.3.6 Normal Residual Heat Removal System

Design Description

The normal residual heat removal system (RNS) removes heat from the core and reactor coolant system (RCS) and provides RCS low temperature over-pressure (LTOP) protection at reduced RCS pressure and temperature conditions after shutdown. The RNS also provides a means for cooling the in-containment refueling water storage tank (IRWST) during normal plant operation.

The RNS is as shown in Figure 2.3.6-1 and the RNS component locations are as shown in Table 2.3.6-5.

1. The functional arrangement of the RNS is as described in the Design Description of this Section 2.3.6.
2. a) The components identified in Table 2.3.6-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
   b) The piping identified in Table 2.3.6-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.3.6-1 as ASME Code Section III meet ASME Code Section III requirements.
   b) Pressure boundary welds in piping identified in Table 2.3.6-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.3.6-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.6-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. a) The seismic Category I equipment identified in Table 2.3.6-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.3.6-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

6. Each of the as-built lines identified in Table 2.3.6-2 as designed for leak before break (LBB) meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.

7. a) The Class 1E equipment identified in Table 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.3.6-1 are powered from their respective Class 1E division.

   c) Separation is provided between RNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

8. The RNS provides the following safety-related functions:

   a) The RNS preserves containment integrity by isolation of the RNS lines penetrating the containment.

   b) The RNS provides a flow path for long-term, post-accident makeup to the RCS.

9. The RNS provides the following nonsafety-related functions:

   a) The RNS provides low temperature overpressure protection (LTOP) for the RCS during shutdown operations.

   b) The RNS provides heat removal from the reactor coolant during shutdown operations.

   c) The RNS provides low pressure makeup flow from the SFS cask loading pit to the RCS for scenarios following actuation of the automatic depressurization system (ADS).

   d) The RNS provides heat removal from the in-containment refueling water storage tank.

10. Safety-related displays identified in Table 2.3.6-1 can be retrieved in the main control room (MCR).

11. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.6-1 to perform active functions.

    b) The valves identified in Table 2.3.6-1 as having protection and safety monitoring system (PMS) control perform active safety functions after receiving a signal from the PMS.

12. a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.

    b) After loss of motive power, the remotely operated valves identified in Table 2.3.6-1 assume the indicated loss of motive power position.

13. Controls exist in the MCR to cause the pumps identified in Table 2.3.6-3 to perform the listed function.

14. Displays of the RNS parameters identified in Table 2.3.6-3 can be retrieved in the MCR.

Table 2.3.6-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E / Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
RNS Pump A (Pressure Boundary)	RNS-MP-01A	Yes	Yes	-	-/-	-	-	No	-
RNS Pump B (Pressure Boundary)	RNS-MP-01B	Yes	Yes	-	-/-	-	-	No	-
RNS Heat Exchanger A (Tube Side)	RNS-ME-01A	Yes	Yes	-	-/-	-	-	-	-
RNS Heat Exchanger B (Tube Side)	RNS-ME-01B	Yes	Yes	-	-/-	-	-	-	-
RCS Inner Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V001A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Inner Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V001B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Outer Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V002A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Outer Hot Leg Suction Motor-operated Isolation Valve	RNS-PL-V002B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is

Note: Dash (-) indicates not applicable.

Table 2.3.6-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
RCS Pressure Boundary Thermal Relief Check Valve	RNS-PL-V003A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RCS Pressure Boundary Thermal Relief Check Valve	RNS-PL-V003B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge Motor-operated Containment Isolation Valve	RNS-PL-V011	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
RNS Discharge Containment Isolation Test Connection	RNS-PL-V012	Yes	Yes	No	-/-	No	No	Transfer Open/ Transfer Closed	-
RNS Discharge Header Containment Isolation Check Valve	RNS-PL-V013	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V015A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V015B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-

Table 2.3.6-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V017A	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Discharge RCS Pressure Boundary Check Valve	RNS-PL-V017B	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Hot Leg Suction Pressure Relief Valve	RNS-PL-V021	Yes	Yes	No	-/-	No	-	Transfer Open/ Transfer Closed	-
RNS Suction Header Motor-operated Containment Isolation Valve	RNS-PL-V022	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
RNS Suction from IRWST Motor-operated Isolation Valve	RNS-PL-V023	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RNS Discharge to IRWST Motor-operated Isolation Valve	RNS-PL-V024	Yes	Yes	Yes	-/-	No	No	No	As Is
RNS Pump Discharge Relief	RNS-PL-V045	Yes	Yes	No	-/-	No	-	No	-
RNS Suction from Cask Loading Pit Motor-operated Isolation Valve	RNS-PL-V055	Yes	Yes	Yes	No/No	No	No	No	As Is

Table 2.3.6-1 (cont.)									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
RNS Suction from Cask Loading Pit Check Valve	RNS-PL-V056	Yes	Yes	No	-/-	No	-	No	-
RNS Pump Miniflow Air-Operated Isolation Valve	RNS-PL-V057A	Yes	Yes	Yes	No/No	No	No	No	Open
RNS Pump Miniflow Air-Operated Isolation Valve	RNS-PL-V057B	Yes	Yes	Yes	No/No	No	No	No	Open
RNS Return from Chemical and Volume Control System (CVS) Containment Isolation Valve	RNS-PL-V061	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

Table 2.3.6-2				
Line Name	**Line No.**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
RNS Suction Lines, from the RCS Hot Leg Connection to the RCS Side of Valves RNS PL-V001A and RNS-PL-V001B	RNS-L001 RNS-L002A RNS-L002B	Yes	Yes	No
RNS Suction Lines, from the RCS Pressure Boundary Valves, RNS-PL-V001A and RNS-PL-V001B, to the RNS pumps	RNS-L004A RNS-L004B RNS-L005 RNS-L006 RNS-L007A RNS-L007B RNS-L009A RNS-L009B	Yes	No	Yes Yes Yes Yes Yes Yes Yes Yes
RNS Suction Line from CVS	RNS-L061	Yes	No	Yes
RNS Suction Line from IRWST	RNS-L029	Yes	No	Yes
RNS Suction Line LTOP Relief	RNS-L040	Yes	No	Yes
RNS Discharge Lines, from the RNS Pumps to the RNS Heat Exchangers RNS-ME-01A and RNS-ME-01B	RNS-L011A RNS-L011B	Yes	No	Yes
RNS Discharge Lines, from RNS Heat Exchanger RNS-ME-01A to Containment Isolation Valve RNS-PL-V011	RNS-L012A RNS-L014	Yes	No	Yes
RNS Discharge Line, from RNS Heat Exchanger RNS-ME-01B to Common Discharge Header RNS-L014	RNS-L012B	Yes	No	Yes
RNS Discharge Lines, Containment Isolation Valve RNS-PL-V011 to Containment Isolation Valve RNS-PL-V013	RNS-L016	Yes	No	Yes
RNS Suction Line from Cask Loading Pit	RNS-L065	Yes	No	No
RNS Discharge Lines, from Containment Isolation Valve RNS-PL-V013 to RCS Pressure Boundary Isolation Valves RNS-PL-V015A and RNS-PL-V015B	RNS-L017 RNS-L018A RNS-L018B	Yes	No	Yes

Table 2.3.6-2				
Line Name	**Line No.**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
RNS Discharge Lines, from Direct Vessel Injection (DVI) Line RNS-BBC-L018A to Passive Core Cooling System (PXS) IRWST Return Isolation Valve RNS-PL-V024	RNS-L020	Yes	No	No
RNS Discharge Lines, from RCS Pressure Boundary Isolation Valves RNS-PL-V015A and RNS-PL-V015B to RCS Pressure Boundary Isolation Valves RNS-PL-V017A and RNS-PL-V017B	RNS-L019A RNS-L019B	Yes	No	Yes
RNS Heat Exchanger Bypass	RNS-L008A RNS-L008B	Yes	No	No
RNS Suction from Spent Fuel Pool	RNS-L052	Yes	No	No
RNS Pump Miniflow Return	RNS-L030A RNS-L030B	Yes	No	No
RNS Discharge to Spent Fuel Pool	RNS-L051	Yes	No	No
RNS Discharge to CVS Purification	RNS-L021	Yes	No	No

Table 2.3.6-3			
Equipment Name	**Tag No.**	**Display**	**Control Function**
RNS Pump 1A (Motor)	RNS-MP-01A	Yes (Run Status)	Start
RNS Pump 1B (Motor)	RNS-MP-01B	Yes (Run Status)	Start
RNS Flow Sensor	RNS-01A	Yes	-
RNS Flow Sensor	RNS-01B	Yes	-
RNS Suction from Cask Loading Pit Isolation Valve (Position Indicator)	RNS-PL-V055	Yes	-
RNS Pump Miniflow Isolation Valve (Position Indicator)	RNS-PL-V057A	Yes	-
RNS Pump Miniflow Isolation Valve (Position Indicator)	RNS-PL-V057B	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
354	2.3.06.01	1. The functional arrangement of the RNS is as described in the Design Description of this Section 2.3.6.	Inspection of the as-built system will be performed.	The as-built RNS conforms with the functional arrangement described in the Design Description of this Section 2.3.6.
355	2.3.06.02a	2.a) The components identified in Table 2.3.6-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.3.6-1 as ASME Code Section III.
356	2.3.06.02b	2.b) The piping identified in Table 2.3.6-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.3.6-2 as ASME Code Section III.
357	2.3.06.03a	3.a) Pressure boundary welds in components identified in Table 2.3.6-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
358	2.3.06.03b	3.b) Pressure boundary welds in piping identified in Table 2.3.6-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
359	2.3.06.04a	4.a) The components identified in Table 2.3.6-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.6-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
360	2.3.06.04b	4.b) The piping identified in Table 2.3.6-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.3.6-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
361	2.3.06.05a.i	5.a) The seismic Category I equipment identified in Table 2.3.6-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.6-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.6-1 is located on the Nuclear Island.
362	2.3.06.05a.ii	5.a) The seismic Category I equipment identified in Table 2.3.6-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
363	2.3.06.05a.iii	5.a) The seismic Category I equipment identified in Table 2.3.6-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
364	2.3.06.05b	5.b) Each of the lines identified in Table 2.3.6-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.3.6-2 for which functional capability is required meets the requirements for functional capability.
365	2.3.06.06	6. Each of the as-built lines identified in Table 2.3.6-2 as designed for LBB meets the LBB criteria, or an evaluation is performed of the protection from the dynamic effects of a rupture of the line.	Inspection will be performed for the existence of an LBB evaluation report or an evaluation report on the protection from dynamic effects of a pipe break. Section 3.3, Nuclear Island Buildings, contains the design descriptions and inspections, tests, analyses, and acceptance criteria for protection from the dynamic effects of pipe rupture.	An LBB evaluation report exists and concludes that the LBB acceptance criteria are met by the as-built RNS piping and piping materials, or a pipe break evaluation report exists and concludes that protection from the dynamic effects of a line break is provided.
366	2.3.06.07a.i	7.a) The Class 1E equipment identified in Tables 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
367	2.3.06.07a.ii	7.a) The Class 1E equipment identified in Tables 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.3.6-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
368	2.3.06.07b	7.b) The Class 1E components identified in Table 2.3.6-1 are powered from their respective Class 1E division.	Testing will be performed on the RNS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.6-1 when the assigned Class 1E division is provided the test signal.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
369	2.3.06.07c	7.c) Separation is provided between RNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
370	2.3.06.08a	8.a) The RNS preserves containment integrity by isolation of the RNS lines penetrating the containment.	See ITAAC Table 2.2.1-3, item 7.	See ITAAC Table 2.2.1-3, item 7.
371	2.3.06.08b	8.b) The RNS provides a flow path for long-term, post-accident makeup to the RCS.	See item 1 in this table.	See item 1 in this table.
372	2.3.06.09a.i	9.a) The RNS provides LTOP for the RCS during shutdown operations.	i) Inspections will be conducted on the low temperature overpressure protection relief valve to confirm that the capacity of the vendor code plate rating is greater than or equal to system relief requirements.	i) The rated capacity recorded on the valve vendor code plate is not less than the flow required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material.
373	2.3.06.09a.ii	9.a) The RNS provides LTOP for the RCS during shutdown operations.	ii) Testing and analysis in accordance with the ASME Code Section III will be performed to determine set pressure.	ii) A report exists and concludes that the relief valve opens at a pressure not greater than the set pressure required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the pressure-temperature curves developed for the as-procured reactor vessel material.
374	2.3.06.09b.i	9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	i) Inspection will be performed for the existence of a report that determines the heat removal capability of the RNS heat exchangers.	i) A report exists and concludes that the product of the overall heat transfer coefficient and the effective heat transfer area, UA, of each RNS heat exchanger is greater than or equal to 2.2 million Btu/hr-°F.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
375	2.3.06.09b.ii	9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	ii) Testing will be performed to confirm that the RNS can provide flow through the RNS heat exchangers when the pump suction is aligned to the RCS hot leg and the discharge is aligned to both PXS DVI lines with the RCS at atmospheric pressure.	ii) Each RNS pump provides at least 1400 gpm net flow to the RCS when the hot leg water level is at an elevation 15.5 inches ± 2 inches above the bottom of the hot leg.
376	2.3.06.09b.iii	9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	iii) Inspection will be performed of the reactor coolant loop piping.	iii) The RCS cold legs piping centerline is 17.5 inches ± 2 inches above the hot legs piping centerline.
377	2.3.06.09b.iv	9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	iv) Inspection will be performed of the RNS pump suction piping.	iv) The RNS pump suction piping from the hot leg to the pump suction piping low point does not form a local high point (defined as an upward slope with a vertical rise greater than 3 inches).
378	2.3.06.09b.v	9.b) The RNS provides heat removal from the reactor coolant during shutdown operations.	v) Inspection will be performed of the RNS pump suction nozzle connection to the RCS hot leg.	v) The RNS suction line connection to the RCS is constructed from 20-inch Schedule 140 pipe.
379	2.3.06.09c	9.c) The RNS provides low pressure makeup flow from the cask loading pit to the RCS for scenarios following actuation of the ADS.	Testing will be performed to confirm that the RNS can provide low pressure makeup flow from the cask loading pit to the RCS when the pump suction is aligned to the cask loading pit and the discharge is aligned to both PXS DVI lines with RCS at atmospheric pressure.	Each RNS pump provides at least 1100 gpm net flow to the RCS when the water level above the bottom of the cask loading pit is 1 foot ± 6 inches.
380	2.3.06.09d	9.d) The RNS provides heat removal from the in-containment refueling water storage tank (IRWST).	Testing will be performed to confirm that the RNS can provide flow through the RNS heat exchangers when the pump suction is aligned to the IRWST and the discharge is aligned to the IRWST.	Two operating RNS pumps provide at least 2000 gpm to the IRWST.
381	2.3.06.10	10. Safety-related displays identified in Table 2.3.6-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.6-1 can be retrieved in the MCR.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
382	2.3.06.11a	11.a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.6-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.6-1 using the controls in the MCR.	Controls in the MCR operate to cause those remotely operated valves identified in Table 2.3.6-1 to perform active functions.
383	2.3.06.11b	11.b) The valves identified in Table 2.3.6-1 as having PMS control perform active safety functions after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The valves identified in Table 2.3.6-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
384	2.3.06.12a.i	12.a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.3.6-1 under design conditions.
385	2.3.06.12a.ii	12.a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.	ii) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tested conditions.	ii) A report exists and concludes that the as-built motor-operated valves are bounded by the tested conditions.
386	2.3.06.12a.iii	12.a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.	iii) Tests of the motor-operated valves will be performed under preoperational flow, differential pressure and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.3.6-1 under preoperational test conditions.
387	2.3.06.12a.iv	12.a) The motor-operated and check valves identified in Table 2.3.6-1 perform an active safety-related function to change position as indicated in the table.	iv) Exercise testing of the check valves active safety functions identified in Table 2.3.6-1 will be performed under preoperational test pressure, temperature and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.3.6-1.
388	2.3.06.12b	12.b) After loss of motive power, the remotely operated valves identified in Table 2.3.6-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.3.6-1 assumes the indicated loss of motive power position.
389	2.3.06.13	13. Controls exist in the MCR to cause the pumps identified in Table 2.3.6-3 to perform the listed function.	Testing will be performed to actuate the pumps identified in Table 2.3.6-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.6-3 to perform the listed action.

Table 2.3.6-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
390	2.3.06.14	14. Displays of the RNS parameters identified in Table 2.3.6-3 can be retrieved in the MCR.	Inspection will be performed for retrievability in the MCR of the displays identified in Table 2.3.6-3.	Displays of the RNS parameters identified in Table 2.3.6-3 are retrieved in the MCR.

Table 2.3.6-5		
Component Name	**Tag No.**	**Component Location**
RNS Pump A	RNS-MP-01A	Auxiliary Building
RNS Pump B	RNS-MP-01B	Auxiliary Building
RNS Heat Exchanger A	RNS-ME-01A	Auxiliary Building
RNS Heat Exchanger B	RNS-ME-01B	Auxiliary Building

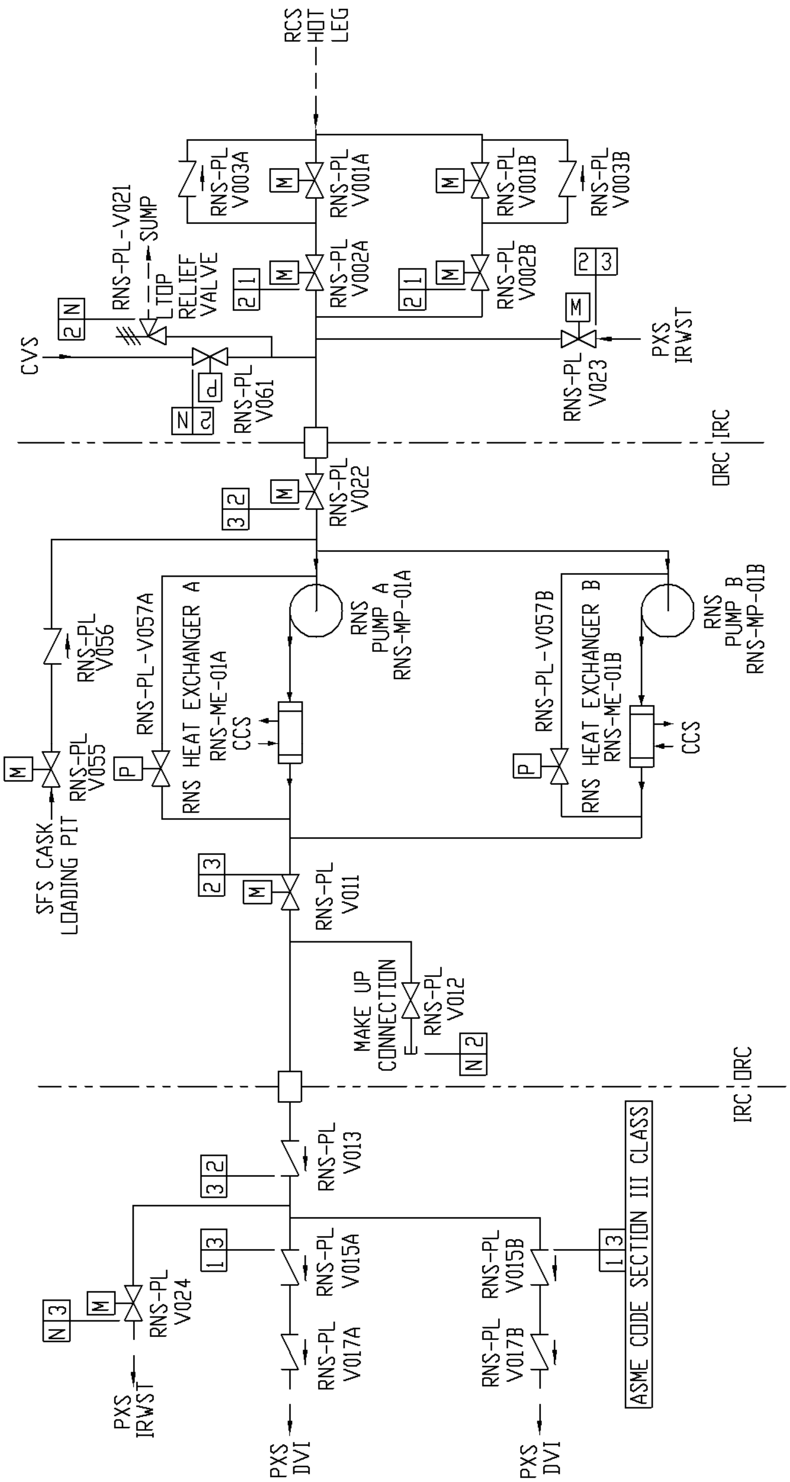


Figure 2.3.6-1
Normal Residual Heat Removal System

2.3.7 Spent Fuel Pool Cooling System

Design Description

The spent fuel pool cooling system (SFS) removes decay heat from spent fuel by transferring heat from the water in the spent fuel pool to the component cooling water system during normal modes of operation. The SFS purifies the water in the spent fuel pool, fuel transfer canal, and in-containment refueling water storage tank during normal modes of operation. Following events such as earthquakes, or fires, if the normal heat removal method is not available, decay heat is removed from spent fuel by boiling water in the pool. In the event of long-term station blackout, makeup water is supplied to the spent fuel pool from onsite storage tanks.

The SFS is as shown in Figure 2.3.7-1 and the component locations of the SFS are as shown in Table 2.3.7-5.

1. The functional arrangement of the SFS is as described in the Design Description of this Section 2.3.7.
2. a) The components identified in Table 2.3.7-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
   b) The piping lines identified in Table 2.3.7-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
3. Pressure boundary welds in piping lines identified in Table 2.3.7-2 as ASME Code Section III meet ASME Code Section III requirements.
4. The piping lines identified in Table 2.3.7-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
5. The seismic Category I components identified in Table 2.3.7-1 can withstand seismic design basis loads without loss of safety function.
6. a) The Class 1E components identified in Table 2.3.7-1 are powered from their respective Class 1E division.
   b) Separation is provided between SFS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.
7. The SFS performs the following safety-related functions:
   a) The SFS preserves containment integrity by isolating the SFS piping lines penetrating the containment.
   b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site water storage tanks.
   c) The SFS provides check valves in the drain line from the refueling cavity to prevent flooding of the refueling cavity during containment flooding.
8. The SFS provides the nonsafety-related function of removing spent fuel decay heat using pumped flow through a heat exchanger.

9. Safety-related displays identified in Table 2.3.7-1 can be retrieved in the main control room (MCR).
10. Controls exist in the MCR to cause the pumps identified in Table 2.3.7-3 to perform their listed functions.
11. Displays of the SFS parameters identified in Table 2.3.7-3 can be retrieved in the MCR.

Table 2.3.7-1									
Component Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat 1**	**Remotely Operated Valve**	**Class 1E/ Qual for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Spent Fuel Pool Level Sensor	SFS-019A	No	Yes	-	Yes/No	Yes	-	-	-
Spent Fuel Pool Level Sensor	SFS-019B	No	Yes	-	Yes/No	Yes	-	-	-
Spent Fuel Pool Level Sensor	SFS-019C	No	Yes	-	Yes/No	Yes	-	-	-
Refueling Cavity Drain to SGS Compartment Isolation Valve	SFS-PL-V031	Yes	Yes	No	-/-	Yes	-	-	-
Refueling Cavity to SFS Pump Suction Isolation Valve	SFS-PL-V032	Yes	Yes	No	-/-	No	-	-	-
Refueling Cavity Drain to Containment Sump Isolation Valve	SFS-PL-V033	Yes	Yes	No	-/-	Yes	-	-	-
IRWST to SFS Pump Suction Line Isolation Valve	SFS-PL-V039	Yes	Yes	No	-/-	No	-	-	-
Fuel Transfer Canal to SFS Pump Suction Iso. Valve	SFS-PL-V040	Yes	Yes	No	-/-	No	-	-	-
Cask Loading Pit to SFS Pump Suction Isolation Valve	SFS-PL-V041	Yes	Yes	No	-/-	No	-	-	-

Note: Dash (-) indicates not applicable.

Table 2.3.7-1 (cont.)									
Component Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat 1**	**Remotely Operated Valve**	**Class 1E/ Qual for Harsh Envir.**	**Safety-Related Display**	**Control PMS**	**Active Function**	**Loss of Motive Power Position**
Cask Loading Pit to SFS Pump Suction Isolation Valve	SFS-PL-V042	Yes	Yes	No	-/-	No	-	Transfer Closed	-
SFS Pump Discharge Line to Cask Loading Pit Isolation Valve	SFS-PL-V045	Yes	Yes	No	-/-	No	-	Transfer Closed	-
Cask Loading Pit to WLS Isolation Valve	SFS-PL-V049	Yes	Yes	No	-/-	No	-	Transfer Closed	-
Spent Fuel Pool to Cask Washdown Pit Isolation Valve	SFS-PL-V066	Yes	Yes	No	-/-	No	-	Transfer Open	-
Cask Washdown Pit Drain Isolation Valve	SFS-PL-V068	Yes	Yes	No	-/-	No	-	Transfer Open	-
Refueling Cavity Drain Line Check Valve	SFS-PL-V071	Yes	Yes	No	-/-	No	-	Transfer Open Transfer Closed	-
Refueling Cavity Drain Line Check Valve	SFS-PL-V072	Yes	Yes	No	-/-	No	-	Transfer Open Transfer Closed	-
SFS Containment Floodup Isolation Valve	SFS-PL-V075	Yes	Yes	No	-/-	Yes	-	-	-

Note: Dash (-) indicates not applicable.

Table 2.3.7-2		
Piping Line Name	**Line Number**	**ASME Code Section III**
Spent Fuel Pool to RNS Pump Suction	L014	Yes
Cask Loading Pit to RNS Pump Suction	L115	Yes
Refueling Cavity Drain	L033	Yes
PXS IRWST to SFS Pump Suction	L035	Yes
Refueling Cavity Skimmer to SFS Pump Suction	L036	Yes
Refueling Cavity Drain	L037	Yes
Refueling Cavity Drain	L044	Yes
Fuel Transfer Canal Drain	L047	Yes
Cask Washdown Pit Drain	L068	Yes
Cask Loading Pit Drain	L043	Yes
Cask Pit Transfer Branch Line	L045	Yes
Spent Fuel Pool Containment Isolation Thermal Relief Line	L052	Yes
Refueling Cavity Drain	L030	Yes
Upender Pit Drain/Fill Line	L121	Yes
Spent Fuel Pool Drain	L066	Yes
Cask Loading Pit to WLS	L067	Yes
RNS Return to Spent Fuel Pool	L100	Yes
SFS Containment Floodup Line	L120	Yes

Table 2.3.7-3			
Component Name	**Tag No.**	**Display**	**Control Function**
SFS Pump 1A	SFS-MP-01A	Yes (Run Status)	Start
SFS Pump 1B	SFS-MP-01B	Yes (Run Status)	Start
SFS Flow Sensor	SFS-13A	Yes	-
SFS Flow Sensor	SFS-13B	Yes	-
Spent Fuel Pool Temperature Sensor	SFS-018	Yes	-
Cask Loading Pit Level Sensor	SFS-022	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.7-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
391	2.3.07.01	1. The functional arrangement of the SFS is as described in the Design Description of this Section 2.3.7.	Inspection of the as-built system will be performed.	The as-built SFS conforms with the functional arrangement as described in the Design Description of this Section 2.3.7.
392	2.3.07.02a	2.a) The components identified in Table 2.3.7-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the ASME as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.3.7-1 as ASME Code Section III.
393	2.3.07.02b	2.b) The piping lines identified in Table 2.3.7-2 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping lines as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping lines identified in Table 2.3.7-2 as ASME Code Section III.
394	2.3.07.03	3. Pressure boundary welds in piping lines identified in Table 2.3.7-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
395	2.3.07.04	4. The piping lines identified in Table 2.3.7-2 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the piping lines required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping lines identified in Table 2.3.7-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
396	2.3.07.05.i	5. The seismic Category I components identified in Table 2.3.7-1 can withstand seismic design basis loads without loss of safety functions.	i) Inspection will be performed to verify that the seismic Category I components identified in Table 2.3.7-1 are located on the Nuclear Island.	i) The seismic Category I components identified in Table 2.3.7-1 are located on the Nuclear Island.
397	2.3.07.05.ii	5. The seismic Category I components identified in Table 2.3.7-1 can withstand seismic design basis loads without loss of safety functions.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.

Table 2.3.7-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
398	2.3.07.05.iii	5. The seismic Category I components identified in Table 2.3.7-1 can withstand seismic design basis loads without loss of safety functions.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
399	2.3.07.06a	6.a) The Class 1E components identified in Table 2.3.7-1 are powered from their respective Class 1E division.	Testing will be performed on the SFS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E components identified in Table 2.3.7-1 when the assigned Class 1E division is provided the test signal.
400	2.3.07.06b	6.b) Separation is provided between SFS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
401	2.3.07.07a	7.a) The SFS preserves containment integrity by isolation of the SFS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
402	2.3.07.07b.i	7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	i) Inspection will be performed to verify that the spent fuel pool includes a sufficient volume of water.	i) The volume of the spent fuel pool and fuel transfer canal above the fuel and to the elevation 6 feet below the operating deck is greater than or equal to 129,500 gallons.
403	2.3.07.07b.ii	7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	ii) Inspection will be performed to verify the cask washdown pit includes sufficient volume of water.	ii) The water volume of the cask washdown pit is greater than or equal to 30,900 gallons.
404	2.3.07.07b.iii	7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	iii) A safety-related flow path exists from the cask washdown pit to the spent fuel pool.	iii) See item 1 of this table.
405	2.3.07.07b.iv	7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	iv) See ITAAC Table 2.2.2-3, item 7.f for inspection, testing, and acceptance criteria for the makeup water supply from the passive containment cooling system (PCS) water storage tank to the spent fuel pool.	iv) See ITAAC Table 2.2.2-3, item 7.f for inspection, testing, and acceptance criteria for the makeup water supply from the PCS water storage tank to the spent fuel pool.

Table 2.3.7-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
406	2.3.07.07b.v	7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	v) Inspection will be performed to verify that the passive containment cooling system water storage tank includes a sufficient volume of water.	v) See ITAAC Table 2.2.2-3, item 7.f for the volume of the passive containment cooling system water storage tank.
407	2.3.07.07b.vi	7.b) The SFS provides spent fuel cooling for 7 days by boiling the spent fuel pool water and makeup water from on-site storage tanks.	vi) See ITAAC Table 2.2.2-3, items 8.a and 8.b for inspection, testing, and acceptance criteria to verify that the passive containment cooling system ancillary water storage tank includes a sufficient volume of water.	vi) See ITAAC Table 2.2.2-3, items 8.a and 8.b for inspection, testing, and acceptance criteria for the volume of the passive containment cooling system ancillary water storage tank.
408	2.3.07.07c	7c) The SFS provides check valves in the drain line from the refueling cavity to prevent flooding of the refueling cavity during containment flooding.	Exercise testing of the check valves with active safety-functions identified in Table 2.3.7-1 will be performed under pre-operational test pressure, temperature and flow conditions.	Each check valve changes position as indicated on Table 2.3.7-1.
409	2.3.07.08.i	8. The SFS provides the nonsafety-related function of removing spent fuel decay heat using pumped flow through a heat exchanger.	i) Inspection will be performed for the existence of a report that determines the heat removal capability of the SFS heat exchangers.	i) A report exists and concludes that the heat transfer characteristic, UA, of each SFS heat exchanger is greater than or equal to 2.2 million Btu/hr-°F.
410	2.3.07.08.ii	8. The SFS provides the nonsafety-related function of removing spent fuel decay heat using pumped flow through a heat exchanger.	ii) Testing will be performed to confirm that each SFS pump provides flow through its heat exchanger when taking suction from the SFP and returning flow to the SFP.	ii) Each SFS pump produces at least 900 gpm through its heat exchanger.
411	2.3.07.09	9. Safety-related displays identified in Table 2.3.7-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.7-1 can be retrieved in the MCR.
412	2.3.07.10	10. Controls exist in the MCR to cause the pumps identified in Table 2.3.7-3 to perform their listed functions.	Testing will be performed to actuate the pumps identified in Table 2.3.7-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.7-3 to perform the listed functions.
413	2.3.07.11	11. Displays of the SFS parameters identified in Table 2.3.7-3 can be retrieved in the MCR.	Inspection will be performed for retrievability in the MCR of the displays identified in Table 2.3.7-3.	Displays of the SFS parameters identified in Table 2.3.7-3 are retrieved in the MCR.

Table 2.3.7-5		
Component Name	**Tag No.**	**Component Location**
SFS Pump A	SFS-MP-01A	Auxiliary Building
SFS Pump B	SFS-MP-01B	Auxiliary Building
SFS Heat Exchanger A	SFS-ME-01A	Auxiliary Building
SFS Heat Exchanger B	SFS-ME-01B	Auxiliary Building

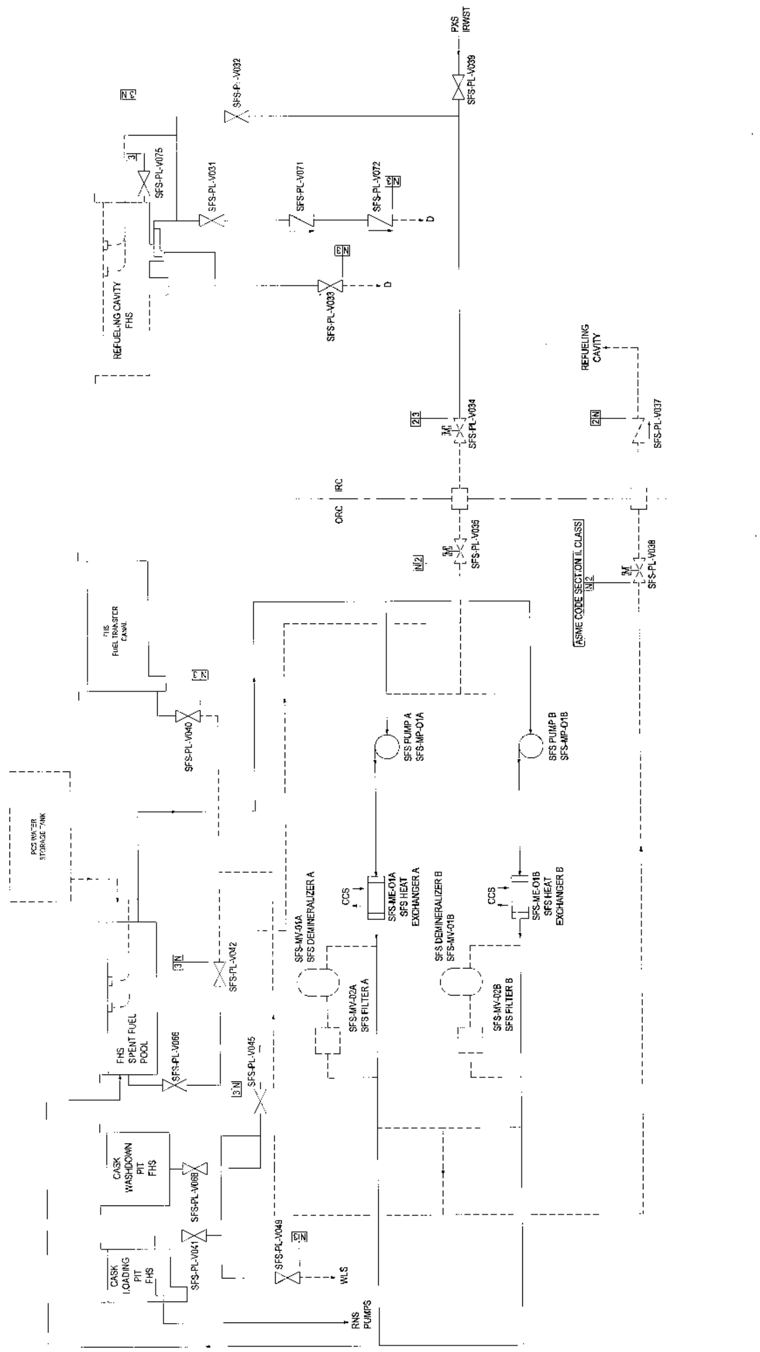


Figure 2.3.7-1
Spent Fuel Pool Cooling System

2.3.8 Service Water System

Design Description

The service water system (SWS) transfers heat from the component cooling water heat exchangers to the atmosphere. The SWS operates during normal modes of plant operation, including startup, power operation (full and partial loads), cooldown, shutdown, and refueling.

The SWS is as shown in Figure 2.3.8-1 and the component locations of the SWS are as shown Table 2.3.8-3.

1. The functional arrangement of the SWS is as described in the Design Description of this Section 2.3.8.
2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system (CCS) to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.
3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.3.8-1 to perform the listed function.
4. Displays of the parameters identified in Table 2.3.8-1 can be retrieved in the MCR.

Table 2.3.8-1

Equipment Name	Tag No.	Display	Control Function
Service Water Pump A (Motor)	SWS-MP-01A	Yes (Run Status)	Start
Service Water Pump B (Motor)	SWS-MP-01B	Yes (Run Status)	Start
Service Water Cooling Tower Fan A (Motor)	SWS-MA-01A	Yes (Run Status)	Start
Service Water Cooling Tower Fan B (Motor)	SWS-MA-01B	Yes (Run Status)	Start
Service Water Pump 1A Flow Sensor	SWS-004A	Yes	-
Service Water Pump 1B Flow Sensor	SWS-004B	Yes	-
Service Water Pump A Discharge Valve	SWS-PL-V002A	Yes (Valve Position)	Open
Service Water Pump B Discharge Valve	SWS-PL-V002B	Yes (Valve Position)	Open

Table 2.3.8-1			
Equipment Name	**Tag No.**	**Display**	**Control Function**
Service Water Pump A Discharge Temperature Sensor	SWS-005A	Yes	-
Service Water Pump B Discharge Temperature Sensor	SWS-005B	Yes	-
Service Water Cooling Tower Basin Level	SWS-009	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.8-2

Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
414	2.3.08.01	1. The functional arrangement of the SWS is as described in the Design Description of this Section 2.3.8.	Inspection of the as-built system will be performed.	The as-built SWS conforms with the functional arrangement as described in the Design Description of this Section 2.3.8.
415	2.3.08.02.i	2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.	i) Testing will be performed to confirm that the SWS can provide cooling water to the CCS heat exchangers.	i) Each SWS pump can provide at least 10,000 gpm of cooling water through its CCS heat exchanger.
416	2.3.08.02.ii	2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.	ii) Inspection will be performed for the existence of a report that determines the heat transfer capability of each cooling tower cell.	ii) A report exists and concludes that the heat transfer rate of each cooling tower cell is greater than or equal to 170 million Btu/hr at a 80.1°F ambient wet bulb temperature and a cold water temperature of 90°F.
417	2.3.08.02.iii	2. The SWS provides the nonsafety-related function of transferring heat from the component cooling water system to the surrounding atmosphere to support plant shutdown and spent fuel pool cooling.	iii) Testing will be performed to confirm that the SWS cooling tower basin has adequate reserve volume.	iii) The SWS tower basin contains a usable volume of at least 230,000 gallons at the basin low level alarm setpoint.
418	2.3.08.03	3. Controls exist in the MCR to cause the components identified in Table 2.3.8-1 to perform the listed function.	Testing will be performed on the components in Table 2.3.8-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.3.8-1 to perform the listed functions.

Table 2.3.8-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
419	2.3.08.04	4. Displays of the parameters identified in Table 2.3.8-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.8-1 can be retrieved in the MCR.

Table 2.3.8-3		
Component Name	**Tag No.**	**Component Location**
Service Water Pump A	SWS-MP-01A	Turbine Building or yard
Service Water Pump B	SWS-MP-01B	Turbine Building or yard
Service Water Cooling Tower	SWS-ME-01	Yard

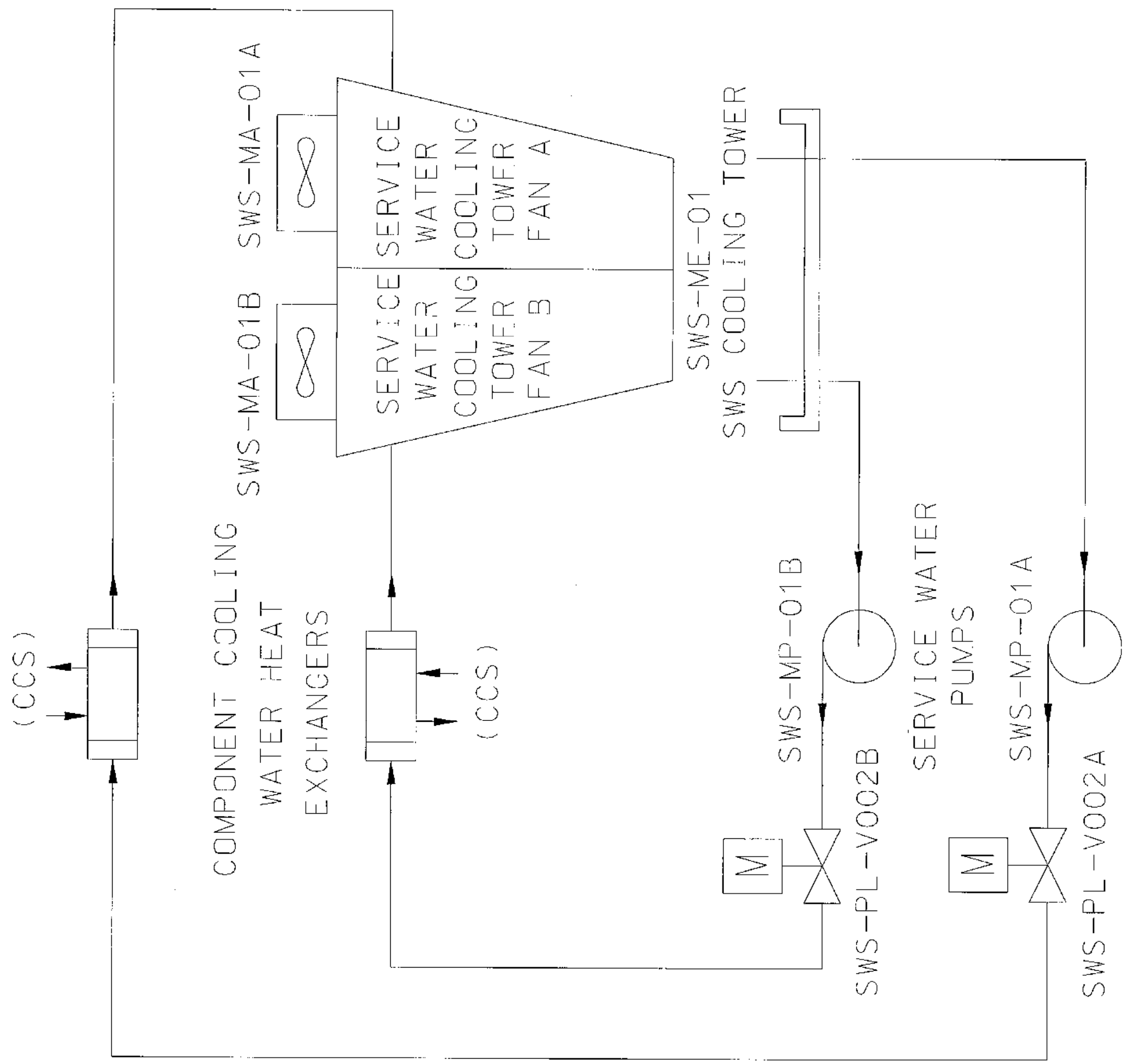


Figure 2.3.8-1
Service Water System

2.3.9 Containment Hydrogen Control System

Design Description

The containment hydrogen control system (VLS) limits hydrogen gas concentration in containment during accidents.

The VLS has catalytic hydrogen recombiners (VLS-MY-E01A and VLS-MY-E01B) that are located inside containment. The VLS has hydrogen igniters located as shown on Table 2.3.9-2.

1. The functional arrangement of the VLS is as described in the Design Description of this Section 2.3.9.
2. a) The hydrogen monitors identified in Table 2.3.9-1 are powered by the non-Class 1E dc and UPS system.
   b) The components identified in Table 2.3.9-2 are powered from their respective non-Class 1E power group.
3. The VLS provides the non-safety related function to control the containment hydrogen concentration for beyond design basis accidents.
4. a) Controls exist in the MCR to cause the components identified in Table 2.3.9-2 to perform the listed function.
   b) The components identified in Table 2.3.9-2 perform the listed function after receiving a manual signal from the diverse actuation system (DAS).
5. Displays of the parameters identified in Table 2.3.9-1 can be retrieved in the MCR.

Table 2.3.9-1

Equipment	Tag No.	Display
Containment Hydrogen Monitor	VLS-001	Yes
Containment Hydrogen Monitor	VLS-002	Yes
Containment Hydrogen Monitor	VLS-003	Yes

Table 2.3.9-2

Equipment Name	Tag Number	Function	Power Group Number	Location	Room No.
Hydrogen Igniter 01	VLS-EH-01	Energize	1	Tunnel connection loop compartments	11204
Hydrogen Igniter 02	VLS-EH-02	Energize	2	Tunnel connection loop compartments	11204
Hydrogen Igniter 03	VLS-EH-03	Energize	1	Tunnel connection loop compartments	11204
Hydrogen Igniter 04	VLS-EH-04	Energize	2	Tunnel connection loop compartments	11204

Table 2.3.9-2					
Equipment Name	**Tag Number**	**Function**	**Power Group Number**	**Location**	**Room No.**
Hydrogen Igniter 05	VLS-EH-05	Energize	1	Loop compartment 02	11402
Hydrogen Igniter 06	VLS-EH-06	Energize	2	Loop compartment 02	11502
Hydrogen Igniter 07	VLS-EH-07	Energize	2	Loop compartment 02	11402
Hydrogen Igniter 08	VLS-EH-08	Energize	1	Loop compartment 02	11502
Hydrogen Igniter 09	VLS-EH-09	Energize	1	In-containment refueling water storage tank (IRWST)	11305
Hydrogen Igniter 10	VLS-EH-10	Energize	2	IRWST	11305
Hydrogen Igniter 11	VLS-EH-11	Energize	2	Loop compartment 01	11401
Hydrogen Igniter 12	VLS-EH-12	Energize	1	Loop compartment 01	11501
Hydrogen Igniter 13	VLS-EH-13	Energize	1	Loop compartment 01	11401
Hydrogen Igniter 14	VLS-EH-14	Energize	2	Loop compartment 01	11501
Hydrogen Igniter 15	VLS-EH-15	Energize	2	IRWST	11305
Hydrogen Igniter 16	VLS-EH-16	Energize	1	IRWST	11305
Hydrogen Igniter 17	VLS-EH-17	Energize	2	Northeast valve room	11207
Hydrogen Igniter 18	VLS-EH-18	Energize	1	Northeast accumulator room	11207
Hydrogen Igniter 19	VLS-EH-19	Energize	2	East valve room	11208
Hydrogen Igniter 20	VLS-EH-20	Energize	2	Southeast accumulator room	11206
Hydrogen Igniter 21	VLS-EH-21	Energize	1	Southeast valve room	11206
Hydrogen Igniter 22	VLS-EH-22	Energize	1	Lower compartment area (core makeup tank [CMT] and valve area)	11400
Hydrogen Igniter 23	VLS-EH-23	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 24	VLS-EH-24	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 25	VLS-EH-25	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 26	VLS-EH-26	Energize	2	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 27	VLS-EH-27	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 28	VLS-EH-28	Energize	1	Lower compartment area (CMT and valve area)	11400

Table 2.3.9-2					
Equipment Name	Tag Number	Function	Power Group Number	Location	Room No.
Hydrogen Igniter 29	VLS-EH-29	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 30	VLS-EH-30	Energize	2	Loop compartment 01	11401
Hydrogen Igniter 31	VLS-EH-31	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 32	VLS-EH-32	Energize	1	Lower compartment area (CMT and valve area)	11400
Hydrogen Igniter 33	VLS-EH-33	Energize	2	North CVS equipment room	11209
Hydrogen Igniter 34	VLS-EH-34	Energize	1	North CVS equipment room	11209
Hydrogen Igniter 35	VLS-EH-35	Energize	1	IRWST	11305
Hydrogen Igniter 36	VLS-EH-36	Energize	2	IRWST	11305
Hydrogen Igniter 37	VLS-EH-37	Energize	1	IRWST	11305
Hydrogen Igniter 38	VLS-EH-38	Energize	2	IRWST	11305
Hydrogen Igniter 39	VLS-EH-39	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 40	VLS-EH-40	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 41	VLS-EH-41	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 42	VLS-EH-42	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 43	VLS-EH-43	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 44	VLS-EH-44	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 45	VLS-EH-45	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 46	VLS-EH-46	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 47	VLS-EH-47	Energize	1	Upper compartment lower region	11500
Hydrogen Igniter 48	VLS-EH-48	Energize	2	Upper compartment lower region	11500
Hydrogen Igniter 49	VLS-EH-49	Energize	1	Pressurizer compartment	11503
Hydrogen Igniter 50	VLS-EH-50	Energize	2	Pressurizer compartment	11503
Hydrogen Igniter 51	VLS-EH-51	Energize	1	Upper compartment mid-region	11500
Hydrogen Igniter 52	VLS-EH-52	Energize	2	Upper compartment mid-region	11500
Hydrogen Igniter 53	VLS-EH-53	Energize	2	Upper compartment mid-region	11500
Hydrogen Igniter 54	VLS-EH-54	Energize	1	Upper compartment mid-region	11500

Table 2.3.9-2					
Equipment Name	**Tag Number**	**Function**	**Power Group Number**	**Location**	**Room No.**
Hydrogen Igniter 55	VLS-EH-55	Energize	1	Refueling cavity	11504
Hydrogen Igniter 56	VLS-EH-56	Energize	2	Refueling cavity	11504
Hydrogen Igniter 57	VLS-EH-57	Energize	2	Refueling cavity	11504
Hydrogen Igniter 58	VLS-EH-58	Energize	1	Refueling cavity	11504
Hydrogen Igniter 59	VLS-EH-59	Energize	2	Pressurizer compartment	11503
Hydrogen Igniter 60	VLS-EH-60	Energize	1	Pressurizer compartment	11503
Hydrogen Igniter 61	VLS-EH-61	Energize	1	Upper compartment-upper region	11500
Hydrogen Igniter 62	VLS-EH-62	Energize	2	Upper compartment-upper region	11500
Hydrogen Igniter 63	VLS-EH-63	Energize	1	Upper compartment-upper region	11500
Hydrogen Igniter 64	VLS-EH-64	Energize	2	Upper compartment-upper region	11500
Hydrogen Igniter 65	VLS-EH-65	Energize	1	IRWST roof vents	11500
Hydrogen Igniter 66	VLS-EH-66	Energize	2	IRWST roof vents	11500

Table 2.3.9-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
420	2.3.09.01	1. The functional arrangement of the VLS is as described in the Design Description of this Section 2.3.9.	Inspection of the as-built system will be performed.	The as-built VLS conforms with the functional arrangement as described in the Design Description of this Section 2.3.9.
421	2.3.09.02a	2.a) The hydrogen monitors identified in Table 2.3.9-1 are powered by the non-Class 1E dc and UPS system.	Testing will be performed by providing a simulated test signal in each power group of the non-Class 1E dc and UPS system.	A simulated test signal exists at the hydrogen monitors identified in Table 2.3.9-1 when the non-Class 1E dc and UPS system is provided the test signal.
422	2.3.09.02b	2.b) The components identified in Table 2.3.9-2 are powered from their respective non-Class 1E power group.	Testing will be performed by providing a simulated test signal in each non-Class 1E power group.	A simulated test signal exists at the equipment identified in Table 2.3.9-2 when the assigned non-Class 1E power group is provided the test signal.
423	2.3.09.03.i	3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	i) Inspection for the number of igniters will be performed.	i) At least 66 hydrogen igniters are provided inside containment at the locations specified in Table 2.3.9-2.

Table 2.3.9-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
424	2.3.09.03.ii	3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	ii) Operability testing will be performed on the igniters.	ii) The surface temperature of the igniter meets or exceeds 1700°F.
425	2.3.09.03.iii	3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	iii) An inspection of the as-built containment internal structures will be performed.	iii) The minimum distance between the primary openings through the ceilings of the passive core cooling system valve/accumulator rooms (11206, 11207) and the containment shell is at least 19 feet. Primary openings are those that constitute 98% of the opening area. Other openings through the ceilings of these rooms must be at least 3 feet from the containment shell.
426	2.3.09.03.iv	3. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	iv) An inspection will be performed of the as-built IRWST vents that are located in the roof of the IRWST along the side of the IRWST next to the containment shell.	iv) The discharge from each of these IRWST vents is oriented generally away from the containment shell.
427	2.3.09.04a	4.a) Controls exist in the MCR to cause the components identified in Table 2.3.9-2 to perform the listed function.	Testing will be performed on the igniters using the controls in the MCR.	Controls in the MCR operate to energize the igniters.
428	2.3.09.04b	4.b) The components identified in Table 2.3.9-2 perform the listed function after receiving manual a signal from DAS.	Testing will be performed on the igniters using the DAS controls.	The igniters energize after receiving a signal from DAS.
429	2.3.09.05	5. Displays of the parameters identified in Table 2.3.9-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.9-1 in the MCR.	Displays identified in Table 2.3.9-1 can be retrieved in the MCR.

2.3.10 Liquid Radwaste System

Design Description

The liquid radwaste system (WLS) receives, stores, processes, samples and monitors the discharge of radioactive wastewater.

The WLS has components which receive and store radioactive or potentially radioactive liquid waste. These are the reactor coolant drain tank, the containment sump, the effluent holdup tanks and the waste holdup tanks. The WLS components store and process the waste during normal operation and during anticipated operational occurrences. Monitoring of the liquid waste is performed prior to discharge.

The WLS is as shown in Figure 2.3.10-1 and the component locations of the WLS are as shown in Table 2.3.10-5.

1. The functional arrangement of the WLS is as described in the Design Description of this Section 2.3.10.
2. a) The components identified in Table 2.3.10-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.3.10-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.3.10-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.3.10-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.3.10-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.10-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.
5. a) The seismic Category I equipment identified in Table 2.3.10-1 can withstand seismic design basis loads without loss of safety function.

   b) Each of the lines identified in Table 2.3.10-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.
6. The WLS provides the following safety-related functions:

   a) The WLS preserves containment integrity by isolation of the WLS lines penetrating the containment.

   b) Check valves in drain lines to the containment sump limit cross flooding of compartments.
7. The WLS provides the nonsafety-related functions of:

   a) Detecting leaks within containment to the containment sump.

   b) Controlling releases of radioactive materials in liquid effluents.
8. Controls exist in the main control room (MCR) to cause the remotely operated valve identified in Table 2.3.10-3 to perform its active function.

9. The check valves identified in Table 2.3.10-1 perform an active safety-related function to change position as indicated in the table.
10. Displays of the parameters identified in Table 2.3.10-3 can be retrieved in the MCR.

Table 2.3.10-1							
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Active Function**
WLS Containment Sump Level Sensor	WLS-034	No	Yes	No	No/No	No	-
WLS Containment Sump Level Sensor	WLS-035	No	Yes	No	No/No	No	-
WLS Containment Sump Level Sensor	WLS-036	No	Yes	No	No/No	No	-
WLS Drain from Passive Core Cooling System (PXS) Compartment A (Room 11206) Check Valve	WLS-PL-V071B	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment A (Room 11206) Check Valve	WLS-PL-V072B	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment B (Room 11207) Check Valve	WLS-PL-V071C	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from PXS Compartment B (Room 11207) Check Valve	WLS-PL-V072C	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from Chemical and Volume Control System (CVS) Compartment (Room 11209) Check Valve	WLS-PL-V071A	Yes	Yes	No	-/-	No	Transfer Closed
WLS Drain from CVS Compartment (Room 11209) Check Valve	WLS-PL-V072A	Yes	Yes	No	-/-	No	Transfer Closed

Note: Dash (-) indicates not applicable.

Table 2.3.10-2			
Line Name	**Line No.**	**ASME Section III**	**Functional Capability Required**
WLS Drain from PXS Compartment A	WLS-PL-L062	Yes	Yes
WLS Drain from PXS Compartment B	WLS-PL-L063	Yes	Yes
WLS Drain from CVS Compartment	WLS-PL-L061	Yes	Yes

Table 2.3.10-3			
Equipment Name	**Tag No.**	**Display**	**Active Function**
WLS Effluent Discharge Isolation Valve	WLS-PL-V223	-	Close
Reactor Coolant Drain Tank Level	WLS-JE-LT002	Yes	-
Letdown Flow from CVS to WLS	WLS-JE-FT020	Yes	-

Table 2.3.10-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
430	2.3.10.01	1. The functional arrangement of the WLS is as described in the Design Description of this Section 2.3.10.	Inspection of the as-built system will be performed.	The as-built WLS conforms with the functional arrangement as described in the Design Description of this Section 2.3.10.
431	2.3.10.02a	2.a) The components identified in Table 2.3.10-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design report exists for the as built components identified in Table 2.3.10-1 as ASME Code Section III.
432	2.3.10.02b	2.b) The piping identified in Table 2.3.10-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.3.10-2 as ASME Code Section III.
433	2.3.10.03a	3.a) Pressure boundary welds in components identified in Table 2.3.10-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.3.10-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
434	2.3.10.03b	3.b) Pressure boundary welds in piping identified in Table 2.3.10-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
435	2.3.10.04a	4.a) The components identified in Table 2.3.10-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.10-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
436	2.3.10.04b	4.b) The piping identified in Table 2.3.10-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.3.10-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
437	2.3.10.05a.i	5.a) The seismic Category I equipment identified in Table 2.3.10-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.10-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.10-1 is located on the Nuclear Island.
438	2.3.10.05a.ii	5.a) The seismic Category I equipment identified in Table 2.3.10-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
439	2.3.10.05a.iii	5.a) The seismic Category I equipment identified in Table 2.3.10-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.

Table 2.3.10-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
440	2.3.10.05b	5.b) Each of the lines identified in Table 2.3.10-2 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.3.10-2 for which functional capability is required meets the requirements for functional capability.
441	2.3.10.06a	6.a) The WLS preserves containment integrity by isolation of the WLS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
442	2.3.10.06b	6.b) Check valves in drain lines to the containment sump limit cross flooding of compartments.	Refer to item 9 in this table.	Refer to item 9 in this table.
443	2.3.10.07a.i	7.a) The WLS provides the nonsafety-related function of detecting leaks within containment to the containment sump.	i) Inspection will be performed for retrievability of the displays of containment sump level channels WLS-034, WLS-035, and WLS-036 in the MCR.	i) Nonsafety-related displays of WLS containment sump level channels WLS-034, WLS-035, and WLS-036 can be retrieved in the MCR.
444	2.3.10.07a.ii	7.a) The WLS provides the nonsafety-related function of detecting leaks within containment to the containment sump.	ii) Testing will be performed by adding water to the sump and observing display of sump level.	ii) A report exists and concludes that sump level channels WLS-034, WLS-035, and WLS-036 can detect a change of 1.75 ± 0.1 inches.
445	2.3.10.07b	7.b) The WLS provides the nonsafety-related function of controlling releases of radioactive materials in liquid effluents.	Tests will be performed to confirm that a simulated high radiation signal from the discharge radiation monitor, WLS-RE-229, causes the discharge isolation valve WLS-PL-V223 to close.	A simulated high radiation signal causes the discharge control isolation valve WLS-PL-V223 to close.
446	2.3.10.08	8. Controls exist in the MCR to cause the remotely operated valve identified in Table 2.3.10-3 to perform its active function.	Stroke testing will be performed on the remotely operated valve listed in Table 2.3.10-3 using controls in the MCR.	Controls in the MCR operate to cause the remotely operated valve to perform its active function.
447	2.3.10.09	9. The check valves identified in Table 2.3.10-1 perform an active safety-related function to change position as indicated in the table.	Exercise testing of the check valves with active safety functions identified in Table 2.3.10-1 will be performed under pre-operational test pressure, temperature and flow conditions.	Each check valve changes position as indicated on Table 2.3.10-1.

Table 2.3.10-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
448	2.3.10.10	10. Displays of the parameters identified in Table 2.3.10-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.10-3 in the MCR.	Displays identified in Table 2.3.10-3 can be retrieved in the MCR.

Table 2.3.10-5		
Component Name	**Tag No.**	**Component Location**
WLS Reactor Coolant Drain Tank	WLS-MT-01	Containment
WLS Containment Sump	WLS-MT-02	Containment
WLS Degasifier Column	WLS-MV-01	Auxiliary Building
WLS Effluent Holdup Tanks	WLS-MT-05A WLS-MT-05B	Auxiliary Building
WLS Waste Holdup Tanks	WLS-MT-06A WLS-MT-06B	Auxiliary Building
WLS Waste Pre-Filter	WLS-MV-06	Auxiliary Building
WLS Ion Exchangers	WLS-MV-03 WLS-MV-04A WLS-MV-04B WLS-MV-04C	Auxiliary Building
WLS Waste After-Filter	WLS-MV-07	Auxiliary Building
WLS Monitor Tanks	WLS-MT-07A WLS-MT-07B WLS-MT-07C	Auxiliary Building
	WLS-MT-07D WLS-MT-07E WLS-MT-07F	Radwaste Building

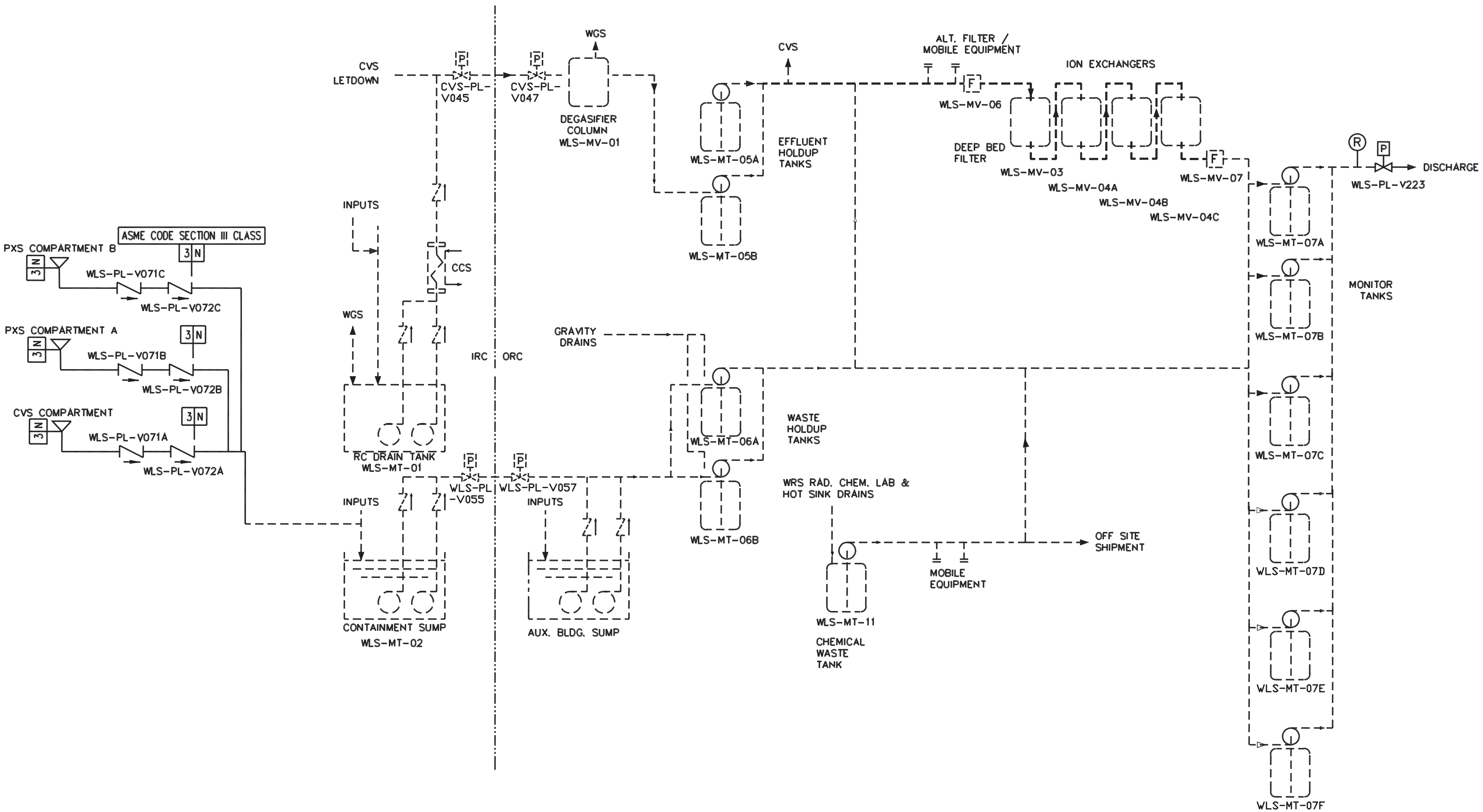


Figure 2.3.10-1
Liquid Radwaste System

2.3.11 Gaseous Radwaste System

Design Description

The gaseous radwaste system (WGS) receives, processes, and discharges the radioactive waste gases received within acceptable off-site release limits during normal modes of plant operation including power generation, shutdown and refueling.

The WGS is as shown in Figure 2.3.11-1 and the component locations of the WGS are as shown in Table 2.3.11-3.

1. The functional arrangement of the WGS is as described in the Design Description of this Section 2.3.11.
2. The equipment identified in Table 2.3.11-1 can withstand the appropriate seismic design basis loads without loss of its structural integrity function.
3. The WGS provides the nonsafety-related functions of:
   a. Processing radioactive gases prior to discharge.
   b. Controlling the releases of radioactive materials in gaseous effluents.
   c. The WGS is purged with nitrogen on indication of high oxygen levels in the system.

Table 2.3.11-1

Equipment Name	Tag No.	Seismic Category I
WGS Activated Carbon Delay Bed A	WGS-MV-02A	No(1)
WGS Activated Carbon Delay Bed B	WGS-MV-02B	No(1)
WGS Discharge Isolation Valve	WGS-PL-V051	No

Note:

1. The WGS activated carbon delay beds (WGS-MV-02A and B) are designed to one-half SSE.

Table 2.3.11-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
449	2.3.11.01	1. The functional arrangement of the WGS is as described in the Design Description of this Section 2.3.11.	Inspection of the as-built system will be performed.	The as-built WGS conforms with the functional arrangement as described in the Design Description of this Section 2.3.11.

Table 2.3.11-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
450	2.3.11.02.i	2. The equipment identified as having seismic design requirements in Table 2.3.11-1 can withstand seismic design basis loads without loss of its structural integrity function.	i) Inspection will be performed to verify that the equipment identified as having seismic design requirements in Table 2.3.11-1 is located on the Nuclear Island.	i) The equipment identified as having seismic design requirements in Table 2.3.11-1 is located on the Nuclear Island.
451	2.3.11.02.ii	2. The equipment identified as having seismic design requirements in Table 2.3.11-1 can withstand seismic design basis loads without loss of its structural integrity function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismically designed equipment will be performed.	ii) A report exists and concludes that the seismically designed equipment can withstand appropriate seismic design basis loads without loss of its structural integrity function.
452	2.3.11.02.iii	2. The equipment identified as having seismic design requirements in Table 2.3.11-1 can withstand seismic design basis loads without loss of its structural integrity function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
453	2.3.11.03a	3.a) The WGS provides the nonsafety-related function of processing radioactive gases prior to discharge.	Inspection will be performed to verify the contained volume of each of the activated carbon delay beds, WGS-MV02A and WGS-MV02B.	A report exists and concludes that the contained volume in each of the activated carbon delay beds, WGS-MV02A and WGS-MV02B, is at least 80 ft^3.
454	2.3.11.03b	3.b) The WGS provides the nonsafety-related function of controlling the releases of radioactive materials in gaseous effluents.	Tests will be performed to confirm that the presence of a simulated high radiation signal from the discharge radiation monitor, WGS-017, causes the discharge control isolation valve WGS-PL-V051 to close.	A simulated high radiation signal causes the discharge control isolation valve WGS-PL-V051 to close.
455	2.3.11.03c	3.c) The WGS is purged with nitrogen on indication of high oxygen levels in the system.	Tests will be performed to confirm that the presence of a simulated high oxygen level signal from the oxygen monitors (WGS-025A, -025B) causes the nitrogen purge valve (WGS-PL-V002) to open and the WLS degasifier vacuum pumps (WLS-MP-03A, -03B) to stop.	A simulated high oxygen level signal causes the nitrogen purge valve (WGS-PL-V002) to open and the WLS degasifier vacuum pumps (WLS-MP-03A, -03B) to stop.

Table 2.3.11-3		
Equipment Name	**Tag No.**	**Component Location**
WGS Gas Cooler	WGS-ME-01	Auxiliary Building
WGS Moisture Separator	WGS-MV-03	Auxiliary Building
WGS Activated Carbon Delay Bed A	WGS-MV-02A	Auxiliary Building
WGS Activated Carbon Delay Bed B	WGS-MV-02B	Auxiliary Building

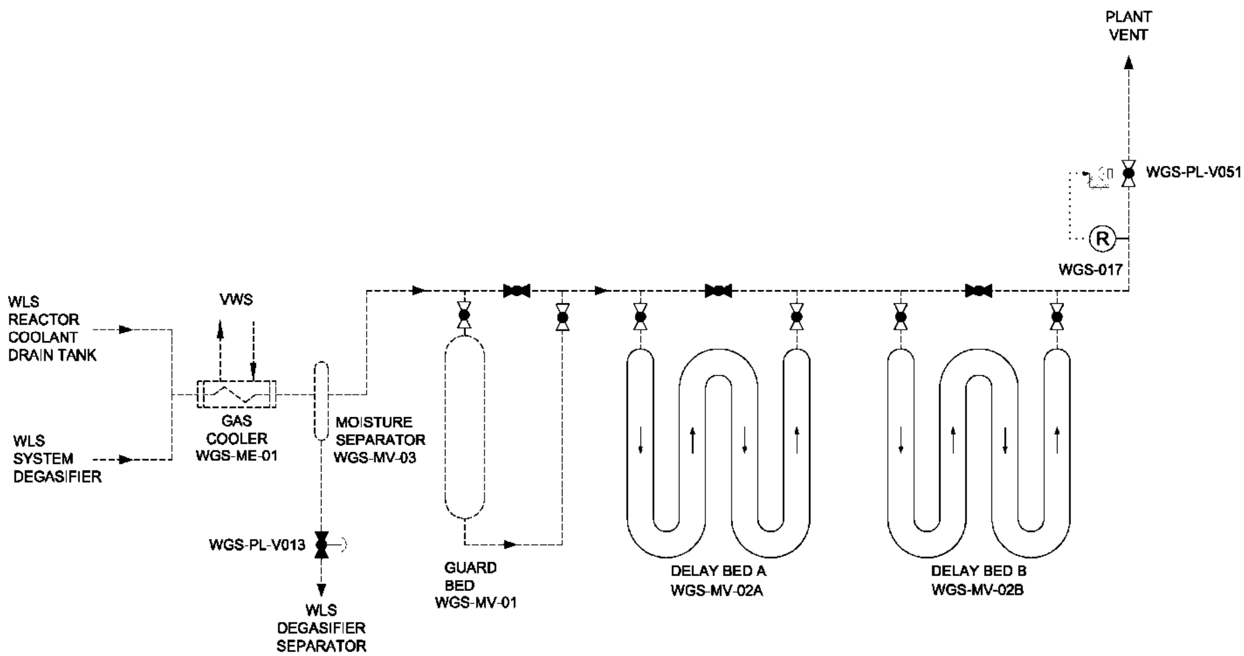


Figure 2.3.11-1
Gaseous Radwaste System

2.3.12 Solid Radwaste System

Design Description

The solid radwaste system (WSS) receives, collects, and stores the solid radioactive wastes received prior to their processing and packaging by mobile equipment for shipment off-site.

The component locations of the WSS are as shown in Table 2.3.12-2.

1. The functional arrangement of the WSS is as described in the Design Description of this Section 2.3.12.
2. The WSS provides the nonsafety-related function of storing radioactive spent resins prior to processing or shipment.

Table 2.3.12-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
456	2.3.12.01	1. The functional arrangement of the WSS is as described in the Design Description of this Section 2.3.12.	Inspection of the as-built system will be performed.	The as-built WSS conforms with the functional arrangement as described in the Design Description of this Section 2.3.12.
457	2.3.12.02	2. The WSS provides the nonsafety-related function of storing radioactive solids prior to processing or shipment.	Inspection will be performed to verify that the volume of each of the spent resin tanks, WSS-MV01A and WSS-MV01B, is at least 250 ft^3.	A report exists and concludes that the volume of each of the spent resin tanks, WSS-MV01A and WSS-MV01B, is at least 250 ft^3.

Table 2.3.12-2

Component Name	Tag No.	Component Location
WSS Spent Resin Tank A	WSS-MV-01A	Auxiliary Building
WSS Spent Resin Tank B	WSS-MV-01B	Auxiliary Building

2.3.13 Primary Sampling System

Design Description

The primary sampling system collects samples of fluids in the reactor coolant system (RCS) and the containment atmosphere during normal operations.

The PSS is as shown in Figure 2.3.13-1. The PSS Grab Sampling Unit (PSS-MS-01) is located in the Auxiliary Building.

1. The functional arrangement of the PSS is as described in the Design Description of this Section 2.3.13.
2. The components identified in Table 2.3.13-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
3. Pressure boundary welds in components identified in Table 2.3.13-1 as ASME Code Section III meet ASME Code Section III requirements.
4. The components identified in Table 2.3.13-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
5. The seismic Category I equipment identified in Table 2.3.13-1 can withstand seismic design basis loads without loss of safety function.
6. a) The Class 1E equipment identified in Table 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of their safety function, for the time required to perform the safety function.
   b) The Class 1E components identified in Table 2.3.13-1 are powered from their respective Class 1E division.
   c) Separation is provided between PSS Class 1E divisions, and between Class 1E divisions and non-Class 1E divisions.
7. The PSS provides the safety-related function of preserving containment integrity by isolation of the PSS lines penetrating the containment.
8. The PSS provides the nonsafety-related function of providing the capability of obtaining reactor coolant and containment atmosphere samples.
9. Safety-related displays identified in Table 2.3.13-1 can be retrieved in the MCR.
10. a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.13-1 to perform active functions.
    b) The valves identified in Table 2.3.13-1 as having protection and safety monitoring system (PMS) control perform an active function after receiving a signal from the PMS.
11. a) Deleted
    b) After loss of motive power, the remotely operated valves identified in Table 2.3.13-1 assume the indicated loss of motive power position.
12. Controls exist in the MCR to cause the valves identified in Table 2.3.13-2 to perform the listed function.

Table 2.3.13-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS**	**Active Function**	**Loss of Motive Power Position**
Containment Air Sample Containment Isolation Valve Inside Reactor Containment (IRC)	PSS-PL-V008	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve IRC	PSS-PL-V010A	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve IRC	PSS-PL-V010B	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve Outside Reactor Containment (ORC)	PSS-PL-V011A	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Liquid Sample Line Containment Isolation Valve ORC	PSS-PL-V011B	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Sample Return Line Containment Isolation Valve ORC	PSS-PL-V023	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Sample Return Containment Isolation Valve IRC	PSS-PL-V024	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes/No	Transfer Closed	Closed
Air Sample Line Containment Isolation Valve ORC	PSS-PL-V046	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	Closed

Note: A dash (-) indicates not applicable.

Table 2.3.13-2		
Equipment Name	**Tag No.**	**Control Function**
Hot Leg 1 Sample Isolation Valve	PSS-PL-V001A	Transfer Open/Transfer Closed
Hot Leg 2 Sample Isolation Valve	PSS-PL-V001B	Transfer Open/Transfer Closed

Table 2.3.13-3
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
458	2.3.13.01	1. The functional arrangement of the PSS is as described in the Design Description of this Section 2.3.13.	Inspection of the as-built system will be performed.	The as-built PSS conforms with the functional arrangement as described in the Design Description of this Section 2.3.13.
459	2.3.13.02	2. The components identified in Table 2.3.13-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.3.13-1 as ASME Code Section III.
460	2.3.13.03	3. Pressure boundary welds in components identified in Table 2.3.13-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
461	2.3.13.04	4. The components identified in Table 2.3.13-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.13-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
462	2.3.13.05.i	5. The seismic Category I equipment identified in Table 2.3.13-1 can withstand seismic design basis loads without loss of its safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.3.13-1 are located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.13-1 is located on the Nuclear Island.
463	2.3.13.05.ii	5. The seismic Category I equipment identified in Table 2.3.13-1 can withstand seismic design basis loads without loss of its safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.

Table 2.3.13-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
464	2.3.13.05.iii	5. The seismic Category I equipment identified in Table 2.3.13-1 can withstand seismic design basis loads without loss of its safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
465	2.3.13.06a.i	6.a) The Class 1E equipment identified in Tables 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of their safety function, for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of its safety function for the time required to perform the safety function.
466	2.3.13.06a.ii	6.a) The Class 1E equipment identified in Tables 2.3.13-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of their safety function, for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.3.13-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
467	2.3.13.06b	6.b) The Class 1E components identified in Table 2.3.13-1 are powered from their respective Class 1E division.	Testing will be performed on the PSS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.13-1 when the assigned Class 1E division is provided the test signal.
468	2.3.13.06c	6.c) Separation is provided between PSS Class 1E divisions, and between Class 1E divisions and non-Class 1E divisions.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
469	2.3.13.07	7. The PSS provides the safety- related function of preserving containment integrity by isolation of the PSS lines penetrating the containment.	See ITAAC Table 2.2.1-3, item 7.	See ITAAC Table 2.2.1-3, item 7.

Table 2.3.13-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
470	2.3.13.08	8. The PSS provides the nonsafety-related function of providing the capability of obtaining reactor coolant and containment atmosphere samples.	Testing will be performed to obtain samples of the reactor coolant and containment atmosphere.	A sample is drawn from the reactor coolant and the containment atmosphere.
471	2.3.13.09	9. Safety-related displays identified in Table 2.3.13-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	The safety-related displays identified in Table 2.3.13-1 can be retrieved in the MCR.
472	2.3.13.10a	10.a) Controls exist in the MCR to cause those remotely operated valves identified in Table 2.3.13-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.13-1 using the controls in the MCR.	Controls in the MCR operate to cause those remotely operated valves identified in Table 2.3.13-1 to perform active functions.
473	2.3.13.10b	10.b) The valves identified in Table 2.3.13-1 as having PMS control perform an active function after receiving a signal from the PMS.	Testing will be performed on remotely operated valves listed in Table 2.3.13-1 using real or simulated signals into the PMS.	The remotely operated valves identified in Table 2.3.13-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
474	2.3.13.11a	11.a) Deleted		
475	2.3.13.11b	11.b) After loss of motive power, the remotely operated valves identified in Table 2.3.13-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	After loss of motive power, each remotely operated valve identified in Table 2.3.13-1 assumes the indicated loss of motive power position.
476	2.3.13.12	12. Controls exist in the MCR to cause the valves identified in Table 2.3.13-2 to perform the listed function.	Testing will be performed on the components in Table 2.3.13-2 using controls in the MCR.	Controls in the MCR cause valves identified in Table 2.3.13-2 to perform the listed functions.

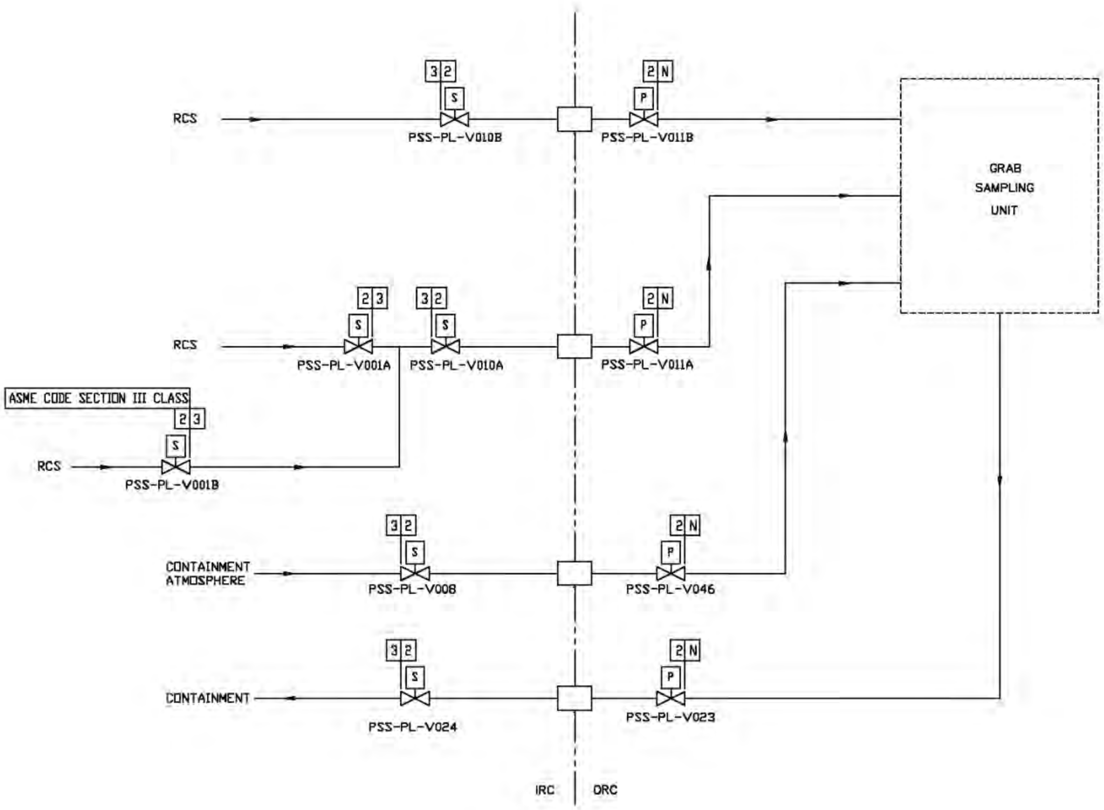


Figure 2.3.13-1
Primary Sampling System

2.3.14 Demineralized Water Transfer and Storage System

Design Description

The demineralized water transfer and storage system (DWS) receives water from the demineralized water treatment system (DTS), and provides a reservoir of demineralized water to supply the condensate storage tank and for distribution throughout the plant. Demineralized water is processed in the DWS to remove dissolved oxygen. In addition to supplying water for makeup of systems which require pure water, the demineralized water is used to sluice spent radioactive resins from the ion exchange vessels in the chemical and volume control system (CVS), the spent fuel pool cooling system (SFS), and the liquid radwaste system (WLS) to the solid radwaste system (WSS).

The component locations of the DWS are as shown in Table 2.3.14-3.

1. The functional arrangement of the DWS is as described in the Design Description of this Section 2.3.14.
2. The DWS provides the safety-related function of preserving containment integrity by isolation of the DWS lines penetrating the containment.
3. The DWS condensate storage tank (CST) provides the nonsafety-related function of water supply to the FWS startup feedwater pumps.
4. Displays of the parameters identified in Table 2.3.14-1 can be retrieved in the main control room (MCR).

Table 2.3.14-1

Equipment Name	Tag No.	Display	Control Function
Condensate Storage Tank Water Level	DWS-006	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.14-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
477	2.3.14.01	1. The functional arrangement of the DWS is as described in the Design Description of this Section 2.3.14.	Inspection of the as-built system will be performed.	The as-built DWS conforms with the functional arrangement as described in the Design Description of this Section 2.3.14.
478	2.3.14.02	2. The DWS provides the safety-related function of preserving containment integrity by isolation of the DWS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.

Table 2.3.14-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
479	2.3.14.03	3. The DWS CST provides the nonsafety-related function of water supply to the FWS startup feedwater pumps.	Inspection of the DWS CST will be performed.	The volume of the CST between the tank overflow and the startup feedwater pumps supply connection is greater than or equal to 325,000 gallons.
480	2.3.14.04	4. Displays of the parameters identified in Table 2.3.14-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.14-1 can be retrieved in the MCR.

Table 2.3.14-3		
Component Name	**Tag No.**	**Component Location**
Demineralizer Water Storage Tank Degasification System Package	DWS-MS-01	Annex Building
Condensate Storage Tank Degasification System Package	DWS-MS-02	Turbine Building
Demineralized Water Storage Tank	DWS-MT-01	Yard
Condensate Storage Tank	DWS-MT-02	Yard

2.3.15 Compressed and Instrument Air System

Design Description

The compressed and instrument air system (CAS) consists of three subsystems: instrument air, service air, and high-pressure air. The instrument air subsystem supplies compressed air for air-operated valves and dampers. The service air subsystem supplies compressed air at outlets throughout the plant to power air-operated tools and is used as a motive force for air-powered pumps. The service air subsystem is also utilized as a supply source for breathing air. The high-pressure air subsystem supplies air to the main control room emergency habitability system (VES) and fire fighting apparatus recharge station.

The CAS is required for normal operation and startup of the plant.

The component locations of the CAS are as shown in Table 2.3.15-3.

1. The functional arrangement of the CAS is as described in the Design Description of this Section 2.3.15.
2. The CAS provides the safety-related function of preserving containment integrity by isolation of the CAS lines penetrating the containment.

3. Displays of the parameters identified in Table 2.3.15-1 can be retrieved in the main control room (MCR).

Table 2.3.15-1			
Equipment Name	**Tag No.**	**Display**	**Control Function**
Instrument Air Pressure	CAS-011	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.3.15-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
481	2.3.15.01	1. The functional arrangement of the CAS is as described in the Design Description of this Section 2.3.15.	Inspection of the as-built system will be performed.	The as-built CAS conforms with the functional arrangement as described in the Design Description of this Section 2.3.15.
482	2.3.15.02	2. The CAS provides the safety-related function of preserving containment integrity by isolation of the CAS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
483	2.3.15.03	3. Displays of the parameters identified in Table 2.3.15-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.3.15-1 can be retrieved in the MCR.

Table 2.3.15-3		
Component Name	**Tag No.**	**Component Location**
Instrument Air Compressor Package A	CAS-MS-01A	Turbine Building
Instrument Air Compressor Package B	CAS-MS-01B	Turbine Building
Instrument Air Dryer Package A	CAS-MS-02A	Turbine Building
Instrument Air Dryer Package B	CAS-MS-02B	Turbine Building
Service Air Compressor Package A	CAS-MS-03A	Turbine Building
Service Air Compressor Package B	CAS-MS-03B	Turbine Building
Service Air Dryer Package A	CAS-MS-04A	Turbine Building
Service Air Dryer Package B	CAS-MS-04B	Turbine Building
High Pressure Air Compressor and Filter Package	CAS-MS-05	Turbine Building

Table 2.3.15-3		
Component Name	**Tag No.**	**Component Location**
Instrument Air Receiver A	CAS-MT-01A	Turbine Building
Instrument Air Receiver B	CAS-MT-01B	Turbine Building
Service Air Receiver	CAS-MT-02	Turbine Building

2.3.16 Potable Water System

No entry for this system.

2.3.17 Waste Water System

No entry for this system.

2.3.18 Plant Gas System

No entry. Covered in Section 3.3, Buildings.

2.3.19 Communication System

Design Description

The communication system (EFS) provides intraplant communications during normal, maintenance, transient, fire, and accident conditions, including loss of offsite power.

1. a) The EFS has handsets, amplifiers, loudspeakers, and siren tone generators connected as a telephone/page system.

   b) The EFS has sound-powered equipment connected as a system.

2. The EFS provides the following nonsafety-related functions:

   a) The EFS telephone/page system provides intraplant, station-to-station communications and area broadcasting between the main control room (MCR) and the locations listed in Table 2.3.19-1.

   b) The EFS provides sound-powered communications between the MCR, the remote shutdown workstation (RSW), the Division A, B, C, D dc equipment rooms (Rooms 12201/12203/12205/12207), the Division A, B, C, D I&C rooms (Rooms 12301/12302/12304/12305), and the diesel generator building (Rooms 60310/60320) without external power.

Table 2.3.19-1	
Telephone/Page System Equipment	**Location**
Fuel Handling Area	12562
Division A, B, C, D dc Equipment Rooms	12201/12203/12205/12207
Division A, B, C, D I&C Rooms	12301/12302/12304/12305
Maintenance Floor Staging Area	12351
Containment Maintenance Floor	11300
Containment Operating Deck	11500

Table 2.3.19-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
484	2.3.19.01a	1.a) The EFS has handsets, amplifiers, loudspeakers, and siren tone generators connected as a telephone/page system.	Inspection of the as-built system will be performed.	The as-built EFS has handsets, amplifiers, loudspeakers, and siren tone generators connected as a telephone/page system.
485	2.3.19.01b	1.b) The EFS has sound-powered equipment connected as a system.	Inspection of the as-built system will be performed.	The as-built EFS has sound-powered equipment connected as a system.
486	2.3.19.02a	2.a) The EFS telephone/page system provides intraplant, station-to-station communications and area broadcasting between the MCR and the locations listed in Table 2.3.19-1.	An inspection and test will be performed on the telephone/page communication equipment.	Telephone/page equipment is installed and voice transmission and reception from the MCR are accomplished.
487	2.3.19.02b	2.b) EFS provides sound-powered communications between the MCR, the RSW, the Division A, B, C, D dc equipment rooms (Rooms 12201/12203/12205/ 12207), the Division A, B, C, D I&C rooms (Rooms 12301/12302/ 12304/12305), and the diesel generator building (Rooms 60310/60320) without external power.	An inspection and test will be performed of the sound-powered communication equipment.	Sound-powered equipment is installed and voice transmission and reception are accomplished.

2.3.20 Turbine Building Closed Cooling Water System

No entry for this system.

2.3.21 Secondary Sampling System

No entry for this system.

2.3.22 Containment Leak Rate Test System

No entry. Covered in Section 2.2.1, Containment System.

2.3.23 This section intentionally blank

2.3.24 Demineralized Water Treatment System

No entry for this system.

2.3.25 Gravity and Roof Drain Collection System

No entry for this system.

2.3.26 This section intentionally blank

2.3.27 Sanitary Drainage System

No entry for this system.

2.3.28 Turbine Island Vents, Drains, and Relief System

No entry for this system.

2.3.29 Radioactive Waste Drain System

Design Description

The radioactive waste drain system (WRS) collects radioactive and potentially radioactive liquid wastes from equipment and floor drains during normal operation, startup, shutdown, and refueling. The liquid wastes are then transferred to appropriate processing and disposal systems.

Nonradioactive wastes are collected by the waste water system (WWS). The WRS is as shown in Figure 2.3.29-1.

1. The functional arrangement of the WRS is as described in the Design Description of this Section 2.3.29.
2. The WRS collects liquid wastes from the equipment and floor drainage of the radioactive portions of the auxiliary building, annex building, and radwaste building and directs these wastes to a WRS sump or WLS waste holdup tanks located in the auxiliary building.
3. The WRS collects chemical wastes from the auxiliary building chemical laboratory drains and the decontamination solution drains in the annex building and directs these wastes to the chemical waste tank of the liquid radwaste system.
4. The WWS stops the discharge from the turbine building sump upon detection of high radiation in the discharge stream to the oil separator.

Table 2.3.29-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
488	2.3.29.01	1. The functional arrangement of the WRS is as described in the Design Description of this Section 2.3.29.	Inspection of the as-built system will be performed.	The as-built WRS conforms with the functional arrangement as described in the Design Description of this Section 2.3.29.
489	2.3.29.02	2. The WRS collects liquid wastes from the equipment and floor drainage of the radioactive portions of the auxiliary building, annex building, and radwaste building and directs these wastes to a WRS sump or WLS waste holdup tanks located in the auxiliary building.	A test is performed by pouring water into the equipment and floor drains in the radioactive portions of the auxiliary building, annex building, and radwaste building.	The water poured into these drains is collected either in the auxiliary building radioactive drains sump or the WLS waste holdup tanks.
490	2.3.29.03	3. The WRS collects chemical wastes from the auxiliary building chemical laboratory drains and the decontamination solution drains in the annex building and directs these wastes to the chemical waste tank of the liquid radwaste system.	A test is performed by pouring water into the auxiliary building chemical laboratory and the decontamination solution drains in the annex building.	The water poured into these drains is collected in the chemical waste tank of the liquid radwaste system.
491	2.3.29.04	4. The WWS stops the discharge from the turbine building sump upon detection of high radiation in the discharge stream to the oil separator.	Tests will be performed to confirm that a simulated high radiation signal from the turbine building sump discharge radiation monitor, WWS-021 causes the sump pumps (WWS-MP-01A and B) to stop operating, stopping the spread of radiation outside of the turbine building.	A simulated high radiation signal causes the turbine building sump pumps (WWS-MP-01A and B) to stop operating, stopping the spread of radiation outside of the turbine building.

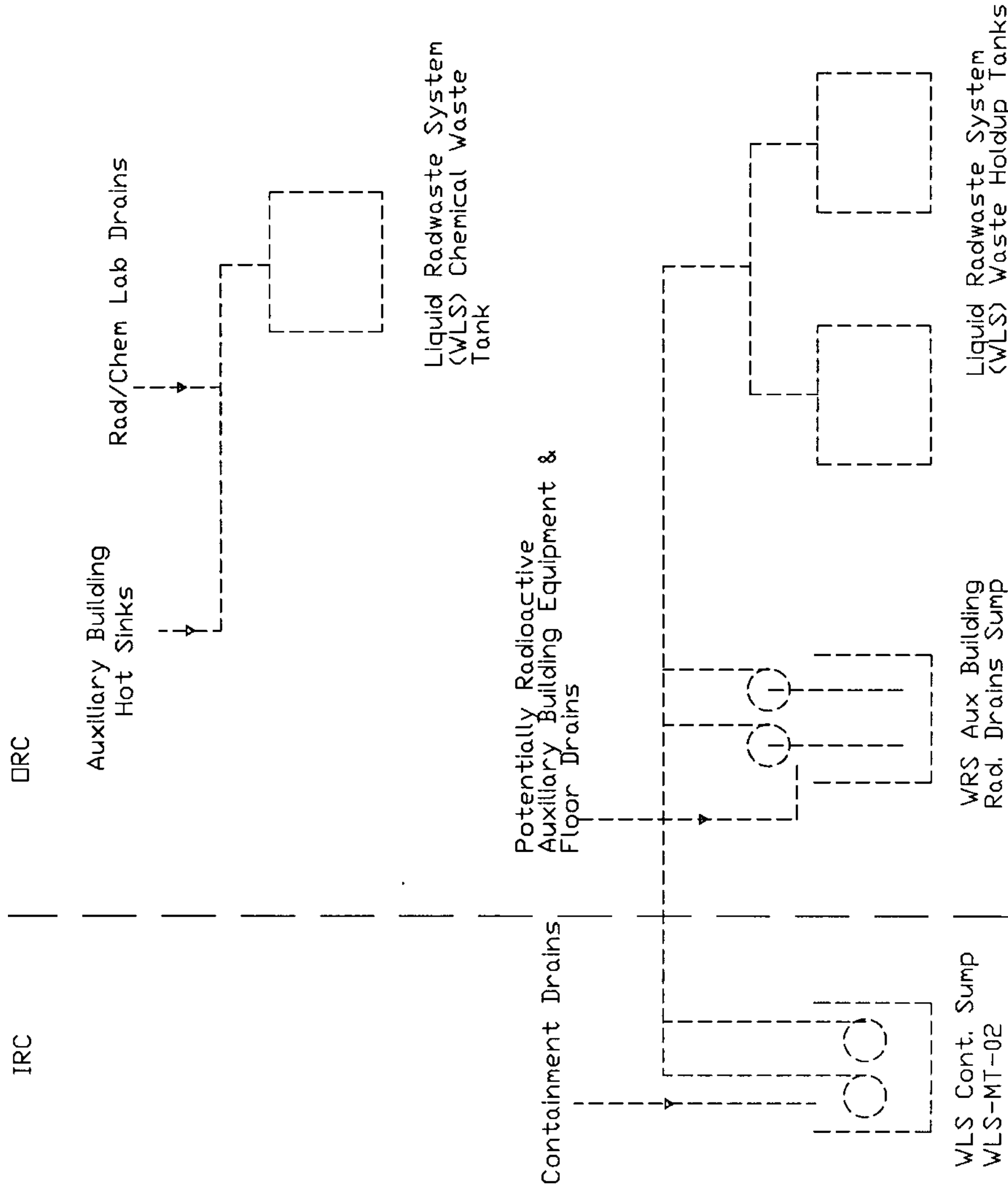


Figure 2.3.29-1
Radioactive Waste Drain System

C.2.3.30 Storm Drain System

No entry for this system.

C.2.3.31 Raw Water System

No entry for this system.

C.2.3.32 Yard Fire Water System

No entry for this system.

2.4 Steam and Power Conversion Systems

2.4.1 Main and Startup Feedwater System

See Section 2.2.4 for information on the main feedwater system.

Design Description

The startup feedwater system supplies feedwater to the steam generators during plant startup, hot standby and shutdown conditions, and during transients in the event of main feedwater system unavailability.

1. The functional arrangement of the startup feedwater system is as described in the Design Description of this Section 2.4.1.
2. The FWS provides the following nonsafety-related functions:

   The FWS provides startup feedwater flow from the condensate storage tank (CST) to the steam generator system (SGS) for heat removal from the RCS.
3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.4.1-1 to perform the listed function.
4. Displays of the parameters identified in Table 2.4.1-1 can be retrieved in the MCR.

Table 2.4.1-1

Equipment Name	Tag No.	Display	Control Function
Startup Feedwater Pump A (Motor)	FWS-MP-03A	Yes (Run Status)	Start
Startup Feedwater Pump B (Motor)	FWS-MP-03B	Yes (Run Status)	Start
Startup Feedwater Pump A Isolation Valve	FWS-PL-V013A	Yes (Valve Position)	Open
Startup Feedwater Pump B Isolation Valve	FWS-PL-V013B	Yes (Valve Position)	Open

Table 2.4.1-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
492	2.4.01.01	1. The functional arrangement of the startup feedwater system is as described in the Design Description of this Section 2.4.1.	Inspection of the as-built system will be performed.	The as-built startup feedwater system conforms with the functional arrangement as described in the Design Description of this Section 2.4.1.
493	2.4.01.02	2. The FWS provides startup feedwater flow from the CST to the SGS for heat removal from the RCS.	Testing will be performed to confirm that each of the startup feedwater pumps can provide water from the CST to both steam generators.	Each FWS startup feedwater pump provides a flow rate greater than or equal to 260 gpm to each steam generator system at a steam generator secondary side pressure of at least 1106 psia.
494	2.4.01.03	3. Controls exist in the MCR to cause the components identified in Table 2.4.1-1 to perform the listed function.	Testing will be performed on the components in Table 2.4.1-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.4.1-1 to perform the listed functions.
495	2.4.01.04	4. Displays of the parameters identified in Table 2.4.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.4.1-1 can be retrieved in the MCR.

Table 2.4.1-3		
Component Name	**Tag No.**	**Component Location**
Startup Feedwater Pump A	FWS-MP-03A	Turbine Building
Startup Feedwater Pump B	FWS-MP-03B	Turbine Building

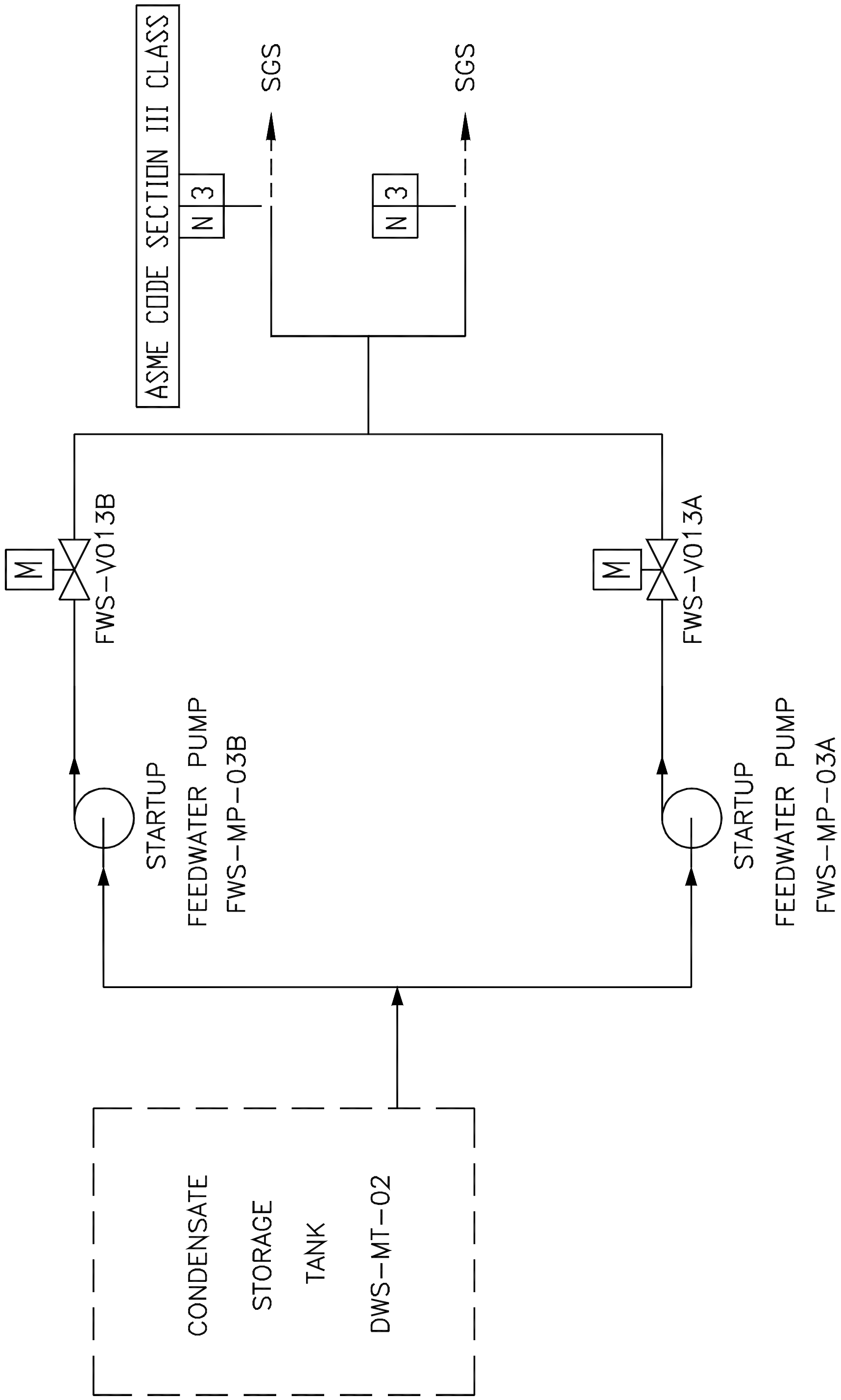


Figure 2.4.1-1
Main and Startup Feedwater System

2.4.2 Main Turbine System

Design Description

The main turbine system (MTS) is designed for electric power production consistent with the capability of the reactor and the reactor coolant system.

The component locations of the MTS are as shown in Table 2.4.2-2.

1. The functional arrangement of the MTS is as described in the Design Description of this Section 2.4.2.
2. a) Controls exist in the MCR to trip the main turbine-generator.
   b) The main turbine-generator trips after receiving a signal from the PMS.
   c) The main turbine-generator trips after receiving a signal from the DAS.
3. The overspeed trips for the AP1000 turbine are set for 110% and 111% (±1% each). Each trip is initiated electrically in separate systems. The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.

Table 2.4.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
496	2.4.02.01	1. The functional arrangement of the MTS is as described in the Design Description of this Section 2.4.2.	Inspection of the as-built system will be performed.	The as-built MTS conforms with the functional arrangement as described in the Design Description of this Section 2.4.2.
497	2.4.02.02a	2.a) Controls exist in the MCR to trip the main turbine-generator.	Testing will be performed on the main turbine-generator using controls in the MCR.	Controls in the MCR operate to trip the main turbine-generator.
498	2.4.02.02b	2.b) The main turbine-generator trips after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The main turbine-generator trips after receiving a signal from the PMS.
499	2.4.02.02c	2.c) The main turbine-generator trips after receiving a signal from the DAS.	Testing will be performed using real or simulated signals into the DAS.	The main turbine-generator trips after receiving a signal from the DAS.
500	2.4.02.03.i	3) The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.	i) The system design will be reviewed.	i) The system design review shows that the trip signals of the two electrical overspeed protection trip systems are isolated from, and independent of, each other.

Table 2.4.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
501	2.4.02.03.ii	3) The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.	ii) Testing of the as-built system will be performed using simulated signals from the turbine speed sensors.	ii) The main turbine-generator trips after overspeed signals are received from the speed sensors of the 110% emergency electrical overspeed trip system, and the main turbine-generator trips after overspeed signals are received from the speed sensors of the 111% backup electrical overspeed trip system.
502	2.4.02.03.iii	3) The trip signals from the two turbine electrical overspeed protection trip systems are isolated from, and independent of, each other.	iii) Inspection will be performed for the existence of a report verifying that the two turbine electrical overspeed protection systems have diverse hardware and software/firmware.	iii) A report exists and concludes that the two electrical overspeed protection systems have diverse hardware and software/firmware.

Table 2.4.2-2

Component Name	Tag No.	Component Location
HP Turbine	MTS-MG-01	Turbine Building
LP Turbine A	MTS-MG-02A	Turbine Building
LP Turbine B	MTS-MG-02B	Turbine Building
LP Turbine C	MTS-MG-02C	Turbine Building
Gland Steam Condenser	GSS-ME-01	Turbine Building
Gland Condenser Vapor Exhauster 1A	GSS-MA-01A	Turbine Building
Gland Condenser Vapor Exhauster 1B	GSS-MA-01B	Turbine Building
Electrical Overspeed Trip Device	--	Turbine Building
Emergency Electrical Overspeed Trip Device	--	Turbine Building

2.4.3 Main Steam System

No entry. Covered in Section 2.2.4, Steam Generator System.

2.4.4 Steam Generator Blowdown System

No entry. Containment isolation function covered in Section 2.2.1, Containment System and 2.2.4, Steam Generator System.

No entry. Steam generator isolation function covered in Section 2.2.4, Steam Generator System.

2.4.5 Condenser Air Removal System

No entry. Covered in Section 3.5, Radiation Monitoring.
(Note: Monitor is TDS-RE001.)

2.4.6 Condensate System

Design Description

The condensate system (CDS) provides feedwater at the required temperature, pressure, and flow rate to the deaerator. Condensate is pumped from the main condenser hotwell by the condensate pumps and passes through the low-pressure feedwater heaters to the deaerator. The circulating water system (CWS) removes heat from the condenser and is site specific starting from the interface at the locations where the CWS piping enters and exits the turbine building.

The CDS operates during plant startup and power operations (full and part loads).

The component locations of the CDS are as shown in Table 2.4.6-3.

1. The functional arrangement of the CDS is as described in the Design Description of this Section 2.4.6.
2. Displays of the parameters identified in Table 2.4.6-1 can be retrieved in the main control room (MCR).

Table 2.4.6-1

Equipment Name	Tag No.	Display
Condenser Backpressure	CDS-056A	Yes
Condenser Backpressure	CDS-056B	Yes
Condenser Backpressure	CDS-056C	Yes

Table 2.4.6-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
503	2.4.06.01	1. The functional arrangement of the CDS is as described in the Design Description of this Section 2.4.6.	Inspection of the as-built system will be performed.	The as-built CDS conforms with the functional arrangement as described in the Design Description of Section 2.4.6.
504	2.4.06.02	2. Displays of the parameters identified in Table 2.4.6-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.4.6-1 can be retrieved in the MCR.

Table 2.4.6-3	
Component Name	**Component Location**
Low Pressure Feedwater Heaters	Turbine Building
Deaerator Feedwater Heater and Storage Tank	Turbine Building
Main Condenser Shell A	Turbine Building
Main Condenser Shell B	Turbine Building
Main Condenser Shell C	Turbine Building
Condensate Pump A	Turbine Building
Condensate Pump B	Turbine Building
Condensate Pump C	Turbine Building

2.4.7 Circulating Water System

No entry for this system.

2.4.8 Auxiliary Steam Supply System

No entry for this system.

2.4.9 Condenser Tube Cleaning System

No entry for this system.

2.4.10 Turbine Island Chemical Feed System

No entry for this system.

2.4.11 Condensate Polishing System

No entry for this system.

2.4.12 Gland Seal System

No entry. Covered in Section 2.4.2, Main Turbine System.

2.4.13 Generator Hydrogen and CO2 System

No entry for this system.

2.4.14 Heater Drain System

No entry for this system.

2.4.15 Hydrogen Seal Oil System

No entry for this system.

2.4.16 Main Turbine and Generator Lube Oil System

No entry for this system.

2.5 Instrumentation and Control Systems

2.5.1 Diverse Actuation System

Design Description

The diverse actuation system (DAS) initiates reactor trip, actuates selected functions, and provides plant information to the operator.

The component locations of the DAS are as shown in Table 2.5.1-5.

1. The functional arrangement of the DAS is as described in the Design Description of this Section 2.5.1.
2. The DAS provides the following nonsafety-related functions:
   a) The DAS provides an automatic reactor trip on low wide-range steam generator water level, or on low pressurizer water level, or on high hot leg temperature, separate from the PMS.
   b) The DAS provides automatic actuation of selected functions, as identified in Table 2.5.1-1, separate from the PMS.
   c) The DAS provides manual initiation of reactor trip and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypasses the control room multiplexers, if any; the PMS cabinets; and the signal processing equipment of the DAS.
   d) The DAS provides main control room (MCR) displays of selected plant parameters, as identified in Table 2.5.1-3, separate from the PMS.

3. The DAS has the following features:

   a) The signal processing hardware of the DAS uses input modules, output modules, and microprocessor or special purpose logic processor boards that are different than those used in the PMS.

   b) The display hardware of the DAS uses a different display device than that used in the PMS.

   c) Software diversity between DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.

   d) The DAS has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency (RFI), and electrostatic discharge (ESD) conditions that exist where the DAS equipment is located in the plant.

   e) The sensors identified on Table 2.5.1-3 are used for DAS input and are separate from those being used by the PMS and plant control system.

   f) The DAS is powered by non-Class 1E uninterruptible power supplies that are independent and separate from the power supplies which power the PMS.

   g) The DAS signal processing cabinets are provided with the capability for channel testing without actuating the controlled components.

   h) The DAS equipment can withstand the room ambient temperature and humidity conditions that will exist at the plant locations in which the DAS equipment is installed at the times for which the DAS is designed to be operational.

4. The DAS hardware and any software are developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages:

   a) Development phase for hardware and any software

   b) System test phase

   c) Installation phase

   The planned design process also provides for the use of commercial off-the-shelf hardware and software.

5. The DAS manual actuation of ADS, IRWST injection, and containment recirculation can be executed correctly and reliably.

Table 2.5.1-1 Functions Automatically Actuated by the DAS
1. Reactor and Turbine Trip on Low Wide-range Steam Generator Water Level or Low Pressurizer Water Level or High Hot Leg Temperature
2. Passive Residual Heat Removal (PRHR) Actuation and In-containment Refueling Water Storage Tank (IRWST) Gutter Isolation on Low Wide-range Steam Generator Water Level or on High Hot Leg Temperature
3. Core Makeup Tank (CMT) Actuation and Trip All Reactor Coolant Pumps on Low Wide-Range Steam Generator Water Level or Low Pressurizer Water Level
4. Isolation of Selected Containment Penetrations and Initiation of Passive Containment Cooling System (PCS) on High Containment Temperature

Table 2.5.1-2 Functions Manually Actuated by the DAS
1. Reactor and Turbine Trip
2. PRHR Actuation and IRWST Gutter Isolation
3. CMT Actuation and Trip All Reactor Coolant Pumps
4. First-stage Automatic Depressurization System (ADS) Valve Actuation
5. Second-stage ADS Valve Actuation
6. Third-stage ADS Valve Actuation
7. Fourth-stage ADS Valve Actuation
8. PCS Actuation
9. Isolation of Selected Containment Penetrations
10. Containment Hydrogen Igniter Actuation
11. IRWST Injection Actuation
12. Containment Recirculation Actuation
13. Actuate IRWST Drain to Containment

Table 2.5.1-3
DAS Sensors and Displays

Equipment Name	Tag Number
Reactor Coolant System (RCS) Hot Leg Temperature	RCS-300A
RCS Hot Leg Temperature	RCS-300B
Steam Generator 1 Wide-range Level	SGS-044
Steam Generator 1 Wide-range Level	SGS-045
Steam Generator 2 Wide-range Level	SGS-046
Steam Generator 2 Wide-range Level	SGS-047
Pressurizer Water Level	RCS-305A
Pressurizer Water Level	RCS-305B

Table 2.5.1-3 DAS Sensors and Displays	
Equipment Name	**Tag Number**
Containment Temperature	VCS-053A
Containment Temperature	VCS-053B
Core Exit Temperature	IIS-009
Core Exit Temperature	IIS-013
Core Exit Temperature	IIS-030
Core Exit Temperature	IIS-034
Rod Control Motor Generator Voltage	PLS-001
Rod Control Motor Generator Voltage	PLS-002

Table 2.5.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
505	2.5.01.01	1. The functional arrangement of the DAS is as described in the Design Description of this Section 2.5.1.	Inspection of the as-built system will be performed.	The as-built DAS conforms with the functional arrangement as described in the Design Description of this Section 2.5.1.
506	2.5.01.02a	2.a) The DAS provides an automatic reactor trip on low wide-range steam generator water level, or on low pressurizer water level, or on high hot leg temperature, separate from the PMS.	Electrical power to the PMS equipment will be disconnected and an operational test of the as-built DAS will be performed using real or simulated test signals.	The generator field control relays (contained in the control cabinets for the rod drive motor-generator sets) open after the test signal reaches the specified limit.
507	2.5.01.02b	2.b) The DAS provides automatic actuation of selected functions, as identified in Table 2.5.1-1, separate from the PMS.	Electrical power to the PMS equipment will be disconnected and an operational test of the as-built DAS will be performed using real or simulated test signals.	Appropriate DAS output signals are generated after the test signal reaches the specified limit.
508	2.5.01.02c.i	2.c) The DAS provides manual initiation of reactor trip, and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypasses the control room multiplexers, if any; the PMS cabinets; and the signal processing equipment of the DAS.	Electrical power to the control room multiplexers, if any, and PMS equipment will be disconnected and the outputs from the DAS signal processing equipment will be disabled. While in this configuration, an operational test of the as-built system will be performed using the DAS manual actuation controls.	i) The generator field control relays (contained in the control cabinets for the rod drive motor-generator sets) open after reactor and turbine trip manual initiation controls are actuated.

Table 2.5.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
509	2.5.01.02c.ii	2.c) The DAS provides manual initiation of reactor trip, and selected functions, as identified in Table 2.5.1-2, separate from the PMS. These manual initiation functions are implemented in a manner that bypasses the control room multiplexers, if any; the PMS cabinets; and the signal processing equipment of the DAS.	Electrical power to the control room multiplexers, if any, and PMS equipment will be disconnected and the outputs from the DAS signal processing equipment will be disabled. While in this configuration, an operational test of the as-built system will be performed using the DAS manual actuation controls.	ii) DAS output signals are generated for the selected functions, as identified in Table 2.5.1-2, after manual initiation controls are actuated.
510	2.5.01.02d	2.d) The DAS provides MCR displays of selected plant parameters, as identified in Table 2.5.1-3, separate from the PMS.	Electrical power to the PMS equipment will be disconnected and inspection will be performed for retrievability of the selected plant parameters in the MCR.	The selected plant parameters can be retrieved in the MCR.
511	2.5.01.03a	3.a) The signal processing hardware of the DAS uses input modules, output modules, and microprocessor or special purpose logic processor boards that are different than those used in the PMS.	Inspection of the as-built DAS and PMS signal processing hardware will be performed.	The DAS signal processing equipment uses input modules, output modules, and micro-processor or special purpose logic processor boards that are different than those used in the PMS. The difference may be a different design, use of different component types, or different manufacturers.
512	2.5.01.03b	3.b) The display hardware of the DAS uses a different display device than that used in the PMS.	Inspection of the as-built DAS and PMS display hardware will be performed.	The DAS display hardware is different than the display hardware used in the PMS. The difference may be a different design, use of different component types, or different manufacturers.
513	2.5.01.03c	3.c) Software diversity between the DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.	Inspection of the DAS and PMS design documentation will be performed.	Any DAS algorithms, logic, program architecture, executable operating systems, and executable software/logic are different than those used in the PMS.
514	2.5.01.03d	3.d) The DAS has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency (RFI), and electrostatic discharge (ESD) conditions that exist where the DAS equipment is located in the plant.	Type tests, analyses, or a combination of type tests and analyses will be performed on the equipment.	A report exists and concludes that the DAS equipment can withstand the SWC, EMI, RFI and ESD conditions that exist where the DAS equipment is located in the plant.

Table 2.5.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
515	2.5.01.03e	3.e) The sensors identified on Table 2.5.1-3 are used for DAS input and are separate from those being used by the PMS and plant control system.	Inspection of the as-built system will be performed.	The sensors identified on Table 2.5.1-3 are used by DAS and are separate from those being used by the PMS and plant control system.
516	2.5.01.03f	3.f) The DAS is powered by non-Class 1E uninterruptible power supplies that are independent and separate from the power supplies which power the PMS.	Electrical power to the PMS equipment will be disconnected. While in this configuration, a test will be performed by providing simulated test signals in the non-Class 1E uninterruptible power supplies.	A simulated test signal exists at the DAS equipment when the assigned non-Class 1E uninterruptible power supply is provided the test signal.
517	2.5.01.03g	3.g) The DAS signal processing cabinets are provided with the capability for channel testing without actuating the controlled components.	Channel tests will be performed on the as built system.	The capability exists for testing individual DAS channels without propagating an actuation signal to a DAS controlled component.
518	2.5.01.03h	3.h) The DAS equipment can withstand the room ambient temperature and humidity conditions that will exist at the plant locations in which the DAS equipment is installed at the times for which the DAS is designed to be operational.	Type tests, analyses, or a combination of type tests and analyses will be performed on the equipment.	A report exists and concludes that the DAS equipment can withstand the room ambient temperature and humidity conditions that will exist at the plant locations in which the DAS equipment is installed at the times for which the DAS is designed to be operational.
519	2.5.01.04	4. The DAS hardware and any software are developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages: a) Development phase for hardware and any software b) System test phase c) Installation phase The planned design process also provides for the use of commercial off-the-shelf hardware and software.	Inspection will be performed of the process used to design the hardware and any software.	A report exists and concludes that the process defines the organizational responsibilities, activities, and configuration management controls for the following: a) Documentation and review of hardware and any software. b) Performance of tests and the documentation of test results during the system test phase. c) Performance of tests and inspections during the installation phase. The process also defines requirements for the use of commercial off-the-shelf hardware and software.

Table 2.5.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
520	2.5.01.05	5. The DAS manual actuation of ADS, IRWST injection, and containment recirculation can be executed correctly and reliably.	See ITAAC Table 3.2-1, item 1.	See ITAAC Table 3.2-1, item 1.

Table 2.5.1-5		
Component Name	**Tag No.**	**Component Location**
DAS Processor Cabinet 1	DAS-JD-001	Auxiliary Building
DAS Processor Cabinet 2	DAS-JD-002	Auxiliary Building
DAS Squib Valve Control Cabinet	DAS-JD-003	Auxiliary Building

2.5.2 Protection and Safety Monitoring System

Design Description

The protection and safety monitoring system (PMS) initiates reactor trip and actuation of engineered safety features in response to plant conditions monitored by process instrumentation and provides safety-related displays. The PMS has the equipment identified in Table 2.5.2-1. The PMS has four divisions of Reactor Trip and Engineered Safety Features Actuation, and four divisions of safety-related post-accident parameter displays. The functional arrangement of the PMS is depicted in Figure 2.5.2-1 and the component locations of the PMS are as shown in Table 2.5.2-9.

1. The functional arrangement of the PMS is as described in the Design Description of this Section 2.5.2.
2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis loads without loss of safety function.
3. The Class 1E equipment, identified in Table 2.5.2-1, has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
4. The Class 1E equipment, identified in Table 2.5.2-1, can withstand the room ambient temperature, humidity, pressure, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

5. a) The Class 1E equipment, identified in Table 2.5.2-1, is powered from its respective Class 1E division.

   b) Separation is provided between PMS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

6. The PMS provides the following safety-related functions:

   a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.

   b) The PMS initiates automatic actuation of engineered safety features, as identified in Table 2.5.2-3, when plant process signals reach specified limits.

   c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.

7. The PMS provides the following nonsafety-related functions:

   a) The PMS provides process signals to the plant control system (PLS) through isolation devices.

   b) The PMS provides process signals to the data display and processing system (DDS) through isolation devices.

   c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.

   d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

   e) The PMS receives signals from non-safety equipment that provides interlocks for PMS test functions through isolation devices.

8. The PMS, in conjunction with the operator workstations, provides the following functions:

   a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the main control room (MCR). The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.

   b) The PMS provides for the transfer of control capability from the MCR to the remote shutdown workstation (RSW) using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related group or with nonsafety-related control capability.

   c) Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.

9. a) The PMS automatically removes blocks of reactor trip and engineered safety features actuation when the plant approaches conditions for which the associated function is designed to provide protection. These blocks are identified in Table 2.5.2-6.

b) The PMS two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. All bypassed channels are alarmed in the MCR.

c) The PMS does not allow simultaneous bypass of two redundant channels.

d) The PMS provides the interlock functions identified in Table 2.5.2-7.

10. Setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.

11. The PMS hardware and software is developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages:

    a) Design requirements phase, may be referred to as conceptual or project definition phase (Complete)

    b) System definition phase

    c) Hardware and software development phase, consisting of hardware and software design and implementation

    d) System integration and test phase

    e) Installation phase

12. The PMS software is designed, tested, installed, and maintained using a process which incorporates a graded approach according to the relative importance of the software to safety and specifies requirements for:

    a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action.

    b) Software configuration management including historical records of software and control of software changes.

    c) Verification and validation including requirements for reviewer independence.

13. The use of commercial grade hardware and software items in the PMS is accomplished through a process that specifies requirements for:

    a) Review of supplier design control, configuration management, problem reporting, and change control.

    b) Review of product performance.

    c) Receipt acceptance of the commercial grade item.

    d) Final acceptance based on equipment qualification and software validation in the integrated system.

14. The Component Interface Module CIM) is developed using a planned design process which provides for specific design documentation and reviews.

Table 2.5.2-1 PMS Equipment Name and Classification			
Equipment Name	**Seismic Cat. I**	**Class 1E**	**Qual. for Harsh Envir.**
PMS Cabinets, Division A	Yes	Yes	No
PMS Cabinets, Division B	Yes	Yes	No
PMS Cabinets, Division C	Yes	Yes	No
PMS Cabinets, Division D	Yes	Yes	No
Reactor Trip Switchgear, Division A	Yes	Yes	No
Reactor Trip Switchgear, Division B	Yes	Yes	No
Reactor Trip Switchgear, Division C	Yes	Yes	No
Reactor Trip Switchgear, Division D	Yes	Yes	No
MCR/RSW Transfer Panels	Yes	Yes	No
MCR Safety-related Display, Division A	Yes	Yes	No
MCR Safety-related Display, Division B	Yes	Yes	No
MCR Safety-related Display, Division C	Yes	Yes	No
MCR Safety-related Display, Division D	Yes	Yes	No
MCR Safety-related Controls	Yes	Yes	No

Table 2.5.2-2 PMS Automatic Reactor Trips
Source Range High Neutron Flux Reactor Trip Intermediate Range High Neutron Flux Reactor Trip Power Range High Neutron Flux (Low Setpoint) Trip Power Range High Neutron Flux (High Setpoint) Trip Power Range High Positive Flux Rate Trip Reactor Coolant Pump High Bearing Water Temperature Trip Overtemperature Delta-T Trip Overpower Delta-T Trip Pressurizer Low Pressure Trip Pressurizer High Pressure Trip Pressurizer High Water Level Trip Low Reactor Coolant Flow Trip Low Reactor Coolant Pump Speed Trip Low Steam Generator Water Level Trip High-2 Steam Generator Water Level Trip Automatic or Manual Safeguards Actuation Trip Automatic or Manual Depressurization System Actuation Trip Automatic or Manual Core Makeup Tank (CMT) Injection Trip Passive Residual Heat Removal (PRHR) Actuation Reactor Trip

Table 2.5.2-3 **PMS Automatically Actuated Engineered Safety Features**
Safeguards Actuation Containment Isolation Automatic Depressurization System (ADS) Actuation Main Feedwater Isolation Reactor Coolant Pump Trip CMT Injection Turbine Trip (Isolated signal to nonsafety equipment) Steam Line Isolation Steam Generator Relief Isolation Steam Generator Blowdown Isolation Passive Containment Cooling Actuation Startup Feedwater Isolation Passive Residual Heat Removal (PRHR) Heat Exchanger Alignment Block of Boron Dilution Chemical and Volume Control System (CVS) Makeup Line Isolation Steam Dump Block (Isolated signal to nonsafety equipment) MCR Isolation and Air Supply Initiation Auxiliary Spray and Letdown Purification Line Isolation Containment Air Filtration System Isolation Normal Residual Heat Removal Isolation Refueling Cavity Isolation In-Containment Refueling Water Storage Tank (IRWST) Injection IRWST Containment Recirculation CVS Letdown Isolation Pressurizer Heater Block (Isolated signal to nonsafety equipment) Containment Vacuum Relief

Table 2.5.2-4 **PMS Manually Actuated Functions**
Reactor Trip Safeguards Actuation Containment Isolation Depressurization System Stages 1, 2, and 3 Actuation Depressurization System Stage 4 Actuation Feedwater Isolation Core Makeup Tank Injection Actuation Steam Line Isolation Passive Containment Cooling Actuation Passive Residual Heat Removal Heat Exchanger Alignment IRWST Injection Containment Recirculation Actuation Control Room Isolation and Air Supply Initiation Steam Generator Relief Isolation Chemical and Volume Control System Isolation Normal Residual Heat Removal System Isolation Containment Vacuum Relief

Table 2.5.2-5 Minimum Inventory of Displays, Alerts, and Fixed Position Controls in the MCR			
Description	**Control**	**Display**	**Alert(1)**
Neutron Flux	-	Yes	Yes
Neutron Flux Doubling(2)	-	No	Yes
Startup Rate	-	Yes	Yes
Reactor Coolant System (RCS) Pressure	-	Yes	Yes
Wide-range Hot Leg Temperature	-	Yes	No
Wide-range Cold Leg Temperature	-	Yes	Yes
RCS Cooldown Rate Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Wide-range Cold Leg Temperature Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Change of RCS Temperature by more than 5°F in the last 10 minutes	-	No	Yes
Containment Water Level	-	Yes	Yes
Containment Pressure	-	Yes	Yes
Pressurizer Water Level	-	Yes	Yes
Pressurizer Water Level Trend	-	Yes	No
Pressurizer Reference Leg Temperature	-	Yes	No
Reactor Vessel-Hot Leg Water Level	-	Yes	Yes
Pressurizer Pressure	-	Yes	No
Core Exit Temperature	-	Yes	Yes
RCS Subcooling	-	Yes	Yes
RCS Cold Overpressure Limit	-	Yes	Yes
IRWST Water Level	-	Yes	Yes
PRHR Flow	-	Yes	Yes
PRHR Outlet Temperature	-	Yes	Yes

Note: Dash (-) indicates not applicable.

1. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the main control room, the visual alerts are embedded in the safety-related displays as visual signals.

Table 2.5.2-5 (cont.) Minimum Inventory of Displays, Alerts, and Fixed Position Controls in the MCR			
Description	**Control**	**Display**	**Alert(1)**
Passive Containment Cooling System (PCS) Storage Tank Water Level	-	Yes	No
PCS Cooling Flow	-	Yes	No
IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status(2)	-	Yes	Yes
Remotely Operated Containment Isolation Valve Status(2)	-	Yes	No
Containment Area High-range Radiation Level	-	Yes	Yes
Containment Pressure (Extended Range)	-	Yes	No
CMT Level	-	Yes	No
Manual Reactor Trip (also initiates turbine trip)	Yes	-	-
Manual Safeguards Actuation	Yes	-	-
Manual CMT Actuation	Yes	-	-
Manual MCR Emergency Habitability System Actuation	Yes	-	-
Manual ADS Stages 1, 2, and 3 Actuation	Yes	-	-
Manual ADS Stage 4 Actuation	Yes	-	-
Manual PRHR Actuation	Yes	-	-
Manual Containment Cooling Actuation	Yes	-	-
Manual IRWST Injection Actuation	Yes	-	-
Manual Containment Recirculation Actuation	Yes	-	-
Manual Containment Isolation	Yes	-	-
Manual Main Steam Line Isolation	Yes	-	-
Manual Feedwater Isolation	Yes	-	-
Manual Containment Vacuum Relief	Yes		

Note: Dash (-) indicates not applicable.

2. These instruments are not required after 24 hours.

Table 2.5.2-6 PMS Blocks
Reactor Trip Functions: Source Range High Neutron Flux Reactor Trip Intermediate Range High Neutron Flux Reactor Trip Power Range High Neutron Flux (Low Setpoint) Trip Pressurizer Low Pressure Trip Pressurizer High Water Level Trip Low Reactor Coolant Flow Trip Low Reactor Coolant Pump Speed Trip High Steam Generator Water Level Trip Engineered Safety Features: Automatic Safeguards Containment Isolation Main Feedwater Isolation Reactor Coolant Pump Trip Core Makeup Tank Injection Steam Line Isolation Startup Feedwater Isolation Block of Boron Dilution Chemical and Volume Control System Isolation Chemical and Volume Control System Letdown Isolation Steam Dump Block Auxiliary Spray and Letdown Purification Line Isolation Passive Residual Heat Removal Heat Exchanger Alignment Normal Residual Heat Removal System Isolation

Table 2.5.2-7 PMS Interlocks
RNS Suction Valves PRHR Heat Exchanger Inlet Isolation Valve CMT Cold Leg Balance Line Isolation Valves Containment Vacuum Relief Isolation Valves

Table 2.5.2-8
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
521	2.5.02.01	1. The functional arrangement of the PMS is as described in the Design Description of this Section 2.5.2.	Inspection of the as-built system will be performed.	The as-built PMS conforms with the functional arrangement as described in the Design Description of this Section 2.5.2.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
522	2.5.02.02.i	2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.5.2-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.5.2-1 is located on the Nuclear Island.
523	2.5.02.02.ii	2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
524	2.5.02.02.iii	2. The seismic Category I equipment, identified in Table 2.5.2-1, can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
525	2.5.02.03	3. The Class 1E equipment, identified in Table 2.5.2-1, has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on the equipment.	A report exists and concludes that the Class 1E equipment identified in Table 2.5.2-1 can withstand the SWC, EMI, RFI, and ESD conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
526	2.5.02.04	4. The Class 1E equipment, identified in Table 2.5.2-1, can withstand the room ambient temperature, humidity, pressure, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on the Class 1E equipment identified in Table 2.5.2-1.	A report exists and concludes that the Class 1E equipment identified in Table 2.5.2-1 can withstand the room ambient temperature, humidity, pressure, and mechanical vibration conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
527	2.5.02.05a	5.a) The Class 1E equipment, identified in Table 2.5.2-1, is powered from its respective Class 1E division.	Tests will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.5.2-1 when the assigned Class 1E division is provided the test signal.
528	2.5.02.05b	5.b) Separation is provided between PMS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, items 7.d and 7.e.	See ITAAC Table 3.3-6, items 7.d and 7.e.
529	2.5.02.06a.i	6.a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.	An operational test of the as-built PMS will be performed using real or simulated test signals.	i) The reactor trip switchgear opens after the test signal reaches the specified limit. This only needs to be verified for one automatic reactor trip function.
530	2.5.02.06a.ii	6.a) The PMS initiates an automatic reactor trip, as identified in Table 2.5.2-2, when plant process signals reach specified limits.	An operational test of the as-built PMS will be performed using real or simulated test signals.	ii) PMS output signals to the reactor trip switchgear are generated after the test signal reaches the specified limit. This needs to be verified for each automatic reactor trip function.
531	2.5.02.06b	6.b) The PMS initiates automatic actuation of engineered safety features, as identified in Table 2.5.2-3, when plant process signals reach specified limits.	An operational test of the as-built PMS will be performed using real or simulated test signals.	Appropriate PMS output signals are generated after the test signal reaches the specified limit. These output signals remain following removal of the test signal. Tests from the actuation signal to the actuated device(s) are performed as part of the system-related inspection, test, analysis, and acceptance criteria.
532	2.5.02.06c.i	6.c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.	An operational test of the as-built PMS will be performed using the PMS manual actuation controls.	i) The reactor trip switchgear opens after manual reactor trip controls are actuated.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
533	2.5.02.06c.ii	6.c) The PMS provides manual initiation of reactor trip and selected engineered safety features as identified in Table 2.5.2-4.	An operational test of the as-built PMS will be performed using the PMS manual actuation controls.	ii) PMS output signals are generated for reactor trip and selected engineered safety features as identified in Table 2.5.2-4 after the manual initiation controls are actuated.
534	2.5.02.07a	7.a) The PMS provides process signals to the PLS through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
535	2.5.02.07b	7.b) The PMS provides process signals to the DDS through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
536	2.5.02.07c	7.c) Data communication between safety and nonsafety systems does not inhibit the performance of the safety function.	Type tests, analyses, or a combination of type tests and analyses of the PMS gateways will be performed.	A report exists and concludes that data communication between safety and nonsafety systems does not inhibit the performance of the safety function.
537	2.5.02.07d	7.d) The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.	Type tests, analyses, or a combination of type tests and analyses of the PMS manual control circuits and algorithms will be performed.	A report exists and concludes that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.
538	2.5.02.07e	7.e) The PMS receives signals from non-safety equipment that provides interlocks for PMS test functions through isolation devices.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.
539	2.5.02.08a.i	8.a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the MCR. The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.	i) An inspection will be performed for retrievability of plant parameters in the MCR.	i) The plant parameters listed in Table 2.5.2-5 with a "Yes" in the "Display" column, can be retrieved in the MCR.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
540	2.5.02.08a.ii	8.a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the MCR. The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.	ii) An inspection and test will be performed to verify that the plant parameters are used to generate visual alerts that identify challenges to critical safety functions.	ii) The plant parameters listed in Table 2.5.2-5 with a "Yes" in the "Alert" column are used to generate visual alerts that identify challenges to critical safety functions. The visual alerts actuate in accordance with their correct logic and values.
541	2.5.02.08a.iii	8.a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved in the MCR. The fixed position controls listed with a "Yes" in the "Control" column are provided in the MCR.	iii) An operational test of the as-built system will be performed using each MCR fixed position control.	iii) For each test of an as-built fixed position control listed in Table 2.5.2-5 with a "Yes" in the "Control" column, an actuation signal is generated. Tests from the actuation signal to the actuated device(s) are performed as part of the system-related inspection, test, analysis and acceptance criteria.
542	2.5.02.08b.i	8.b) The PMS provides for the transfer of control capability from the MCR to the RSW using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related group or with nonsafety-related control capability.	i) An inspection will be performed to verify that a transfer switch exists for each safety-related division and the nonsafety-related control capability.	i) A transfer switch exists for each safety-related division and the nonsafety-related control capability.
543	2.5.02.08b.ii	8.b) The PMS provides for the transfer of control capability from the MCR to the RSW using multiple transfer switches. Each individual transfer switch is associated with only a single safety-related group or with nonsafety-related control capability.	ii) An operational test of the as-built system will be performed to demonstrate the transfer of control capability from the MCR to the RSW.	ii) Actuation of each transfer switch results in an alarm in the MCR and RSW, the activation of operator control capability from the RSW, and the deactivation of operator control capability from the MCR for the associated safety-related division and nonsafety-related control capability.
544	2.5.02.08c	8.c) Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.	Inspection will be performed for retrievability of displays of the open/closed status of the reactor trip breakers in the MCR.	Displays of the open/closed status of the reactor trip breakers can be retrieved in the MCR.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
545	2.5.02.09a	9.a) The PMS automatically removes blocks of reactor trip and engineered safety features actuation when the plant approaches conditions for which the associated function is designed to provide protection. These blocks are identified in Table 2.5.2-6.	An operational test of the as-built PMS will be performed using real or simulated test signals.	The PMS blocks are automatically removed when the test signal reaches the specified limit.
546	2.5.02.09b	9.b) The PMS two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. All bypassed channels are alarmed in the MCR.	An operational test of the as-built PMS will be performed.	The PMS two-out-of-four initiation logic reverts to a two-out-of-three coincidence logic if one of the four channels is bypassed. All bypassed channels are alarmed in the MCR.
547	2.5.02.09c	9.c) The PMS does not allow simultaneous bypass of two redundant channels.	An operational test of the as-built PMS will be performed. With one channel in bypass, an attempt will be made to place a redundant channel in bypass.	The redundant channel cannot be placed in bypass.
548	2.5.02.09d	9.d) The PMS provides the interlock functions identified in Table 2.5.2-7.	An operational test of the as-built PMS will be performed using real or simulated test signals.	Appropriate PMS output signals are generated as the interlock conditions are changed.
549	2.5.02.10	10. Setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.	Inspection will be performed for a document that describes the methodology and input parameters used to determine the PMS setpoints.	A report exists and concludes that the PMS setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
550	2.5.02.11	11. The PMS hardware and software is developed using a planned design process which provides for specific design documentation and reviews during the following life cycle stages: a) Not used b) System definition phase c) Hardware and software development phase, consisting of hardware and software design and implementation d) System integration and test phase e) Installation phase	Inspection will be performed of the process used to design the hardware and software.	A report exists and concludes that the process defines the organizational responsibilities, activities, and configuration management controls for the following: a) Not used. b) Specification of functional requirements. c) Documentation and review of hardware and software. d) Performance of system tests and the documentation of system test results, including a response time test performed under maximum CPU loading to demonstrate that the PMS can fulfill its response time criteria. e) Performance of installation tests and inspections.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
551	2.5.02.12	12. The PMS software is designed, tested, installed, and maintained using a process which incorporates a graded approach according to the relative importance of the software to safety and specifies requirements for: a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action. b) Software configuration management including historical records of software and control of software changes. c) Verification and validation including requirements for reviewer independence.	Inspection will be performed of the process used to design, test, install, and maintain the PMS software.	A report exists and concludes that the process establishes a method for classifying the PMS software elements according to their relative importance to safety and specifies requirements for software assigned to each safety classification. The report also concludes that requirements are provided for the following software development functions: a) Software management including documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action. Software management requirements may be documented in the software quality assurance plan, software management plan, software development plan, software safety plan, and software operation and maintenance plan; or these requirements may be combined into a single software management plan. b) Software configuration management including historical records of software and control of software changes. Software configuration management requirements are provided in the software configuration management plan. c) Verification and validation including requirements for reviewer independence. Verification and validation requirements are provided in the verification and validation plan.

Table 2.5.2-8 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
552	2.5.02.13	13. The use of commercial grade computer hardware and software items in the PMS is accomplished through a process that specifies requirements for: a) Review of supplier design control, configuration management, problem reporting, and change control. b) Review of product performance. c) Receipt acceptance of the commercial grade item. d) Acceptance based on equipment qualification and software validation in the integrated system.	Inspection will be performed of the process defined to use commercial grade components in the application.	A report exists and concludes that the process has requirements for: a) Review of supplier design control, configuration management, problem reporting, and change control. b) Review of product performance. c) Receipt acceptance of the commercial grade item. d) Acceptance based on equipment qualification and software validation in the integrated system.
553	2.5.02.14	14. The Component Interface Module (CIM) is developed using a planned design process which provides for specific design documentation and reviews. {Design Acceptance Criteria}	An inspection and or an audit will be performed of the processes used to design the hardware, development software, qualification and testing.	A report exists and concludes that CIM meets the below listed life cycle stages. Life cycle stages: a. Design requirements phase, may be referred to as conceptual or project definition phase b. System definition phase c. Hardware and software development phase, consisting of hardware and software design and implementation d. System integration and test phase e. Installation phase

Table 2.5.2-9	
Component Name	**Component Location**
PMS Cabinets, Division A	Auxiliary Building
PMS Cabinets, Division B	Auxiliary Building
PMS Cabinets, Division C	Auxiliary Building
PMS Cabinets, Division D	Auxiliary Building
Reactor Trip Switchgear, Division A	Auxiliary Building
Reactor Trip Switchgear, Division B	Auxiliary Building
Reactor Trip Switchgear, Division C	Auxiliary Building
Reactor Trip Switchgear, Division D	Auxiliary Building
MCR/RSW Transfer Panels	Auxiliary Building
MCR Safety-related Displays	Auxiliary Building
MCR Safety-related Controls	Auxiliary Building

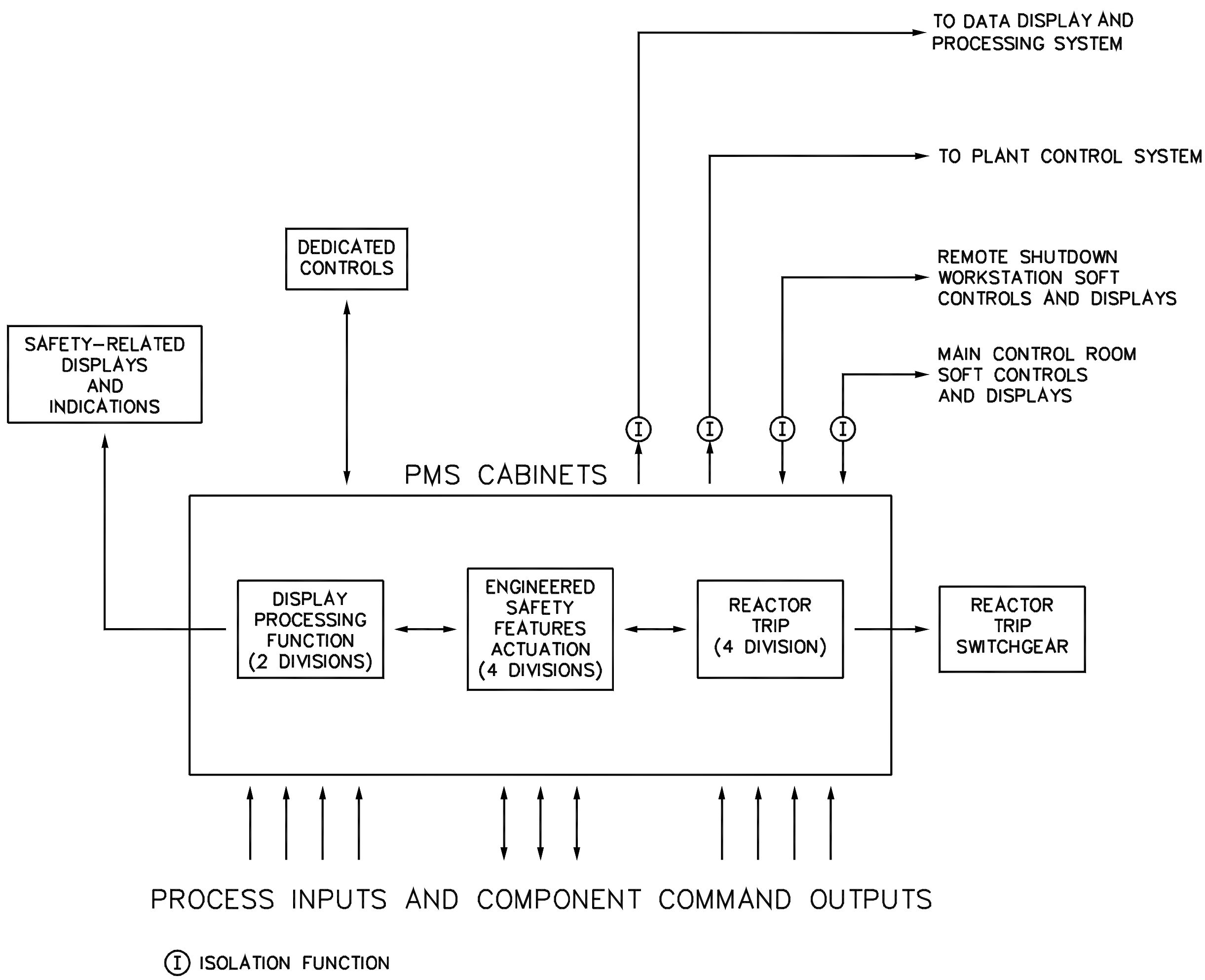


Figure 2.5.2-1
Protection and Safety Monitoring System

2.5.3 Plant Control System

Design Description

The plant control system (PLS) provides for automatic and manual control of nonsafety-related plant components during normal and emergency plant operations. The PLS has distributed controllers and operator controls interconnected by computer data links or data highways.

1. The functional arrangement of the PLS is as described in the Design Description of this Section 2.5.3.
2. The PLS provides control interfaces for the control functions listed in Table 2.5.3-1.

Table 2.5.3-1
Control Functions Supported by the PLS

1. Reactor Power	5. Steam Generator Feedwater
2. Reactor Rod Position	6. Steam Dump
3. Pressurizer Pressure	7. Rapid Power Reduction
4. Pressurizer Water Level	

Table 2.5.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
554	2.5.03.01	1. The functional arrangement of the PLS is as described in the Design Description of this Section 2.5.3.	Inspection of the as-built system will be performed.	The as-built PLS conforms with the functional arrangement as described in the Design Description of this Section 2.5.3.
555	2.5.03.02	2. The PLS provides control interfaces for the control functions listed in Table 2.5.3-1.	An operational test of the system will be performed using simulated input signals. System outputs or component operations will be monitored to determine the operability of the control functions.	The PLS provides control interfaces for the control functions listed in Table 2.5.3-1.

2.5.4 Data Display and Processing System

Design Description

The data display and processing system (DDS) provides nonsafety-related alarms and displays, analysis of plant data, plant data logging and historical storage and retrieval, and operational support for plant personnel. The DDS has distributed computer processors and video display units to support the data processing and display functions.

1. The functional arrangement of the DDS is as described in the Design Description of this Section 2.5.4.
2. The DDS, in conjunction with the operator workstations, provides the following function:

   The DDS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.4-1. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved at the remote shutdown workstation (RSW). The controls listed with a "Yes" in the "Control" column are provided at the RSW.
3. The DDS provides information pertinent to the status of the protection and safety monitoring system.
4. The plant operating instrumentation installed for feedwater flow measurement is one that has been specifically approved by the NRC; the power calorimetric uncertainty calculation includes uncertainties for the associated instrumentation based on an NRC approved methodology; and the calculated calorimetric values are bounded by the uncertainty value assumed for the initial reactor power in the safety analysis.

Table 2.5.4-1 Minimum Inventory of Controls, Displays, and Alerts at the RSW			
Description	**Control**	**Display**	**Alert(1)**
Neutron Flux	-	Yes	Yes
Neutron Flux Doubling	-	No	Yes
Startup Rate	-	Yes	Yes
Reactor Coolant System (RCS) Pressure	-	Yes	Yes
Wide-range Hot Leg Temperature	-	Yes	No
Wide-range Cold Leg Temperature	-	Yes	Yes
RCS Cooldown Rate Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Wide-range Cold Leg Temperature Compared to the Limit Based on RCS Pressure	-	Yes	Yes
Change of RCS Temperature by more than 5°F in the last 10 minutes	-	No	Yes
Containment Water Level	-	Yes	Yes
Containment Pressure	-	Yes	Yes
Pressurizer Water Level	-	Yes	Yes
Pressurizer Water Level Trend	-	Yes	No
Pressurizer Reference Leg Temperature	-	Yes	No
Reactor Vessel-Hot Leg Water Level	-	Yes	Yes
Pressurizer Pressure	-	Yes	No
Core Exit Temperature	-	Yes	Yes
RCS Subcooling	-	Yes	Yes
RCS Cold Overpressure Limit	-	Yes	Yes
In-containment Refueling Water Storage Tank (IRWST) Water Level	-	Yes	Yes
Passive Residual Heat Removal (PRHR) Flow	-	Yes	Yes

Note: Dash (-) indicates not applicable.

1. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the RSW, the visual alerts are embedded in the nonsafety-related displays as visual signals.

Table 2.5.4-1 (cont.) Minimum Inventory of Controls, Displays, and Alerts at the RSW			
Description	**Control**	**Display**	**Alert[1]**
PRHR Outlet Temperature	-	Yes	Yes
Passive Containment Cooling System (PCS) Storage Tank Water Level	-	Yes	No
PCS Cooling Flow	-	Yes	No
IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status	-	Yes	Yes
Remotely Operated Containment Isolation Valve Status	-	Yes	No
Containment Area High-range Radiation Level	-	Yes	Yes
Containment Pressure (Extended Range)	-	Yes	No
Core Makeup Tank (CMT) Level	-	Yes	No
Manual Reactor Trip (also initiates turbine trip)	Yes	-	-
Manual Safeguards Actuation	Yes	-	-
Manual CMT Actuation	Yes	-	-
Manual Automatic Depressurization System (ADS) Stages 1, 2, and 3 Actuation	Yes	-	-
Manual ADS Stage 4 Actuation	Yes	-	-
Manual PRHR Actuation	Yes	-	-
Manual Containment Cooling Actuation	Yes	-	-
Manual IRWST Injection Actuation	Yes	-	-
Manual Containment Recirculation Actuation	Yes	-	-
Manual Containment Isolation	Yes	-	-
Manual Main Steam Line Isolation	Yes	-	-
Manual Feedwater Isolation	Yes	-	-
Manual Containment Hydrogen Igniter (Nonsafety-related)[2]	Yes	-	-

Note: Dash (-) indicates not applicable.

1. These parameters are used to generate visual alerts that identify challenges to the critical safety functions. For the RSW, the visual alerts are embedded in the nonsafety-related displays as visual signals.
2. Containment hydrogen igniter control is provided as a “soft” control.

Table 2.5.4-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
556	2.5.04.01	1. The functional arrangement of the DDS is as described in the Design Description of this Section 2.5.4.	Inspection of the as-built system will be performed.	The as-built DDS conforms with the functional arrangement as described in the Design Description of this Section 2.5.4.
557	2.5.04.02.i	2. The DDS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.4-1. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved at the RSW. The controls listed with a "Yes" in the "Control" column are provided at the RSW.	i) An inspection will be performed for retrievability of plant parameters at the RSW.	i) The plant parameters listed in Table 2.5.4-1 with a "Yes" in the "Display" column can be retrieved at the RSW.
558	2.5.04.02.ii	2. The DDS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.4-1. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved at the RSW. The controls listed with a "Yes" in the "Control" column are provided at the RSW.	ii) An inspection and test will be performed to verify that the plant parameters are used to generate visual alerts that identify challenges to critical safety functions.	ii) The plant parameters listed in Table 2.5.4-1 with a "Yes" in the "Alert" column are used to generate visual alerts that identify challenges to critical safety functions. The visual alerts actuate in accordance with their logic and values.
559	2.5.04.02.iii	2. The DDS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.4-1. The plant parameters listed with a "Yes" in the "Display" column and visual alerts listed with a "Yes" in the "Alert" column can be retrieved at the RSW. The controls listed with a "Yes" in the "Control" column are provided at the RSW.	iii) An operational test of the as-built system will be performed using each RSW control.	iii) For each test of a control listed in Table 2.5.4-1 with a "Yes" in the "Control" column, an actuation signal is generated. Tests from the actuation signal to the actuated device(s) are performed as part of the system-related inspection, test, analysis and acceptance criteria.
560	2.5.04.03	3. The DDS provides information pertinent to the status of the protection and safety monitoring system.	Tests of the as-built system will be performed.	The as-built system provides displays of the bypassed and operable status of the protection and safety monitoring system.

Table 2.5.4-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
561	C.2.5.04.04a	4. The plant calorimetric uncertainty and plant instrumentation performance is bounded by the 1% calorimetric uncertainty value assumed for the initial reactor power in the safety analysis.	Inspection will be performed of the plant operating instrumentation installed for feedwater flow measurement, its associated power calorimetric uncertainty calculation, and the calculated calorimetric values.	a) The as-built system takes input for feedwater flow measurement from a Caldon [Cameron] LEFM CheckPlus™ System;
562	C.2.5.04.04b	4. The plant calorimetric uncertainty and plant instrumentation performance is bounded by the 1% calorimetric uncertainty value assumed for the initial reactor power in the safety analysis.	Inspection will be performed of the plant operating instrumentation installed for feedwater flow measurement, its associated power calorimetric uncertainty calculation, and the calculated calorimetric values.	b) The power calorimetric uncertainty calculation documented for that instrumentation is based on an accepted Westinghouse methodology and the uncertainty values for that instrumentation are not lower than those for the actual installed instrumentation; and
563	C.2.5.04.04c	4. The plant calorimetric uncertainty and plant instrumentation performance is bounded by the 1% calorimetric uncertainty value assumed for the initial reactor power in the safety analysis.	Inspection will be performed of the plant operating instrumentation installed for feedwater flow measurement, its associated power calorimetric uncertainty calculation, and the calculated calorimetric values.	c) The calculated calorimetric power uncertainty measurement values are bounded by the 1% uncertainty value assumed for the initial reactor power in the safety analysis.

2.5.5 In-Core Instrumentation System

Design Description

The in-core instrumentation system (IIS) provides safety-related core exit thermocouple signals to the protection and safety monitoring system (PMS). The IIS also provides nonsafety-related core exit thermocouple signals to the diverse actuation system (DAS). The core exit thermocouples are housed in the core instrument assemblies. Multiple core instrument assemblies are used to provide radial coverage of the core. At least three core instrument assemblies are provided in each core quadrant.

1. The functional arrangement of the IIS is as described in the Design Description of this Section 2.5.5.
2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis loads without loss of safety function.
3. a) The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function.

b) The Class 1E cables between the Incore Thermocouple elements and the connector boxes located on the integrated head package have sheaths.

c) For cables other than those covered by 3.b, separation is provided between IIS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

4. Safety-related displays of the parameters identified in Table 2.5.5-1 can be retrieved in the main control room (MCR).

Table 2.5.5-1

Equipment Name	Seismic Cat. I	ASME Code Classification	Class 1E	Qual. for Harsh Envir.	Safety-Related Display
Incore Thimble Assemblies (at least three assemblies in each core quadrant)	Yes	–	Yes(1)	Yes(1)	Core Exit Temperature(1)

Note: Dash (-) indicates not applicable.

1. Only applies to the safety-related assemblies. There are at least two safety-related assemblies in each core quadrant.

Table 2.5.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
564	2.5.05.01	1. The functional arrangement of the IIS is as described in the Design Description of this Section 2.5.5.	Inspection of the as-built system will be performed.	The as-built IIS conforms with the functional arrangement as described in the Design Description of this Section 2.5.5.
565	2.5.05.02.i	2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis dynamic loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.5.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.5.5-1 is located on the Nuclear Island.
566	2.5.05.02.ii	2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis dynamic loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function.

Table 2.5.5-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
567	2.5.05.02.iii	2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis dynamic loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
568	2.5.05.03a.i	3.a) The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function.	i) Type tests, analysis, or a combination of type tests and analysis will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that the Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment. This equipment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
569	2.5.05.03a.ii	3.a) The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.5.5-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
570	2.5.05.03b	3.b) The Class 1E cables between the Incore Thermocouple elements and the connector boxes located on the integrated head package have sheaths.	Inspection of the as-built system will be performed.	The as-built Class 1E cables between the Incore Thermocouple elements and the connector boxes located on the integrated head package have sheaths.
571	2.5.05.03c	3.c) For cables other than those covered by 3.b, separation is provided between IIS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
572	2.5.05.04	4. Safety-related displays of the parameters identified in Table 2.5.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.5.5-1 can be retrieved in the MCR.

2.5.6 Special Monitoring System

Design Description

The special monitoring system (SMS) monitors the reactor coolant system (RCS) for the occurrence of impacts characteristic of metallic loose parts. Metal impact monitoring sensors are provided to monitor the RCS at the upper and lower head region of the reactor pressure vessel, and at the reactor coolant inlet region of each steam generator.

1. The functional arrangement of the SMS is as described in the Design Description of this Section 2.5.6.
2. Data obtained from the metal impact monitoring sensors can be retrieved in the main control room (MCR).

Table 2.5.6-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
573	2.5.06.01	1. The functional arrangement of the SMS is as described in the Design Description of this Section 2.5.6.	Inspection of the as-built system will be performed.	The as-built SMS conforms with the functional arrangement as described in the Design Description of this Section 2.5.6.
574	2.5.06.02	2. Data obtained from the metal impact monitoring sensors can be retrieved in the MCR.	Inspection will be performed for retrievability of data from the metal impact monitoring sensors in the MCR.	Data obtained from the metal impact monitoring sensors can be retrieved in the MCR.

2.5.7 Operation and Control Centers System

No ITAAC for this system.

2.5.8 Radiation Monitoring System

No entry. Radiation monitoring function covered in Section 3.5, Radiation Monitoring.

2.5.9 Seismic Monitoring System

Design Description

The seismic monitoring system (SJS) provides for the collection of seismic data in digital format, analysis of seismic data, notification of the operator if the ground motion exceeds a threshold value, and notification of the operator (after analysis of data) that a predetermined cumulative absolute velocity (CAV) has been exceeded. The SJS has at least four triaxial acceleration sensor units and a time-history analyzer and recording system. The time-history analyzer and recording system are located in the auxiliary building.

1. The functional arrangement of the SJS is as described in the Design Description of this Section 2.5.9.
2. The SJS can compute CAV and the 5 percent of critical damping response spectrum for frequencies between 1 and 10 Hertz.
3. The SJS has a dynamic range of 0.001g to 1.0g and a frequency range of 0.2 to 50 Hertz.

Table 2.5.9-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
575	2.5.09.01	1. The functional arrangement of the SJS is as described in the Design Description of this Section 2.5.9.	Inspection of the as-built system will be performed.	The as-built SJS conforms with the functional arrangement as described in the Design Description of this Section 2.5.9.
576	2.5.09.02	2. The SJS can compute CAV and the 5 percent of critical damping response spectrum for frequencies between 1 and 10 Hz.	Type tests using simulated input signals, analyses, or a combination of type tests and analyses, of the SJS time-history analyzer and recording system will be performed.	A report exists and concludes that the SJS time-history analyzer and recording system can record data at a sampling rate of at least 200 samples per second, that the pre-event recording time is adjustable from less than or equal to 1.2 seconds to greater than or equal to 15.0 seconds, and that the initiation value is adjustable from less than or equal to 0.002g to greater than or equal to 0.02g.
577	2.5.09.03	3. The SJS has a dynamic range of 0.001g to 1.0g and a frequency range of 0.2 to 50 Hertz.	Type tests, analyses, or a combination of type tests and analyses, of the SJS triaxial acceleration sensors will be performed.	A report exists and concludes that the SJS triaxial acceleration sensors have a dynamic range of at least 0.001g to 1.0g and a frequency range of at least 0.2 to 50 Hertz.

2.5.10 Main Turbine Control and Diagnostic System

No entry. Covered in Section 2.4.2, Main Turbine System.

C.2.5.11 Meteorological and Environmental Monitoring System

No entry for this system.

C.2.5.12 Closed Circuit TV System

No entry for this system.

2.6 Electrical Power Systems

2.6.1 Main ac Power System

Design Description

The main ac power system (ECS) provides electrical ac power to nonsafety-related loads and non-Class 1E power to the Class 1E battery chargers and regulating transformers during normal and off-normal conditions.

The ECS is as shown in Figures 2.6.1-1 and the component locations of the ECS are as shown in Table 2.6.1-5.

1. The functional arrangement of the ECS is as described in the Design Description of this Section 2.6.1.
2. The seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.
3. a) The Class 1E breaker control power for the equipment identified in Table 2.6.1-1 are powered from their respective Class 1E division.
   b) Separation is provided between ECS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.
4. The ECS provides the following nonsafety-related functions:
   a) The ECS provides the capability for distributing non-Class 1E ac power from onsite sources (ZOS) to nonsafety-related loads listed in Table 2.6.1-2.
   b) The 6900 Vac circuit breakers in switchgear ECS-ES-1 and ECS-ES-2 open after receiving a signal from the onsite standby power system.
   c) Each standby diesel generator 6900 Vac circuit breaker closes after receiving a signal from the onsite standby power system.
   d) Each ancillary diesel generator unit is sized to supply power to long-term safety-related post-accident monitoring loads and control room lighting and ventilation through a regulating transformer; and for one passive containment cooling system (PCS) recirculation pump.
   e) The ECS provides two loss-of-voltage signals to the onsite standby power system (ZOS), one for each diesel-backed 6900 Vac switchgear bus.

f) The ECS provides a reverse-power trip of the generator circuit breaker which is blocked for at least 15 seconds following a turbine trip.

5. Controls exist in the main control room (MCR) to cause the circuit breakers identified in Table 2.6.1-3 to perform the listed functions.

6. Displays of the parameters identified in Table 2.6.1-3 can be retrieved in the MCR.

Table 2.6.1-1

Equipment Name	Tag No.	Seismic Category I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Reactor Coolant Pump (RCP) Circuit Breaker	ECS-ES-31	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-32	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-41	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-42	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-51	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-52	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-61	Yes	Yes/No (Trip open only)	No
RCP Circuit Breaker	ECS-ES-62	Yes	Yes/No (Trip open only)	No

Table 2.6.1-2	
Load Description	**Power Source**
Load Center Transformers EK-11, EK-12, EK-13, EK-14	ZOS-MG-02A
Diesel Oil Transfer Module Enclosure A Electric Unit Heater	ZOS-MG-02A
Diesel Oil Transfer Module Enclosure A Fan	ZOS-MG-02A
Class 1E Division A Regulating Transformer	ZOS-MG-02A
Class 1E Division C Regulating Transformer	ZOS-MG-02A
Diesel Generator Fuel Oil Transfer Pump 1A	ZOS-MG-02A
Diesel Generator Room A Building Standby Exhaust Fans 1A and 2A	ZOS-MG-02A
Diesel Generator Service Module A Air Handling Unit (AHU) 01A Fan	ZOS-MG-02A
Startup Feedwater Pump A	ZOS-MG-02A
Service Water Pump A	ZOS-MG-02A
Service Water Cooling Tower Fan A	ZOS-MG-02A
MCR/Control Support Area (CSA) AHU A Supply and Return Fans	ZOS-MG-02A
Divisions A/C Class 1E Electrical Room AHU A Supply and Return Fans	ZOS-MG-02A
Divisions B/D Class 1E Electrical Room AHU D Supply and Return Fans	ZOS-MG-02A
Air-cooled Chiller Pump 2	ZOS-MG-02A
Component Cooling Water Pump 1A	ZOS-MG-02A
Air-cooled Chiller 2	ZOS-MG-02A
Chemical and Volume Control System (CVS) Makeup Pump 1A	ZOS-MG-02A
CVS Pump Room Unit Cooler Fan A	ZOS-MG-02A
Normal Residual Heat Removal System (RNS) Pump 1A	ZOS-MG-02A
RNS Pump Room Unit Cooler Fan A	ZOS-MG-02A
Equipment Room AHU Supply and Return Fans VXS-MA-01A/02A	ZOS-MG-02A
Switchgear Room A AHU Supply and Return Fans VXS-MA-05A/06A	ZOS-MG-02A
Non-1E Battery Charger EDS1-DC-1	ZOS-MG-02A
Non-1E Battery Room A Exhaust Fan	ZOS-MG-02A
Non-1E Battery Charger EDS3-DC-1	ZOS-MG-02A

Table 2.6.1-2 (cont.)	
Load Description	**Power Source**
Class 1E Division A Battery Charger 1 (24-hour)	ZOS-MG-02A
Class 1E Division C Battery Charger 1 (24-hour)	ZOS-MG-02A
Class 1E Division C Battery Charger 2 (72-hour)	ZOS-MG-02A
Divisions A/C Class 1E Battery Room Exhaust Fan A	ZOS-MG-02A
Supplemental Air Filtration Unit Fan A	ZOS-MG-02A
Backup Group 4A Pressurizer Heaters	ZOS-MG-02A
Spent Fuel Cooling Pump 1A	ZOS-MG-02A
Load Center Transformers EK-21, EK-22, EK-23, EK-24	ZOS-MG-02B
Diesel Oil Transfer Module Enclosure B Electric Unit Heater	ZOS-MG-02B
Diesel Oil Transfer Module Enclosure B Fan	ZOS-MG-02B
Class 1E Division B Regulating Transformer	ZOS-MG-02B
Class 1E Division D Regulating Transformer	ZOS-MG-02B
Diesel Generator Fuel Oil Transfer Pump 1B	ZOS-MG-02B
Diesel Generator Room B Building Standby Exhaust Fans 1B and 2B	ZOS-MG-02B
Diesel Generator Service Module B AHU 01B Fan	ZOS-MG-02B
Startup Feedwater Pump B	ZOS-MG-02B
Service Water Pump B	ZOS-MG-02B
Service Water Cooling Tower Fan B	ZOS-MG-02B
MCR/CSA AHU B Supply and Return Fans	ZOS-MG-02B
Divisions B/D Class 1E Electrical Room AHU B Supply and Return Fans	ZOS-MG-02B
Divisions A/C Class 1E Electrical Room AHU C Supply and Return Fans	ZOS-MG-02B
Air-cooled Chiller Pump 3	ZOS-MG-02B
Component Cooling Water Pump 1B	ZOS-MG-02B
Air-cooled Chiller 3	ZOS-MG-02B
CVS Makeup Pump 1B	ZOS-MG-02B
CVS Pump Room Unit Cooler Fan B	ZOS-MG-02B
RNS Pump 1B	ZOS-MG-02B
RNS Pump Room Unit Cooler Fan B	ZOS-MG-02B
Equipment Room B AHU Supply and Return Fans VXS-MA-01B/02B	ZOS-MG-02B

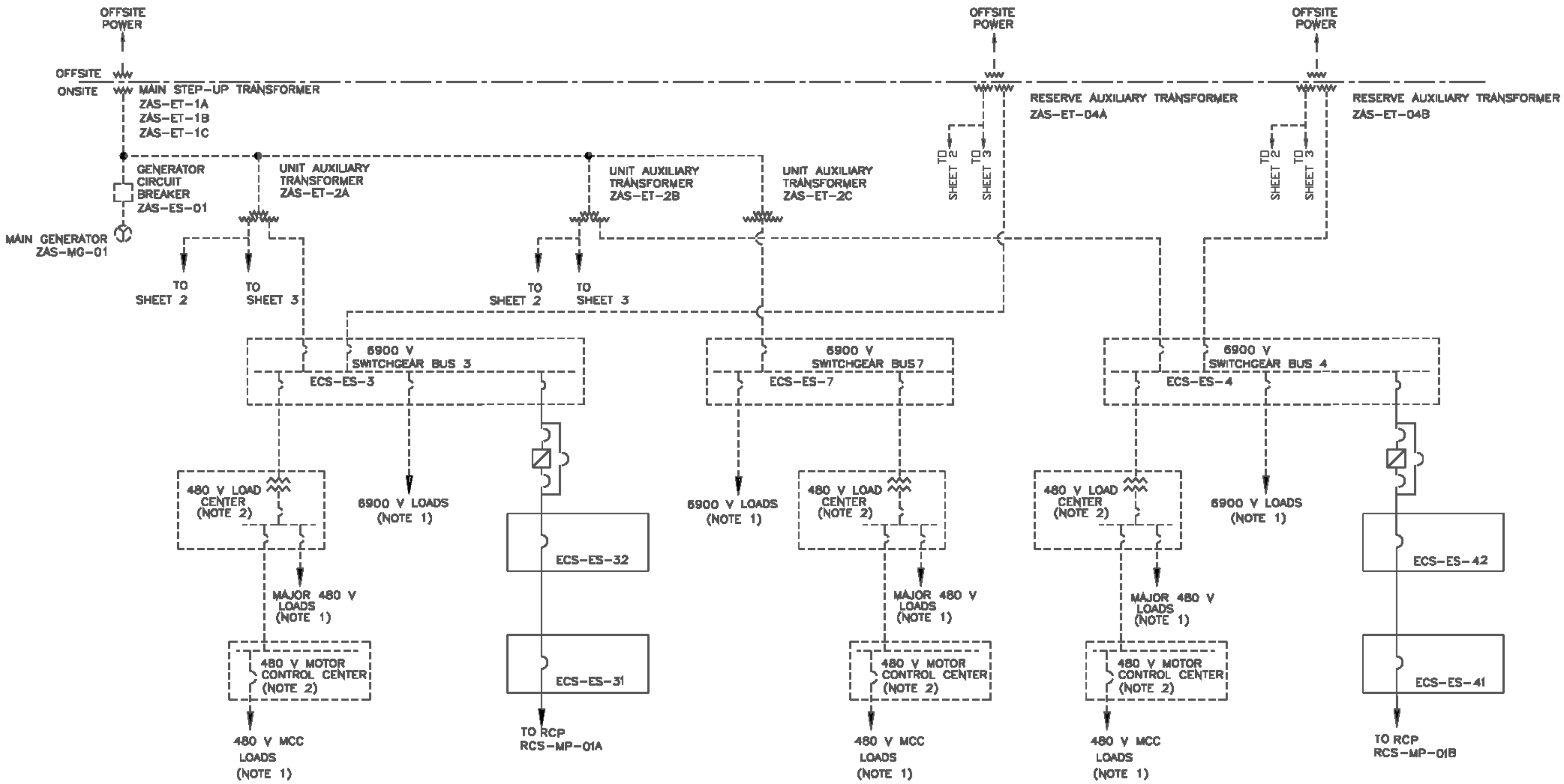
Table 2.6.1-2 (cont.)	
Load Description	**Power Source**
Switchgear Room B AHU Supply and Return Fans VXS-MA-05B/06B	ZOS-MG-02B
Non-1E Battery Charger EDS2-DC-1	ZOS-MG-02B
Non-1E Battery Charger EDS4-DC-1	ZOS-MG-02B
Non-1E Battery Room B Exhaust Fan	ZOS-MG-02B
Class 1E Division B Battery Charger 1 (24-hour)	ZOS-MG-02B
Class 1E Division B Battery Charger 2 (72-hour)	ZOS-MG-02B
Class 1E Division D Battery Charger 1 (24-hour)	ZOS-MG-02B
Divisions B/D Class 1E Battery Room Exhaust Fan B	ZOS-MG-02B
Supplemental Air Filtration Unit Fan B	ZOS-MG-02B
Backup Group 4B Pressurizer Heaters	ZOS-MG-02B
Spent Fuel Cooling Pump 1B	ZOS-MG-02B

Table 2.6.1-3			
Equipment	**Tag No.**	**Display**	**Control Function**
6900 V Switchgear Bus 1	ECS-ES-1	Yes (Bus voltage, breaker position for all breakers on bus)	Yes (Breaker open/close)
6900 V Switchgear Bus 2	ECS-ES-2	Yes (Bus voltage, breaker position for all breakers on bus)	Yes (Breaker open/close)
Unit Auxiliary Transformer A	ZAS-ET-2A	Yes (Secondary Voltage)	No
Unit Auxiliary Transformer B	ZAS-ET-2B	Yes (Secondary Voltage)	No
Reserve Auxiliary Transformer A	ZAS-ET-4A	Yes (Secondary Voltage)	No
Reserve Auxiliary Transformer B	ZAS-ET-4B	Yes (Secondary Voltage)	No

Table 2.6.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
578	2.6.01.01	1. The functional arrangement of the ECS is as described in the Design Description of this Section 2.6.1.	Inspection of the as-built system will be performed.	The as-built ECS conforms with the functional arrangement as described in the Design Description of this Section 2.6.1.
579	2.6.01.02.i	2. The seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.6.1-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.6.1-1 is located on the Nuclear Island.
580	2.6.01.02.ii	2. The seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
581	2.6.01.02.iii	2. The seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
582	2.6.01.03a	3.a) The Class 1E breaker control power for the equipment identified in Table 2.6.1-1 are powered from their respective Class 1E division.	Testing will be performed on the ECS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.6.1-1 when the assigned Class 1E division is provided the test signal.
583	2.6.01.03b	3.b) Separation is provided between ECS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
584	2.6.01.04a	4.a) The ECS provides the capability for distributing non-Class 1E ac power from onsite sources (ZOS) to nonsafety-related loads listed in Table 2.6.1-2.	Tests will be performed using a test signal to confirm that an electrical path exists for each selected load listed in Table 2.6.1-2 from an ECS-ES-1 or ECS-ES-2 bus. Each test may be a single test or a series of over-lapping tests.	A test signal exists at the terminals of each selected load.
585	2.6.01.04b	4.b) The 6900 Vac circuit breakers in switchgear ECS-ES-1 and ECS-ES-2 open after receiving a signal from the onsite standby power load system.	See ITAAC Table 2.6.4-1, item 2.a.	See ITAAC Table 2.6.4-1, item 2.a.

Table 2.6.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
586	2.6.01.04c	4.c) Each standby diesel generator 6900 Vac circuit breaker closes after receiving a signal from the onsite standby power system.	Testing will be performed using real or simulated signals from the standby diesel load system.	Each standby diesel generator 6900 Vac circuit breaker closes after receiving a signal from the standby diesel system.
587	2.6.01.04d	4.d) Each ancillary diesel generator unit is sized to supply power to long-term safety-related post-accident monitoring loads and control room lighting and ventilation through a regulating transformer; and for one PCS recirculation pump.	Each ancillary diesel generator will be operated with fuel supplied from the ancillary diesel generator fuel tank and with a load of 35 kW or greater and a power factor between 0.9 and 1.0 for a time period required to reach engine temperature equilibrium plus 2.5 hours.	Each diesel generator provides power to the load with a generator terminal voltage of 480 ± 10% volts and a frequency of 60 ± 5% Hz.
588	2.6.01.04e	4.e) The ECS provides two loss-of-voltage signals to the onsite standby power system (ZOS), one for each diesel-backed 6900 Vac switchgear bus.	Tests on the as-built ECS system will be conducted by simulating a loss-of-voltage condition on each diesel-backed 6900 Vac switchgear bus.	A loss-of-voltage signal is generated when the loss-of-voltage condition is simulated.
589	2.6.01.04f	4.f) The ECS provides a reverse-power trip of the generator circuit breaker which is blocked for at least 15 seconds following a turbine trip.	Tests on the as-built ECS system will be conducted by simulating a turbine trip signal followed by a simulated reverse-power condition. The generator circuit breaker trip signal will be monitored.	The generator circuit breaker trip signal does not occur until at least 15 seconds after the simulated turbine trip.
590	2.6.01.05	5. Controls exist in the MCR to cause the circuit breakers identified in Table 2.6.1-3 to perform the listed functions.	Tests will be performed to verify that controls in the MCR can operate the circuit breakers identified in Table 2.6.1-3.	Controls in the MCR cause the circuit breakers identified in Table 2.6.1-3 to operate.
591	2.6.01.06	6. Displays of the parameters identified in Table 2.6.1-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.6.1-3 in the MCR.	Displays identified in Table 2.6.1-3 can be retrieved in the MCR.

Table 2.6.1-5		
Component Name	**Tag No.**	**Component Location**
RCP Circuit Breaker	ECS-ES-31	Auxiliary Building
RCP Circuit Breaker	ECS-ES-32	Auxiliary Building
RCP Circuit Breaker	ECS-ES-41	Auxiliary Building
RCP Circuit Breaker	ECS-ES-42	Auxiliary Building
RCP Circuit Breaker	ECS-ES-51	Auxiliary Building
RCP Circuit Breaker	ECS-ES-52	Auxiliary Building
RCP Circuit Breaker	ECS-ES-61	Auxiliary Building
RCP Circuit Breaker	ECS-ES-62	Auxiliary Building
6900 V Switchgear Bus 1	ECS-ES-1	Annex Building
6900 V Switchgear Bus 2	ECS-ES-2	Annex Building
6900 V Switchgear Bus 3	ECS-ES-3	Turbine Building
6900 V Switchgear Bus 4	ECS-ES-4	Turbine Building
6900 V Switchgear Bus 5	ECS-ES-5	Turbine Building
6900 V Switchgear Bus 6	ECS-ES-6	Turbine Building
Main Generator	ZAS-MG-01	Turbine Building
Generator Circuit Breaker	ZAS-ES-01	Turbine Building
Main Step-up Transformer	ZAS-ET-1A	Yard
Main Step-up Transformer	ZAS-ET-1B	Yard
Main Step-up Transformer	ZAS-ET-1C	Yard
Unit Auxiliary Transformer A	ZAS-ET-2A	Yard
Unit Auxiliary Transformer B	ZAS-ET-2B	Yard
Reserve Auxiliary Transformer A	ZAS-ET-4A	Yard
Reserve Auxiliary Transformer B	ZAS-ET-4B	Yard
Ancillary Diesel Generator #1	ECS-MG-01	Annex Building
Ancillary Diesel Generator #2	ECS-MG-02	Annex Building
Ancillary Diesel Generator Distribution Panel 1	ECS-ED-01	Annex Building
Ancillary Diesel Generator Distribution Panel 1	ECS-ED-02	Annex Building



NOTES:
1. All loads are typical of one or more.
2. Load centers and motor control centers are typical of one or more.

Figure 2.6.1-1 (Sheet 1 of 4)
Main ac Power Syste

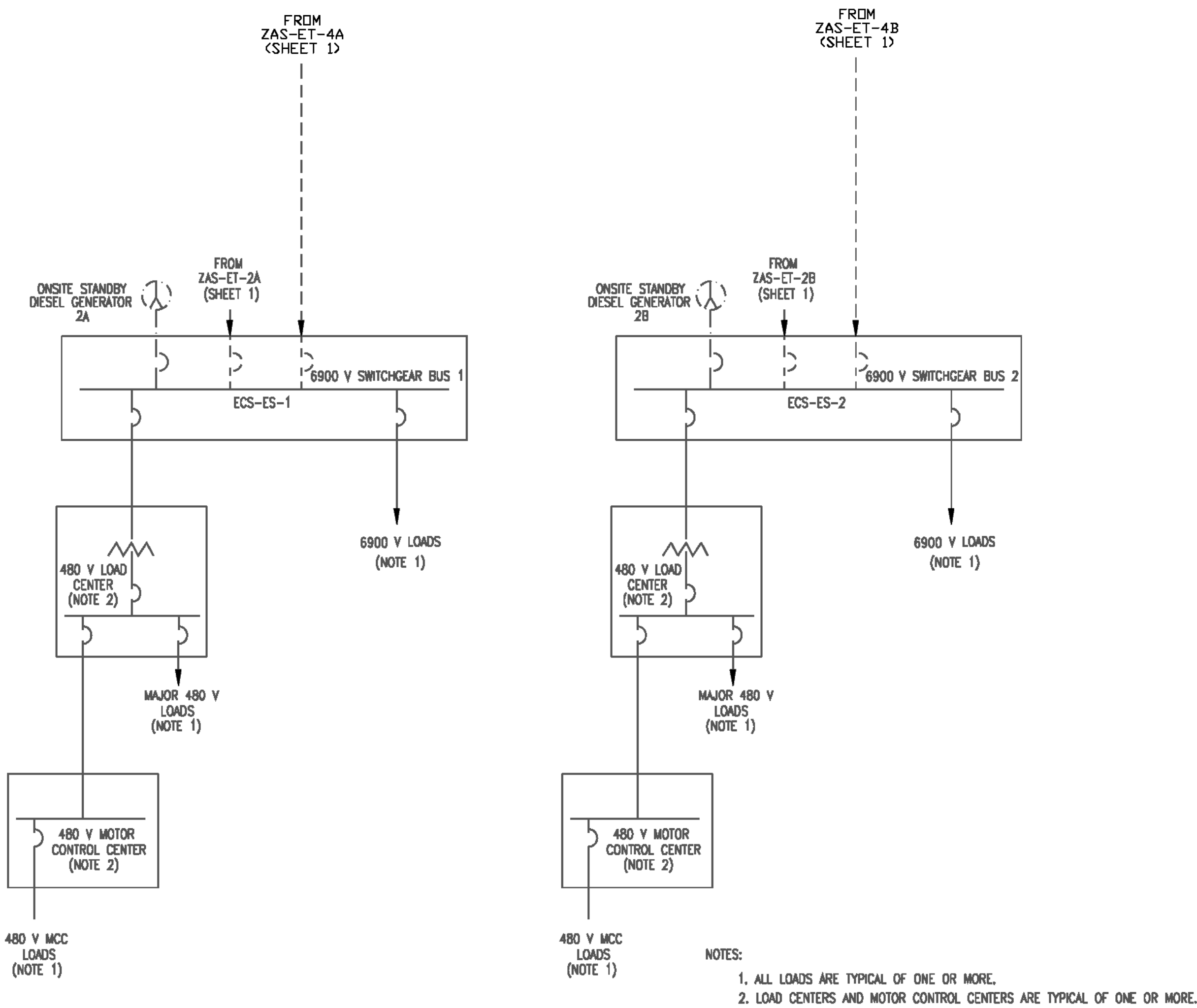


Figure 2.6.1-1 (Sheet 2 of 4)
Main ac Power System

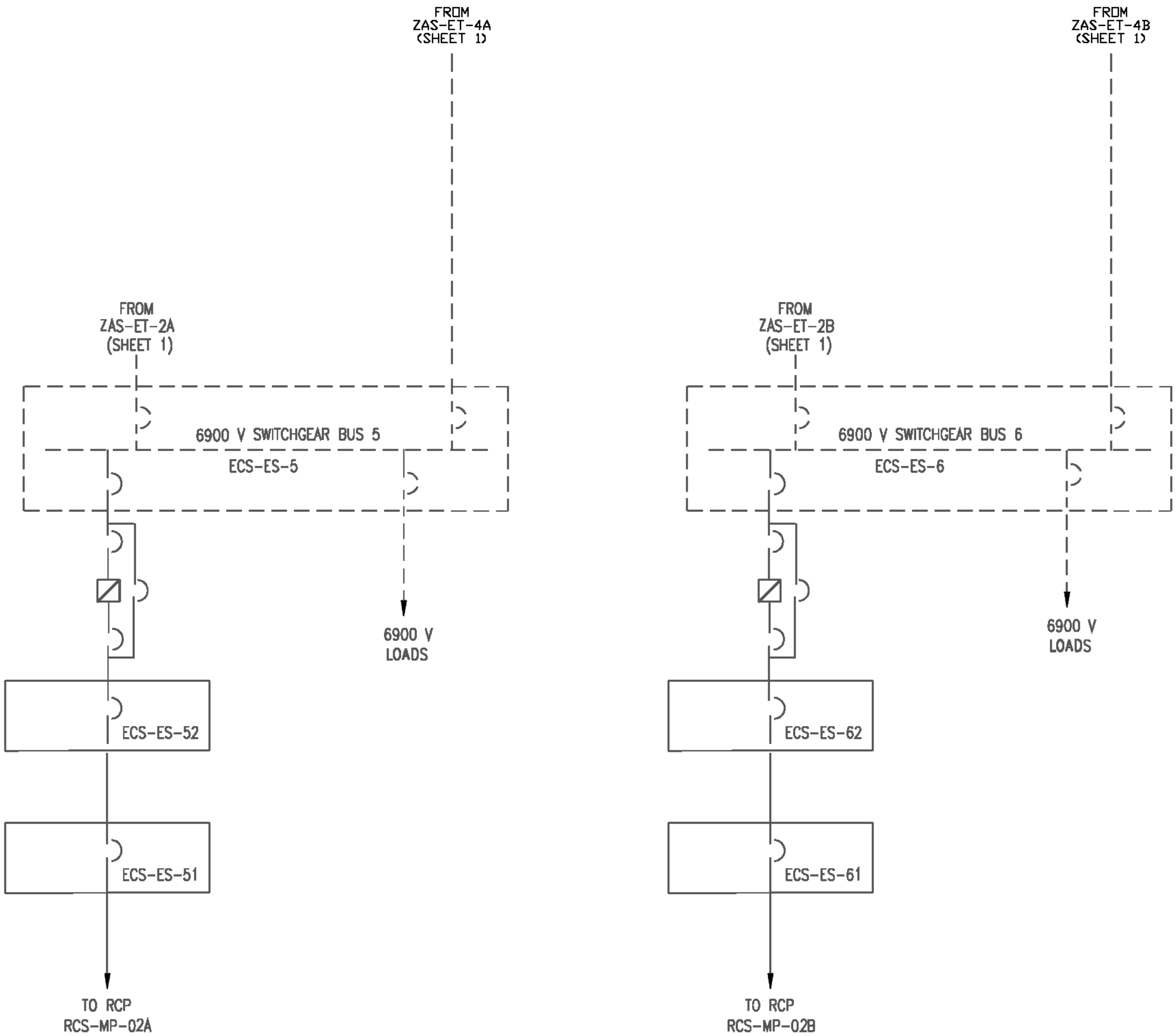


Figure 2.6.1-1 (Sheet 3 of 4)
Main ac Power System

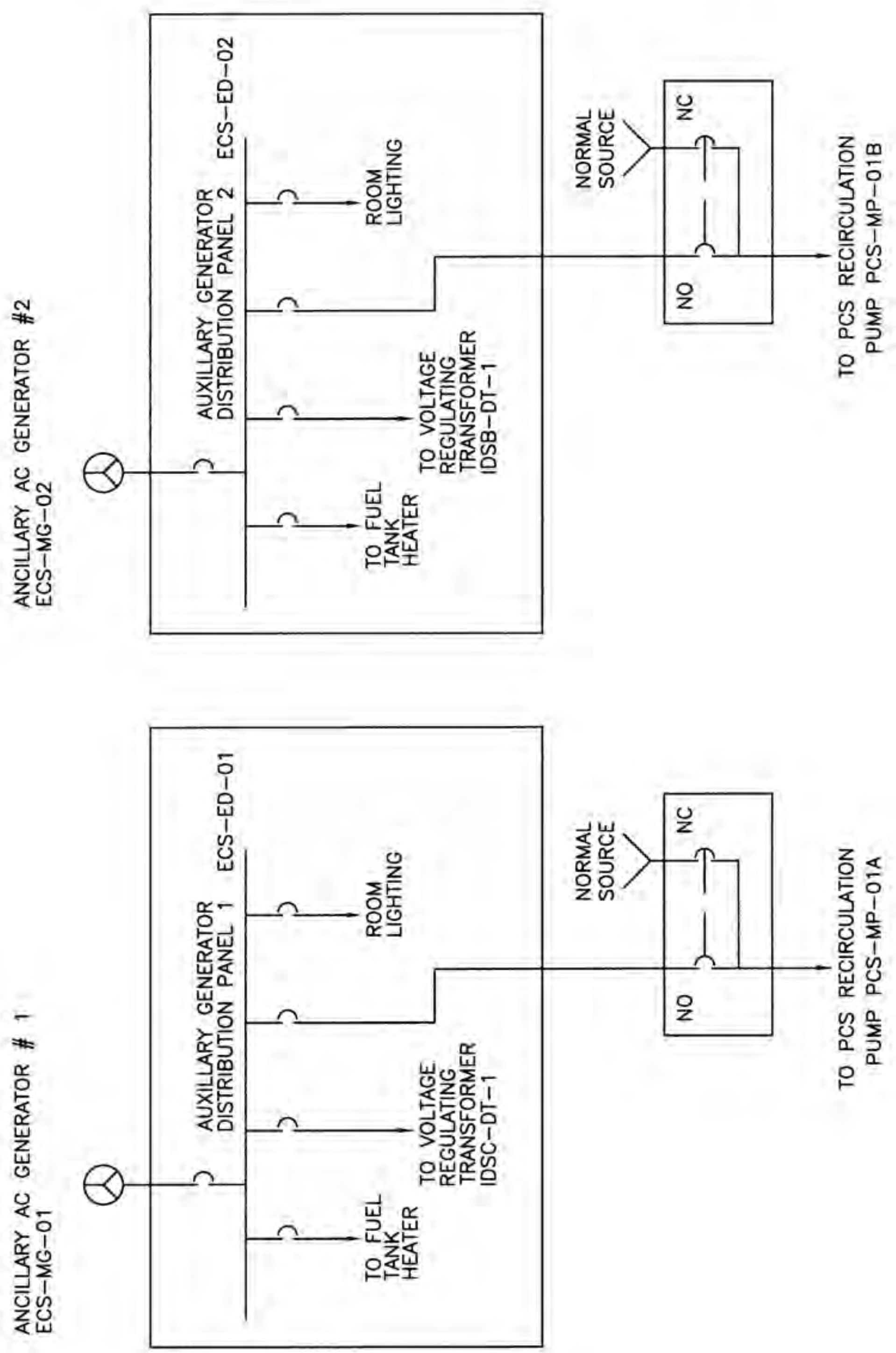


Figure 2.6.1-1 (Sheet 4 of 4)
Main ac Power System

2.6.2 Non-Class 1E dc and Uninterruptible Power Supply System

Design Description

The non-Class 1E dc and uninterruptible power supply system (EDS) provides dc and uninterruptible ac electrical power to nonsafety-related loads during normal and off-normal conditions.

The EDS is as shown in Figure 2.6.2-1 and the component locations of the EDS are as shown in Table 2.6.2-2.

1. The functional arrangement of the EDS is as described in the Design Description of this Section 2.6.2.
2. The EDS provides the following nonsafety-related functions:
   a) Each EDS load group 1, 2, 3, and 4 battery charger supplies the corresponding dc switchboard bus load while maintaining the corresponding battery charged.
   b) Each EDS load group 1, 2, 3, and 4 battery supplies the corresponding dc switchboard bus load for a period of 2 hours without recharging.
   c) Each EDS load group 1, 2, 3, and 4 inverter supplies the corresponding ac load.

Table 2.6.2-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
592	2.6.02.01	1. The functional arrangement of the EDS is as described in the Design Description of this Section 2.6.2.	Inspection of the as-built system will be performed.	The as-built EDS conforms with the functional arrangement as described in the Design Description of this Section 2.6.2.
593	2.6.02.02a	2.a) Each EDS load group 1, 2, 3, and 4 battery charger supplies the corresponding dc switchboard bus load while maintaining the corresponding battery charged.	Testing of each as-built battery charger will be performed by applying a simulated or real load, or a combination of simulated or real loads.	Each battery charger provides an output current of at least 900 amps with an output voltage in the range 105 to 140 V.
594	2.6.02.02b	2.b) Each EDS load group 1, 2, 3, and 4 battery supplies the corresponding dc switchboard bus load for a period of 2 hours without recharging.	Testing of each as-built battery will be performed by applying a simulated or real load, or a combination of simulated or real loads. The test will be conducted on a battery that has been fully charged and has been connected to a battery charger maintained at 135 ± 1 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 105 V after a period of no less than 2 hours, with an equivalent load greater than 850 amps.

Table 2.6.2-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
595	2.6.02.02c	2.c) Each EDS load group 1, 2, 3, and 4 inverter supplies the corresponding ac load.	Testing of each as-built inverter will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 55 kW.	Each inverter provides a line-to-line output voltage of 208 ± 2% V at a frequency of 60 ± 0.5% Hz.

Table 2.6.2-2		
Component Name	**Tag No.**	**Component Location**
Load Group 1 Battery	EDS1-DB-1	Annex Building
Load Group 2 Battery	EDS2-DB-1	Annex Building
Load Group 3 Battery	EDS3-DB-1	Annex Building
Load Group 4 Battery	EDS4-DB-1	Annex Building
Load Group 1 Battery Charger	EDS1-DC-1	Annex Building
Load Group 2 Battery Charger	EDS2-DC-1	Annex Building
Load Group 3 Battery Charger	EDS3-DC-1	Annex Building
Load Group 4 Battery Charger	EDS4-DC-1	Annex Building
Load Group 1 125 Vdc Switchboard	EDS1-DS-1	Annex Building
Load Group 2 125 Vdc Switchboard	EDS2-DS-1	Annex Building
Load Group 3 125 Vdc Switchboard	EDS3-DS-1	Annex Building
Load Group 4 125 Vdc Switchboard	EDS4-DS-1	Annex Building
Load Group 1 Inverter	EDS1-DU-1	Annex Building
Load Group 2 Inverter	EDS2-DU-1	Annex Building
Load Group 3 Inverter	EDS3-DU-1	Annex Building
Load Group 4 Inverter	EDS4-DU-1	Annex Building

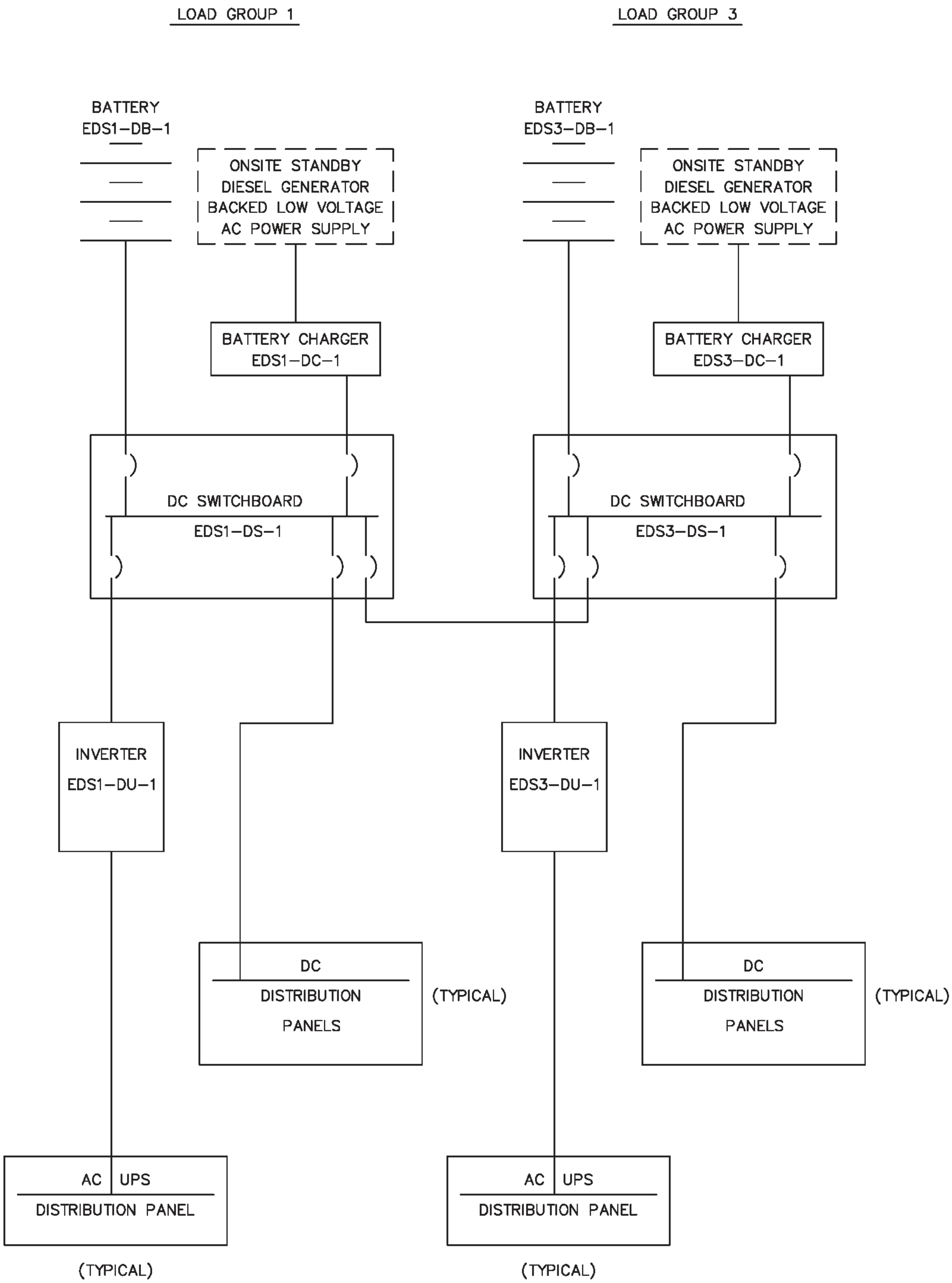


Figure 2.6.2-1 (Sheet 1 of 2)
Non-Class 1E dc and Uninterruptible Power Supply System

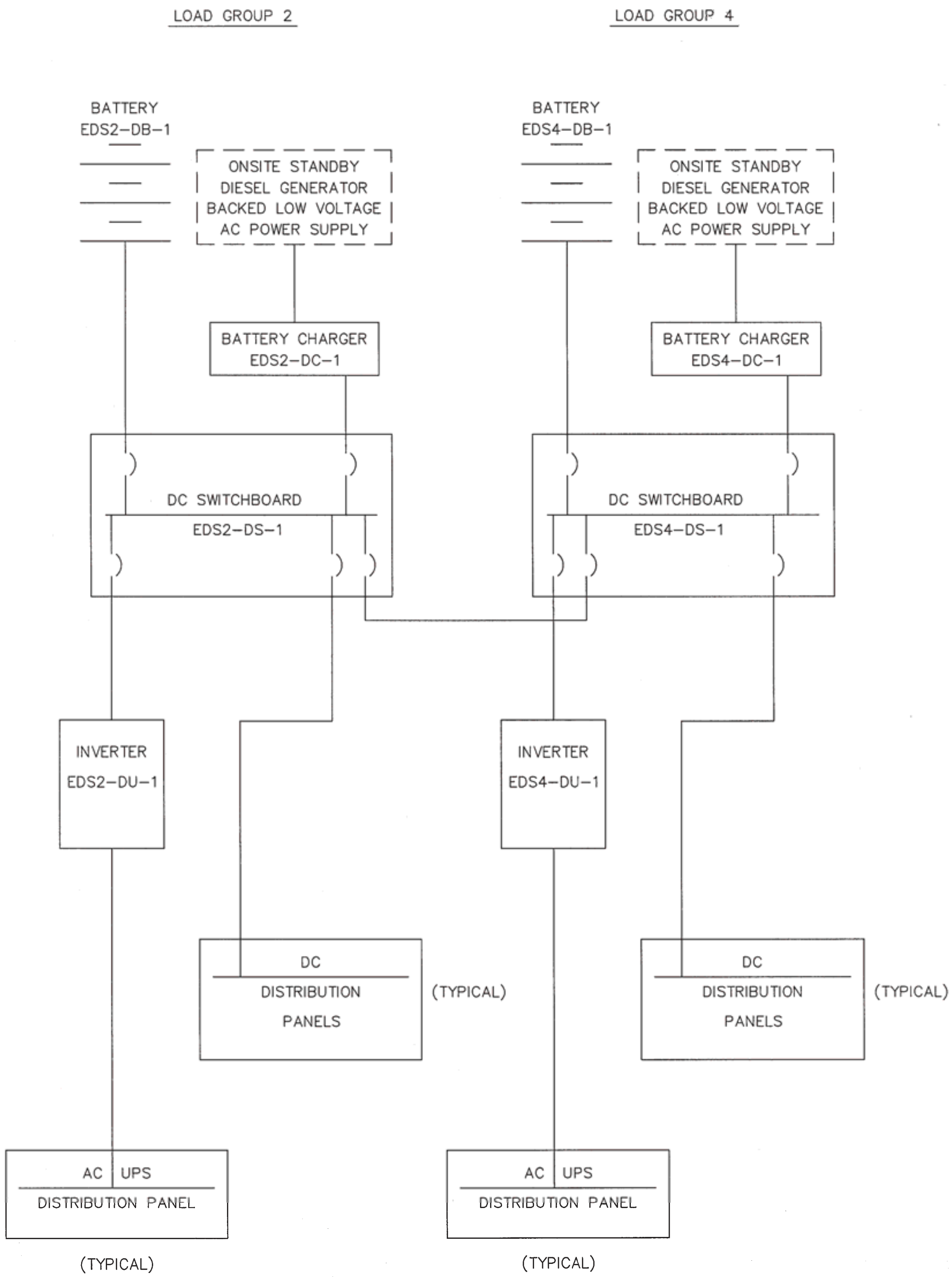


Figure 2.6.2-1 (Sheet 2 of 2)
Non-Class 1E dc and Uninterruptible Power Supply System

2.6.3 Class 1E dc and Uninterruptible Power Supply System

Design Description

The Class 1E dc and uninterruptible power supply system (IDS) provides dc and uninterruptible ac electrical power for safety-related equipment during normal and off-normal conditions.

The IDS is as shown in Figure 2.6.3-1 and the component locations of the IDS are as shown in Table 2.6.3-4.

1. The functional arrangement of the IDS is as described in the Design Description of this Section 2.6.3.
2. The seismic Category I equipment identified in Table 2.6.3-1 can withstand seismic design basis loads without loss of safety function.
3. Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.
4. The IDS provides the following safety-related functions:
   a) The IDS provides electrical independence between the Class 1E divisions.
   b) The IDS provides electrical isolation between the non-Class 1E ac power system and the non-Class 1E lighting in the MCR.
   c) Each IDS 24-hour battery bank supplies a dc switchboard bus load for a period of 24 hours without recharging.
   d) Each IDS 72-hour battery bank supplies a dc switchboard bus load for a period of 72 hours without recharging.
   e) The IDS spare battery bank supplies a dc load equal to or greater than the most severe switchboard bus load for the required period without recharging.
   f) Each IDS 24-hour inverter supplies its ac load.
   g) Each IDS 72-hour inverter supplies its ac load.
   h) Each IDS 24-hour battery charger provides the protection and safety monitoring system (PMS) with two loss-of-ac input voltage signals.
   i) The IDS supplies an operating voltage at the terminals of the Class 1E motor-operated valves identified in subsections 2.1.2, 2.2.1, 2.2.2, 2.2.3, 2.2.4, 2.3.2, 2.3.6, and 2.7.1 that is greater than or equal to the minimum specified voltage.
   j) The IDS provides electrical isolation between the non-Class 1E battery monitors and the Class 1E battery banks.
5. The IDS provides the following nonsafety-related functions:
   a) Each IDS 24-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.
   b) Each IDS 72-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.
   c) Each IDS regulating transformer supplies an ac load when powered from the 480 V motor control center (MCC).

d) The IDS Divisions B and C regulating transformers supply their post-72 hour ac loads when powered from an ancillary diesel generator.

6. Safety-related displays identified in Table 2.6.3-1 can be retrieved in the MCR.
7. The IDS dc battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, are sized to supply their load requirements.
8. Circuit breakers and fuses in IDS battery, battery charger, dc distribution panel, and MCC circuits are rated to interrupt fault currents.
9. The IDS batteries, battery chargers, dc distribution panels, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.
10. The IDS electrical distribution system cables are rated to withstand fault currents for the time required to clear the fault from its power source.
11. Displays of the parameters identified in Table 2.6.3-2 can be retrieved in the MCR.

Table 2.6.3-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Division A 250 Vdc 24-Hour Battery Bank	IDSA-DB-1	Yes	Yes/No	No
Division B 250 Vdc 24-Hour Battery Bank 1	IDSB-DB-1	Yes	Yes/No	No
Division B 250 Vdc 72-Hour Battery Bank 2	IDSB-DB-2	Yes	Yes/No	No
Division C 250 Vdc 24-Hour Battery Bank 1	IDSC-DB-1	Yes	Yes/No	No
Division C 250 Vdc 72-Hour Battery Bank 2	IDSC-DB-2	Yes	Yes/No	No
Division D 250 Vdc 24-Hour Battery Bank	IDSD-DB-1	Yes	Yes/No	No
Spare 250 Vdc Battery Bank	IDSS-DB-1	Yes	Yes/No	No
Division A 24-Hour Battery Charger 1	IDSA-DC-1	Yes	Yes/No	No
Division B 24-Hour Battery Charger 1	IDSB-DC-1	Yes	Yes/No	No
Division B 72-Hour Battery Charger 2	IDSB-DC-2	Yes	Yes/No	No
Division C 24-Hour Battery Charger 1	IDSC-DC-1	Yes	Yes/No	No
Division C 72-Hour Battery Charger 2	IDSC-DC-2	Yes	Yes/No	No
Division D 24-Hour Battery Charger 1	IDSD-DC-1	Yes	Yes/No	No
Spare Battery Charger 1	IDSS-DC-1	Yes	Yes/No	No
Division A 250 Vdc Distribution Panel	IDSA-DD-1	Yes	Yes/No	No
Division B 250 Vdc Distribution Panel	IDSB-DD-1	Yes	Yes/No	No
Division C 250 Vdc Distribution Panel	IDSC-DD-1	Yes	Yes/No	No

Table 2.6.3-1				
Equipment Name	**Tag No.**	**Seismic Cat. I**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**
Division D 250 Vdc Distribution Panel	IDSD-DD-1	Yes	Yes/No	No
Division A 120 Vac Distribution Panel 1	IDSA-EA-1	Yes	Yes/No	No
Division A 120 Vac Distribution Panel 2	IDSA-EA-2	Yes	Yes/No	No
Division B 120 Vac Distribution Panel 1	IDSB-EA-1	Yes	Yes/No	No
Division B 120 Vac Distribution Panel 2	IDSB-EA-2	Yes	Yes/No	No
Division B 120 Vac Distribution Panel 3	IDSB-EA-3	Yes	Yes/No	No
Division C 120 Vac Distribution Panel 1	IDSC-EA-1	Yes	Yes/No	No
Division C 120 Vac Distribution Panel 2	IDSC-EA-2	Yes	Yes/No	No
Division C 120 Vac Distribution Panel 3	IDSC-EA-3	Yes	Yes/No	No
Division D 120 Vac Distribution Panel 1	IDSD-EA-1	Yes	Yes/No	No
Division D 120 Vac Distribution Panel 2	IDSD-EA-2	Yes	Yes/No	No
Division A Fuse Panel 4	IDSA-EA-4	Yes	Yes/No	No
IDSA Battery Monitor Fuse Panel	IDSA-EA-5	Yes	Yes/No	No
Division B Fuse Panel 4	IDSB-EA-4	Yes	Yes/No	No
Division B Fuse Panel 5	IDSB-EA-5	Yes	Yes/No	No
Division B Fuse Panel 6	IDSB-EA-6	Yes	Yes/No	No
IDSB Battery Monitor Fuse Panel	IDSB-EA-7	Yes	Yes/No	No
IDSB Battery Monitor Fuse Panel	IDSB-EA-8	Yes	Yes/No	No
Division C Fuse Panel 4	IDSC-EA-4	Yes	Yes/No	No
Division C Fuse Panel 5	IDSC-EA-5	Yes	Yes/No	No
Division C Fuse Panel 6	IDSC-EA-6	Yes	Yes/No	No
IDSC Battery Monitor Fuse Panel	IDSC-EA-7	Yes	Yes/No	No
IDSC Battery Monitor Fuse Panel	IDSC-EA-8	Yes	Yes/No	No
Division D Fuse Panel 4	IDSD-EA-4	Yes	Yes/No	No
IDSD Battery Monitor Fuse Panel	IDSD-EA-5	Yes	Yes/No	No
IDSS Battery Monitor Fuse Panel	IDSS-EA-1	Yes	Yes/No	No
Division A Fused Transfer Switch Box 1	IDSA-DF-1	Yes	Yes/No	No
Division B Fused Transfer Switch Box 1	IDSB-DF-1	Yes	Yes/No	No
Division B Fused Transfer Switch Box 2	IDSB-DF-2	Yes	Yes/No	No
Division C Fused Transfer Switch Box 1	IDSC-DF-1	Yes	Yes/No	No
Division C Fused Transfer Switch Box 2	IDSC-DF-2	Yes	Yes/No	No
Division D Fused Transfer Switch Box 1	IDSD-DF-1	Yes	Yes/No	No
Spare Fused Transfer Switch Box 1	IDSS-DF-1	Yes	Yes/No	No
Division A 250 Vdc MCC	IDSA-DK-1	Yes	Yes/No	No
Division B 250 Vdc MCC	IDSB-DK-1	Yes	Yes/No	No

Table 2.6.3-1				
Equipment Name	**Tag No.**	**Seismic Cat. I**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**
Division C 250 Vdc MCC	IDSC-DK-1	Yes	Yes/No	No
Division D 250 Vdc MCC	IDSD-DK-1	Yes	Yes/No	No
Division A 250 Vdc Switchboard 1	IDSA-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division B 250 Vdc Switchboard 1	IDSB-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division B 250 Vdc Switchboard 2	IDSB-DS-2	Yes	Yes/No	Yes (Bus Voltage)
Division C 250 Vdc Switchboard 1	IDSC-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division C 250 Vdc Switchboard 2	IDSC-DS-2	Yes	Yes/No	Yes (Bus Voltage)
Division D 250 Vdc Switchboard 1	IDSD-DS-1	Yes	Yes/No	Yes (Bus Voltage)
Division A Regulating Transformer	IDSA-DT-1	Yes	Yes/No	No
Division B Regulating Transformer	IDSB-DT-1	Yes	Yes/No	No
Division C Regulating Transformer	IDSC-DT-1	Yes	Yes/No	No
Division D Regulating Transformer	IDSD-DT-1	Yes	Yes/No	No
Division A 24-Hour Inverter 1	IDSA-DU-1	Yes	Yes/No	No
Division B 24-Hour Inverter 1	IDSB-DU-1	Yes	Yes/No	No
Division B 72-Hour Inverter 2	IDSB-DU-2	Yes	Yes/No	No
Division C 24-Hour Inverter 1	IDSC-DU-1	Yes	Yes/No	No
Division C 72-Hour Inverter 2	IDSC-DU-2	Yes	Yes/No	No
Division D 24-Hour Inverter 1	IDSD-DU-1	Yes	Yes/No	No
Spare Battery Termination Box	IDSS-DF-3	Yes	Yes/No	No

Table 2.6.3-2		
Equipment	**Tag No.**	**Display/Status Indication**
Division A Battery Monitor	IDSA-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division B 24-Hour Battery Monitor	IDSB-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division B 72-Hour Battery Monitor	IDSB-DV-2	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division C 24-Hour Battery Monitor	IDSC-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division C 72-Hour Battery Monitor	IDSC-DV-2	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division D Battery Monitor	IDSD-DV-1	Yes (Battery Ground Detection, Battery High Discharge Rate)
Division A Fused Transfer Switch Box	IDSA-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division B 24-Hour Fused Transfer Switch Box	IDSB-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division B 72-Hour Fused Transfer Switch Box	IDSB-DF-2	Yes (Battery Current, Battery Disconnect Switch Position)
Division C 24-Hour Fused Transfer Switch Box	IDSC-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division C 72-Hour Fused Transfer Switch Box	IDSC-DF-2	Yes (Battery Current, Battery Disconnect Switch Position)
Division D Fused Transfer Switch Box	IDSD-DF-1	Yes (Battery Current, Battery Disconnect Switch Position)
Division A Battery Charger	IDSA-DC-1	Yes (Charger Output Current, Charger Trouble(1))

Table 2.6.3-2		
Equipment	**Tag No.**	**Display/Status Indication**
Division B 24-Hour Battery Charger	IDSB-DC-1	Yes (Charger Output Current, Charger Trouble(1))
Division B 72-Hour Battery Charger	IDSB-DC-2	Yes (Charger Output Current, Charger Trouble(1))
Division C 24-Hour Battery Charger	IDSC-DC-1	Yes (Charger Output Current, Charger Trouble(1))
Division C 72-Hour Battery Charger	IDSC-DC-2	Yes (Charger Output Current, Charger Trouble(1))
Division D Battery Charger	IDSD-DC-1	Yes (Charger Output Current, Charger Trouble(1))

Note: (1) Battery charger trouble includes charger dc output under/over voltage

Table 2.6.3-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
596	2.6.03.01	1. The functional arrangement of the IDS is as described in the Design Description of this Section 2.6.3.	Inspection of the as-built system will be performed.	The as-built IDS conforms with the functional arrangement as described in the Design Description of this Section 2.6.3.
597	2.6.03.02.i	2. The seismic Category I equipment identified in Table 2.6.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.6.3-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.6.3-1 is located on the Nuclear Island.
598	2.6.03.02.ii	2. The seismic Category I equipment identified in Table 2.6.3-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.

Table 2.6.3-3
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
599	2.6.03.02.iii	2. The seismic Category I equipment identified in Table 2.6.3-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
600	2.6.03.03	3. Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
601	2.6.03.04a	4.a) The IDS provides electrical independence between the Class 1E divisions.	Testing will be performed on the IDS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.6.3-1 when the assigned Class 1E division is provided the test signal.
602	2.6.03.04b	4.b) The IDS provides electrical isolation between the non-Class 1E ac power system and the non-Class 1E lighting in the MCR.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the battery chargers, regulating transformers, and isolation fuses prevent credible faults from propagating into the IDS.
603	2.6.03.04c	4.c) Each IDS 24-hour battery bank supplies a dc switchboard bus load for a period of 24 hours without recharging.	Testing of each 24-hour as-built battery bank will be performed by applying a simulated or real load, or a combination of simulated or real loads which envelope the battery bank design duty cycle. The test will be conducted on a battery bank that has been fully charged and has been connected to a battery charger maintained at 270±2 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 210 V after a period of no less than 24 hours with an equivalent load that equals or exceeds the battery bank design duty cycle capacity.

Table 2.6.3-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
604	2.6.03.04d	4.d) Each IDS 72-hour battery bank supplies a dc switchboard bus load for a period of 72 hours without recharging.	Testing of each 72-hour as-built battery bank will be performed by applying a simulated or real load, or a combination of simulated or real loads which envelope the battery bank design duty cycle. The test will be conducted on a battery bank that has been fully charged and has been connected to a battery charger maintained at 270±2 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 210 V after a period of no less than 72 hours with an equivalent load that equals or exceeds the battery bank design duty cycle capacity.
605	2.6.03.04e	4.e) The IDS spare battery bank supplies a dc load equal to or greater than the most severe switchboard bus load for the required period without recharging.	Testing of the as-built spare battery bank will be performed by applying a simulated or real load, or a combination of simulated or real loads which envelope the most severe of the division batteries design duty cycle. The test will be conducted on a battery bank that has been fully charged and has been connected to a battery charger maintained at 270±2 V for a period of no less than 24 hours prior to the test.	The battery terminal voltage is greater than or equal to 210 V after a period with a load and duration that equals or exceeds the most severe battery bank design duty cycle capacity.
606	2.6.03.04f	4.f) Each IDS 24-hour inverter supplies its ac load.	Testing of each 24-hour as-built inverter will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 12 kW. The inverter input voltage will be no more than 210 Vdc during the test.	Each 24-hour inverter supplies a line-to-line output voltage of 208 ± 2% V at a frequency of 60 ± 0.5% Hz.
607	2.6.03.04g	4.g) Each IDS 72-hour inverter supplies its ac load.	Testing of each 72-hour as-built inverter will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 7 kW. The inverter input voltage will be no more than 210 Vdc during the test.	Each 72-hour inverter supplies a line-to-line output voltage of 208 ± 2% V at a frequency of 60 ± 0.5% Hz.

Table 2.6.3-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
608	2.6.03.04h	4.h) Each IDS 24-hour battery charger provides the PMS with two loss-of-ac input voltage signals.	Testing will be performed by simulating a loss of input voltage to each 24-hour battery charger.	Two PMS input signals exist from each 24-hour battery charger indicating loss of ac input voltage when the loss-of-input voltage condition is simulated.
609	2.6.03.04i	4.i) The IDS supplies an operating voltage at the terminals of the Class 1E motor operated valves identified in subsections 2.1.2, 2.2.1, 2.2.2, 2.2.3, 2.2.4, 2.3.2, 2.3.6, and 2.7.1 that is greater than or equal to the minimum specified voltage.	Testing will be performed by stroking each specified motor-operated valve and measuring the terminal voltage at the motor starter input terminals with the motor operating. The battery terminal voltage will be no more than 210 Vdc during the test.	The motor starter input terminal voltage is greater than or equal 200 Vdc with the motor operating.
875	2.6.03.04j	4.j) The IDS provides electrical isolation between the non-Class 1E battery monitors and the Class 1E battery banks.	Type tests, analyses, or a combination of type tests and analyses of the isolation devices will be performed.	A report exists and concludes that the battery monitor fuse isolation panels prevent credible faults from propagating into the Class 1E portions of the IDS.
610	2.6.03.05a	5.a) Each IDS 24-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.	Testing of each as-built 24-hour battery charger will be performed by applying a simulated or real load, or a combination of simulated or real loads.	Each 24-hour battery charger provides an output current of at least 150 A with an output voltage in the range 210 to 280 V.
611	2.6.03.05b	5.b) Each IDS 72-hour battery charger supplies a dc switchboard bus load while maintaining the corresponding battery charged.	Testing of each 72-hour as-built battery charger will be performed by applying a simulated or real load, or a combination of simulated or real loads.	Each 72-hour battery charger provides an output current of at least 125 A with an output voltage in the range 210 to 280 V.
612	2.6.03.05c	5.c) Each IDS regulating transformer supplies an ac load when powered from the 480 V MCC.	Testing of each as-built regulating transformer will be performed by applying a simulated or real load, or a combination of simulated or real loads, equivalent to a resistive load greater than 30 kW when powered from the 480 V MCC.	Each regulating transformer supplies a line-to-line output voltage of 208 ± 2% V.
613	2.6.03.05d.i	5.d) The IDS Divisions B and C regulating transformers supply their post-72-hour ac loads when powered from an ancillary diesel generator.	Inspection of the as-built system will be performed.	i) Ancillary diesel generator 1 is electrically connected to regulating transformer IDSC-DT-1
614	2.6.03.05d.ii	5.d) The IDS Divisions B and C regulating transformers supply their post-72-hour ac loads when powered from an ancillary diesel generator.	Inspection of the as-built system will be performed.	ii) Ancillary diesel generator 2 is electrically connected to regulating transformer IDSB-DT-1.

Table 2.6.3-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
615	2.6.03.06	6. Safety-related displays identified in Table 2.6.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.6.3-1 can be retrieved in the MCR.
616	2.6.03.07	7. The IDS dc battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, are sized to supply their load requirements.	Analyses for the as-built IDS dc electrical distribution system to determine the capacities of the battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the capacities of as-built IDS battery fuses and battery charger circuit breakers, and dc distribution panels, MCCs, and their circuit breakers and fuses, as determined by their nameplate ratings, exceed their analyzed load requirements.
617	2.6.03.08	8. Circuit breakers and fuses in IDS battery, battery charger, dc distribution panel, and MCC circuits are rated to interrupt fault currents.	Analyses for the as-built IDS dc electrical distribution system to determine fault currents will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the analyzed fault currents do not exceed the interrupt capacity of circuit breakers and fuses in the battery, battery charger, dc distribution panel, and MCC circuits, as determined by their nameplate ratings.
618	2.6.03.09	9. The IDS batteries, battery chargers, dc distribution panels, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.	Analyses for the as-built IDS dc electrical distribution system to determine fault currents will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the fault current capacities of as-built IDS batteries, battery chargers, dc distribution panels, and MCCs, as determined by manufacturer's ratings, exceed their analyzed fault currents for the time required to clear the fault from its power source as determined by the circuit interrupting device coordination analyses.

Table 2.6.3-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
619	2.6.03.10	10. The IDS electrical distribution system cables are rated to withstand fault currents for the time required to clear the fault from its power source.	Analyses for the as-built IDS dc electrical distribution system to determine fault currents will be performed.	Analyses for the as-built IDS dc electrical distribution system exist and conclude that the IDS dc electrical distribution system cables will withstand the analyzed fault currents, as determined by manufacturer's ratings, for the time required to clear the fault from its power source as determined by the circuit interrupting device coordination analyses.
620	2.6.03.11	11. Displays of the parameters identified in Table 2.6.3-2 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.6.3-2 in the MCR.	Displays identified in Table 2.6.3-2 can be retrieved in the MCR.

Table 2.6.3-4		
Component Name	**Tag No.**	**Component Location**
Division A 250 Vdc 24-Hour Battery Bank	IDSA-DB-1	Auxiliary Building
Division B 250 Vdc 24-Hour Battery Bank 1	IDSB-DB-2	Auxiliary Building
Division B 250 Vdc 72-Hour Battery Bank 2	IDSB-DB-2	Auxiliary Building
Division C 250 Vdc 24-Hour Battery Bank 1	IDSC-DB-1	Auxiliary Building
Division C 250 Vdc 72-Hour Battery Bank 2	IDSC-DB-2	Auxiliary Building
Division D 250 Vdc 24-Hour Battery Bank	IDSD-DB-1	Auxiliary Building
Spare 250 Vdc Battery Bank	IDSS-DB-1	Auxiliary Building
Division A 24-Hour Battery Charger 1	IDSA-DC-1	Auxiliary Building
Division B 24-Hour Battery Charger 1	IDSB-DC-1	Auxiliary Building
Division B 72-Hour Battery Charger 2	IDSB-DC-2	Auxiliary Building
Division C 24-Hour Battery Charger 1	IDSC-DC-1	Auxiliary Building
Division C 72-Hour Battery Charger 2	IDSC-DC-2	Auxiliary Building
Division D 24-Hour Battery Charger 1	IDSD-DC-1	Auxiliary Building
Spare Battery Charger 1	IDSS-DC-1	Auxiliary Building
Division A 250 Vdc Distribution Panel	IDSA-DD-1	Auxiliary Building

Table 2.6.3-4		
Component Name	**Tag No.**	**Component Location**
Division B 250 Vdc Distribution Panel	IDSB-DD-1	Auxiliary Building
Division C 250 Vdc Distribution Panel	IDSC-DD-1	Auxiliary Building
Division D 250 Vdc Distribution Panel	IDSD-DD-1	Auxiliary Building
Division A 120 Vac Distribution Panel 1	IDSA-EA-1	Auxiliary Building
Division A 120 Vac Distribution Panel 2	IDSA-EA-2	Auxiliary Building
Division B 120 Vac Distribution Panel 1	IDSB-EA-1	Auxiliary Building
Division B 120 Vac Distribution Panel 2	IDSB-EA-2	Auxiliary Building
Division B 120 Vac Distribution Panel 3	IDSB-EA-3	Auxiliary Building
Division C 120 Vac Distribution Panel 1	IDSC-EA-1	Auxiliary Building
Division C 120 Vac Distribution Panel 2	IDSC-EA-2	Auxiliary Building
Division C 120 Vac Distribution Panel 3	IDSC-EA-3	Auxiliary Building
Division D 120 Vac Distribution Panel 1	IDSD-EA-1	Auxiliary Building
Division D 120 Vac Distribution Panel 2	IDSD-EA-2	Auxiliary Building
Division A Fuse Panel 4	IDSA-EA-4	Auxiliary Building
IDSA Battery Monitor Fuse Panel	IDSA-EA-5	Auxiliary Building
Division B Fuse Panel 4	IDSB-EA-4	Auxiliary Building
Division B Fuse Panel 5	IDSB-EA-5	Auxiliary Building
Division B Fuse Panel 6	IDSB-EA-6	Auxiliary Building
IDSB Battery Monitor Fuse Panel	IDSB-EA-7	Auxiliary Building
IDSB Battery Monitor Fuse Panel	IDSB-EA-8	Auxiliary Building
Division C Fuse Panel 4	IDSC-EA-4	Auxiliary Building
Division C Fuse Panel 5	IDSC-EA-5	Auxiliary Building
Division C Fuse Panel 6	IDSC-EA-6	Auxiliary Building
IDSC Battery Monitor Fuse Panel	IDSC-EA-7	Auxiliary Building
IDSC Battery Monitor Fuse Panel	IDSC-EA-8	Auxiliary Building
Division D Fuse Panel 4	IDSD-EA-4	Auxiliary Building
IDSD Battery Monitor Fuse Panel	IDSD-EA-5	Auxiliary Building
IDSS Battery Monitor Fuse Panel	IDSS-EA-1	Auxiliary Building
Division A Fused Transfer Switch Box 1	IDSA-DF-1	Auxiliary Building
Division B Fused Transfer Switch Box 1	IDSB-DF-1	Auxiliary Building
Division B Fused Transfer Switch Box 2	IDSB-DF-2	Auxiliary Building
Division C Fused Transfer Switch Box 1	IDSC-DF-1	Auxiliary Building
Division C Fused Transfer Switch Box 2	IDSC-DF-2	Auxiliary Building
Division D Fused Transfer Switch Box 1	IDSD-DF-1	Auxiliary Building
Spare Fused Transfer Switch Box 1	IDSS-DF-1	Auxiliary Building
Division A 250 Vdc MCC	IDSA-DK-1	Auxiliary Building

Table 2.6.3-4		
Component Name	**Tag No.**	**Component Location**
Division B 250 Vdc MCC	IDSB-DK-1	Auxiliary Building
Division C 250 Vdc MCC	IDSC-DK-1	Auxiliary Building
Division D 250 Vdc MCC	IDSD-DK-1	Auxiliary Building
Division A 250 Vdc Switchboard 1	IDSA-DS-1	Auxiliary Building
Division B 250 Vdc Switchboard 1	IDSB-DS-1	Auxiliary Building
Division B 250 Vdc Switchboard 2	IDSB-DS-2	Auxiliary Building
Division C 250 Vdc Switchboard 1	IDSC-DS-1	Auxiliary Building
Division C 250 Vdc Switchboard 2	IDSC-DS-2	Auxiliary Building
Division D 250 Vdc Switchboard 1	IDSD-DS-1	Auxiliary Building
Division A Regulating Transformer	IDSA-DT-1	Auxiliary Building
Division B Regulating Transformer	IDSB-DT-1	Auxiliary Building
Division C Regulating Transformer	IDSC-DT-1	Auxiliary Building
Division D Regulating Transformer	IDSD-DT-1	Auxiliary Building
Division A 24-Hour Inverter 1	IDSA-DU-1	Auxiliary Building
Division B 24-Hour Inverter 1	IDSB-DU-1	Auxiliary Building
Division B 72-Hour Inverter 2	IDSB-DU-2	Auxiliary Building
Division C 24-Hour Inverter 1	IDSC-DU-1	Auxiliary Building
Division C 72-Hour Inverter 2	IDSC-DU-2	Auxiliary Building
Division D 24-Hour Inverter 1	IDSD-DU-1	Auxiliary Building
Spare Battery Termination Box	IDSS-DF-3	Auxiliary Building

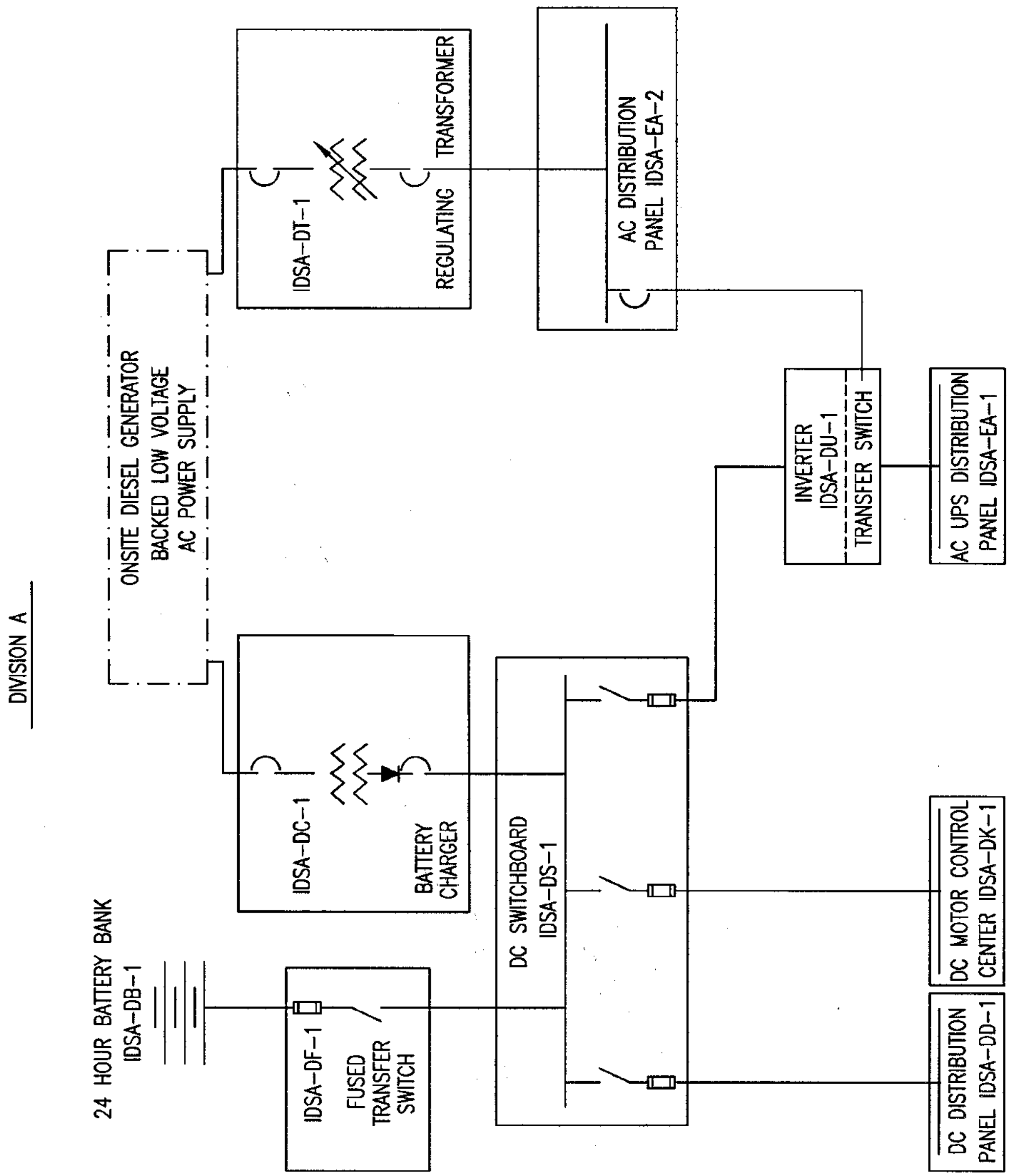


Figure 2.6.3-1 (Sheet 1 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division A)

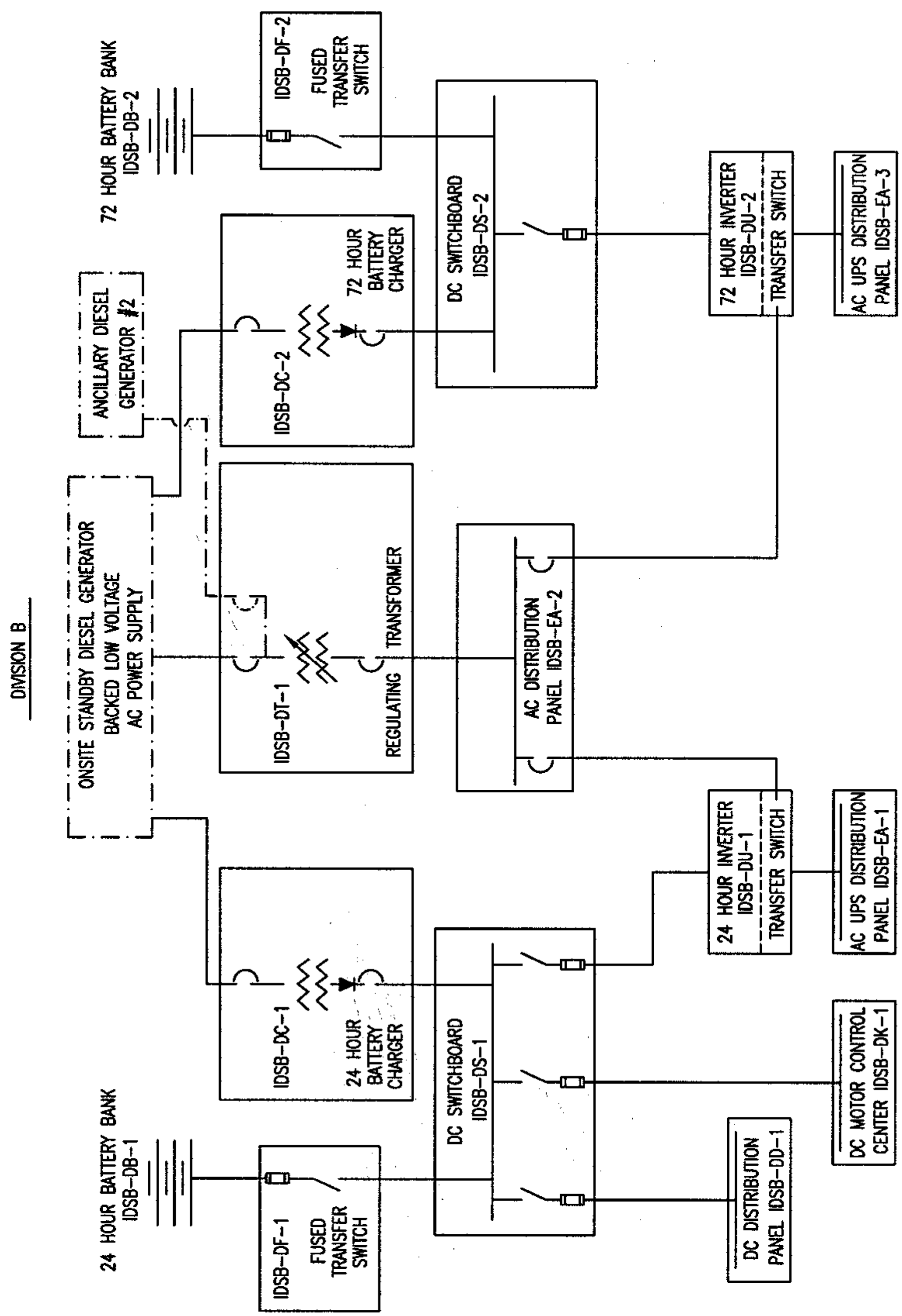


Figure 2.6.3-1 (Sheet 2 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division B)

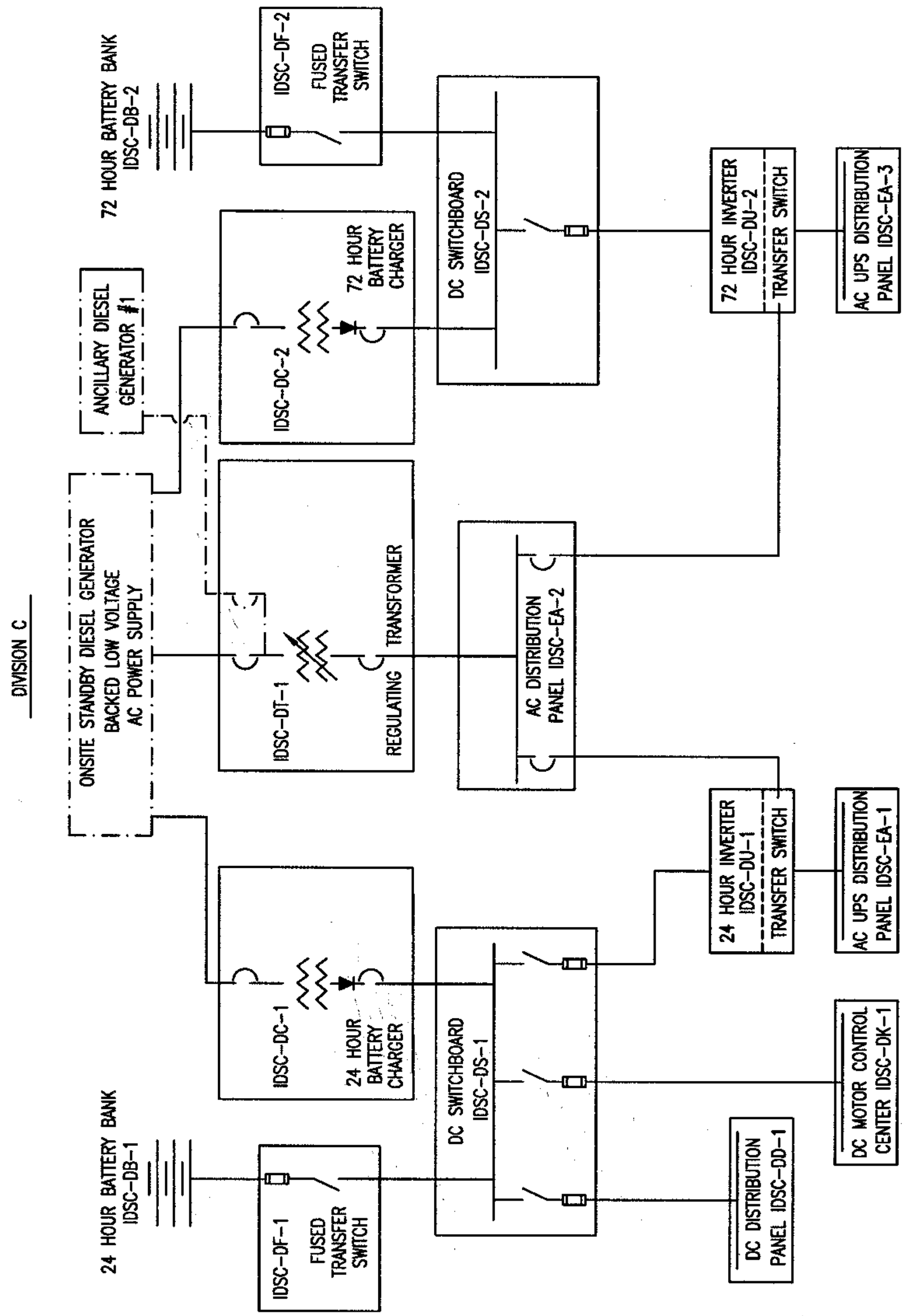


Figure 2.6.3-1 (Sheet 3 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division C)

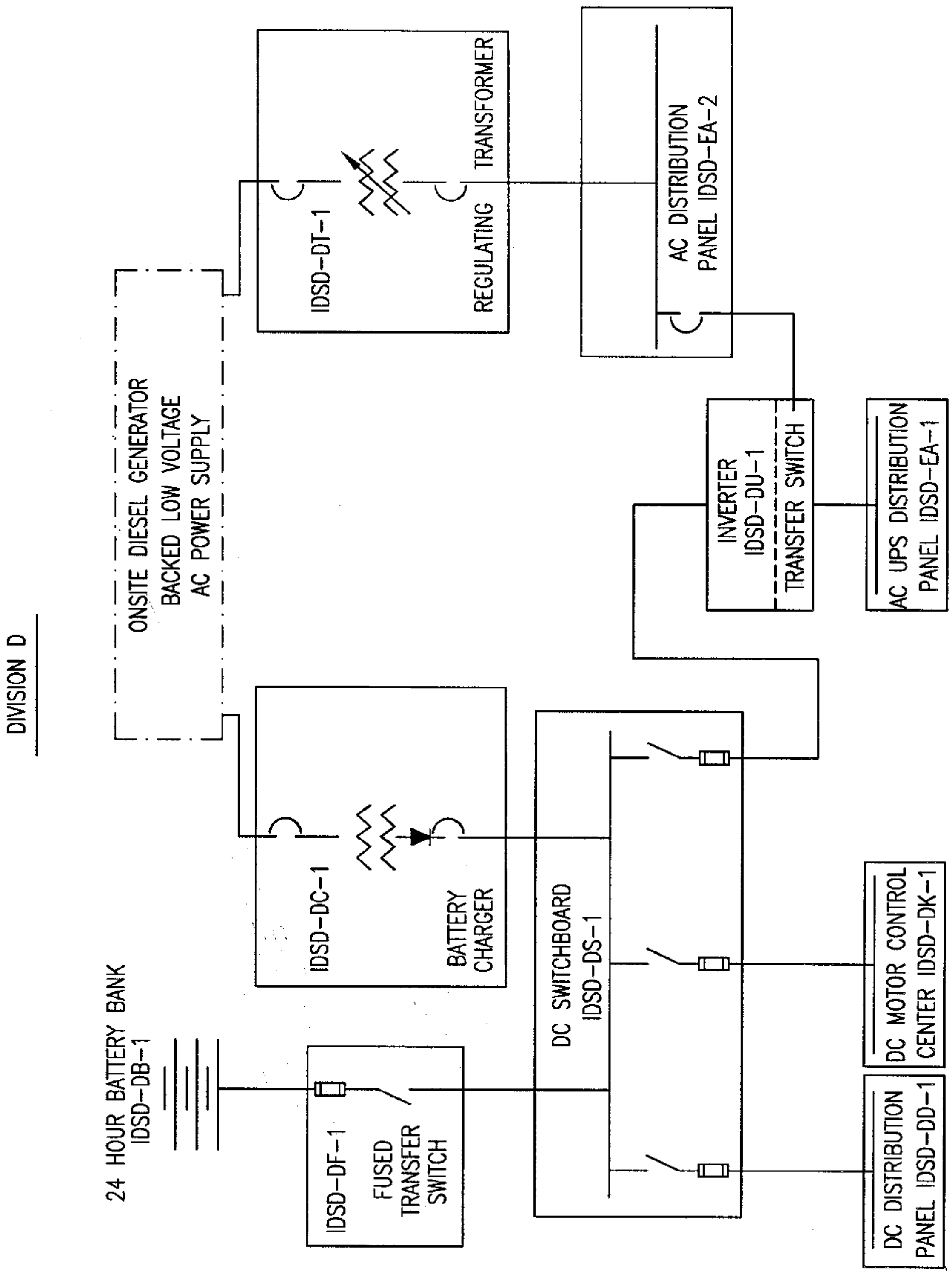


Figure 2.6.3-1 (Sheet 4 of 4)
Class 1E dc and Uninterruptible Power Supply System (Division D)

2.6.4 Onsite Standby Power System

Design Description

The onsite standby power system (ZOS) provides backup ac electrical power for nonsafety-related loads during normal and off-normal conditions.

The ZOS has two standby diesel generator units and the component locations of the ZOS are as shown in Table 2.6.4-2. The centerline of the diesel engine exhaust gas discharge is located more than twenty (20) feet higher than that of the combustion air intake.

1. The functional arrangement of the ZOS is as described in the Design Description of this Section 2.6.4.

2. The ZOS provides the following nonsafety-related functions:

   a) On loss of power to a 6900 volt diesel-backed bus, the associated diesel generator automatically starts and produces ac power at rated voltage and frequency. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus.

   b) Each diesel generator unit is sized to supply power to the selected nonsafety-related electrical components.

   c) Automatic-sequence loads are sequentially loaded on the associated buses.

3. Displays of diesel generator status (running/not running) and electrical output power (watts) can be retrieved in the main control room (MCR).

4. Controls exist in the MCR to start and stop each diesel generator.

Table 2.6.4-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
621	2.6.04.01	1. The functional arrangement of the ZOS is as described in the Design Description of this Section 2.6.4.	Inspection of the as-built system will be performed.	The as-built ZOS conforms with the functional arrangement as described in the Design Description of this Section 2.6.4.

Table 2.6.4-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
622	2.6.04.02a	2.a) On loss of power to a 6900 volt diesel-backed bus, the associated diesel generator automatically starts and produces ac power at rated voltage and frequency. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus.	Tests on the as-built ZOS system will be conducted by providing a simulated loss-of-voltage signal. The starting air supply receiver will not be replenished during the test.	Each as-built diesel generator automatically starts on receiving a simulated loss-of-voltage signal and attains a voltage of 6900 ± 10% V and frequency 60 ± 5% Hz after the start signal is initiated and opens ac power system breakers on the associated 6900 V bus.
623	2.6.04.02b	2.b) Each diesel generator unit is sized to supply power to the selected nonsafety-related electrical components.	Each diesel generator will be operated with a load of 4000 kW or greater and a power factor between 0.9 and 1.0 for a time period required to reach engine temperature equilibrium plus 2.5 hours.	Each diesel generator provides power to the load with a generator terminal voltage of 6900 ± 10% V and a frequency of 60 ± 5% Hz.
624	2.6.04.02c	2.c) Automatic-sequence loads are sequentially loaded on the associated buses.	An actual or simulated signal is initiated to start the load sequencer operation. Output signals will be monitored to determine the operability of the load sequencer. Time measurements are taken to determine the load stepping intervals.	The load sequencer initiates a closure signal within ±5 seconds of the set intervals to connect the loads.
625	2.6.04.03	3. Displays of diesel generator status (running/not running) and electrical output power (watts) can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays in the MCR.	Displays of diesel generator status and electrical output power can be retrieved in the MCR.
626	2.6.04.04	4. Controls exist in the MCR to start and stop each diesel generator.	A test will be performed to verify that controls in the MCR can start and stop each diesel generator.	Controls in the MCR operate to start and stop each diesel generator.

Table 2.6.4-2		
Component Name	**Tag No.**	**Component Location**
Onsite Diesel Generator A Package	ZOS-MS-05A	Diesel Generator Building
Onsite Diesel Generator B Package	ZOS-MS-05B	Diesel Generator Building

2.6.5 Lighting System

Design Description

The lighting system (ELS) provides the normal and emergency lighting in the main control room (MCR) and at the remote shutdown workstation (RSW).

1. The functional arrangement of the ELS is as described in the Design Description of this Section 2.6.5.
2. The ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. Each group is powered by one of the Class 1E inverters. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR. Each group is powered by one of the Class 1E inverters in Divisions B and C (one 24-hour and one 72-hour inverter in each Division).
3. The lighting fixtures located in the MCR utilize seismic supports.
4. The panel lighting circuits are classified as associated and treated as Class 1E. These lighting circuits are routed with the Divisions B and C Class 1E circuits. Separation is provided between ELS associated divisions and between associated divisions and non-Class 1E cable.
5. The normal lighting can provide 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.
6. The emergency lighting can provide 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.

Table 2.6.5-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
627	2.6.05.01	1. The functional arrangement of the ELS is as described in the Design Description of this Section 2.6.5.	Inspection of the as-built system will be performed.	The as-built ELS conforms with the functional arrangement as described in the Design Description of this Section 2.6.5.
628	2.6.05.02.i	2. The ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. Each group is powered by one of the Class 1E inverters. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR. Each group is powered by one of the Class 1E inverters in Divisions B and C (one 24-hour and one 72-hour inverter in each Division).	i) Inspection of the as-built system will be performed.	i) The as-built ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR.

Table 2.6.5-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
629	2.6.05.02.ii	2. The ELS has six groups of emergency lighting fixtures located in the MCR and at the RSW. Each group is powered by one of the Class 1E inverters. The ELS has four groups of panel lighting fixtures located on or near safety panels in the MCR. Each group is powered by one of the Class 1E inverters in Divisions B and C (one 24-hour and one 72-hour inverter in each Division).	ii) Testing of the as-built system will be performed using one Class 1E inverter at a time.	ii) Each of the six as-built emergency lighting groups is supplied power from its respective Class 1E inverter and each of the four as-built panel lighting groups is supplied power from its respective Class 1E inverter.
630	2.6.05.03.i	3. The lighting fixtures located in the MCR utilize seismic supports.	i) Inspection will be performed to verify that the lighting fixtures located in the MCR are located on the Nuclear Island.	i) The lighting fixtures located in the MCR are located on the Nuclear Island.
631	2.6.05.03.ii	3. The lighting fixtures located in the MCR utilize seismic supports.	ii) Analysis of seismic supports will be performed.	ii) A report exists and concludes that the seismic supports can withstand seismic design basis loads.
632	2.6.05.04	4. The panel lighting circuits are classified as associated and treated as Class 1E. These lighting circuits are routed with the Divisions B and C Class 1E circuits. Separation is provided between ELS associated divisions and between associated divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
633	2.6.05.05.i	5. The normal lighting can provide 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	i) Testing of the as-built normal lighting in the MCR will be performed.	i) When adjusted for maximum illumination and powered by the main ac power system, the normal lighting in the MCR provides at least 50 foot candles at the safety panel and at the workstations.
634	2.6.05.05.ii	5. The normal lighting can provide 50 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	ii) Testing of the as-built normal lighting at the RSW will be performed.	ii) When adjusted for maximum illumination and powered by the main ac power system, the normal lighting in the RSW provides at least 50 foot candles at the safety panel and at the workstations.

Table 2.6.5-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
635	2.6.05.06.i	6. The emergency lighting can provide 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	i) Testing of the as-built emergency lighting in the MCR will be performed.	i) When adjusted for maximum illumination and powered by the six Class 1E inverters, the emergency lighting in the MCR provides at least 10 foot candles at the safety panel and at the workstations.
636	2.6.05.06.ii	6. The emergency lighting can provide 10 foot candles at the safety panel and at the workstations in the MCR and at the RSW.	ii) Testing of the as-built emergency lighting at the RSW will be performed.	ii) When adjusted for maximum illumination and powered by the six Class 1E inverters, the emergency lighting provides at least 10 foot candles at the RSW.

2.6.6 Grounding and Lightning Protection System

Design Description

The grounding and lightning protection system (EGS) provides electrical grounding for instrumentation grounding, equipment grounding, and lightning protection during normal and off-normal conditions.

1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.

Table 2.6.6-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
637	2.6.06.01.i	1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, auxiliary and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.	i) An inspection for the instrument/computer grounding system connection to the station grounding grid will be performed.	i) A connection exists between the instrument/computer grounding system and the station grounding grid.
638	2.6.06.01.ii	1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, auxiliary and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.	ii) An inspection for the electrical system grounding connection to the station grounding grid will be performed.	ii) A connection exists between the electrical system grounding and the station grounding grid.

Table 2.6.6-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
639	2.6.06.01.iii	1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, auxiliary and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.	iii) An inspection for the equipment grounding system connection to the station grounding grid will be performed.	iii) A connection exists between the equipment grounding system and the station grounding grid.
640	2.6.06.01.iv	1. The EGS provides an electrical grounding system for: (1) instrument/computer grounding; (2) electrical system grounding of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, auxiliary and onsite standby diesel generators; and (3) equipment grounding of equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers, and control cabinets. Lightning protection is provided for exposed structures and buildings housing safety-related and fire protection equipment. Each grounding system and lightning protection system is grounded to the station grounding grid.	iv) An inspection for the lightning protection system connection to the station grounding grid will be performed.	iv) A connection exists between the lightning protection system and the station grounding grid.

2.6.7 Special Process Heat Tracing System

No entry for this system.

2.6.8 Cathodic Protection System

No entry.

2.6.9 Plant Security System

Design Description

The physical security system provides physical features to detect, delay, assist response to, and defend against the design basis threat (DBT) for radiological sabotage. The physical security system consists of physical barriers and an intrusion detection system. The details of the physical security system are categorized as Safeguards Information. The physical security system provides protection for vital equipment and plant personnel.

1. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.

2. Not used.

3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

4. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.

5. a) Security alarm annunciation and video assessment information is displayed concurrently in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter barrier.

   b) The central and secondary alarm stations are located inside the protected area, and the interior of each alarm station is not visible from the perimeter of the protected area.

   c) The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel.

1. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.

7. a) Vital equipment is located only within a vital area.

   b) Access to vital equipment requires passage through the vital area barrier.

8. Isolation zones and exterior areas within the protected area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

9. Emergency exits through the vital area boundaries are locked, alarmed, and equipped with a crash bar to allow for emergency egress.

10. Not used.

11. Not used.

12. Not used.

13. a) The central and secondary alarm stations have conventional (landline) telephone service with the main control room and local law enforcement authorities.

    b) The central and secondary alarm stations are capable of continuous communications with security personnel.

    c) Non-portable communication equipment in the central and secondary alarm stations remains operable from an independent power source in the event of loss of normal power.

14. Not used.

15. a) Security alarm devices including transmission lines to annunciators are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.

    b) Intrusion detection and assessment systems concurrently provide visual displays and audible annunciation of alarms in the central and secondary alarm station.

16. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

Table 2.6.9-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
641	2.6.09.01	1. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.	See ITAAC Table 3.3-6, item 14.	See ITAAC Table 3.3-6, item 14.
		2. Not used		
642	2.6.09.03	3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within the vital area.	See ITAAC Table 3.3-6, item 16.	See ITAAC Table 3.3-6, item 16.

Table 2.6.9-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
643	2.6.09.04	4. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.	See ITAAC Table 3.3-6, item 17.	See ITAAC Table 3.3-6, item 17.
644	2.6.09.05a	5.a) Security alarm annunciation and video assessment information is displayed concurrently in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability can provide assessment of activities before and after each alarm annunciation within the perimeter area barrier.	Test, inspection, or a combination of test and inspections of the installed systems will be performed.	Security alarm annunciation and video assessment information is displayed concurrently in the central alarm station and the secondary alarm station, and the video image recording with real time playback capability provides assessment of activities before and after alarm annunciation within the perimeter barrier.
645	2.6.09.05b	5.b) The central and secondary alarm stations are located inside the protected area and the interior of each alarm station is not visible from the perimeter of the protected area.	Inspections of the central and secondary alarm stations will be performed.	The central and secondary alarm stations are located inside the protected area and the interior of each alarm station is not visible from the perimeter of the protected area.
646	2.6.09.05c	5.c) The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, the design enables the survivability of equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel.	Inspections and/or analysis of the central and secondary alarm station will be performed.	The central and secondary alarm stations are designed and equipped such that, in the event of a single act, in accordance with the design basis threat of radiological sabotage, equipment needed to maintain the functional capability of either alarm station to detect and assess alarms and communicate with onsite and offsite response personnel exists.
647	2.6.09.06	6. The vehicle barrier system is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.	Inspections and analysis will be performed for the vehicle barrier system.	The vehicle barrier system will protect against the DBT vehicle bombs based upon the stand-off distance of the system.
648	2.6.09.07a	7.a) Vital equipment is located only within a vital area.	Inspection will be performed to confirm that vital equipment is located within a vital area.	All vital equipment is located only within a vital area.

Table 2.6.9-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
649	2.6.09.07b	7.b) Access to vital equipment requires passage through the vital area barrier.	Inspection will be performed to confirm that access to vital equipment requires passage through the vital area barrier.	Vital equipment is located within a protected area such that access to vital equipment requires passage through the vital area barrier.
650	2.6.09.08	8. Isolation zones and exterior areas within the protected area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.	Inspection of the illumination in the isolation zones and external areas of the protected area will be performed.	The illumination in isolation zones and exterior areas within the protected area is 0.2 foot candles measured horizontally at ground level or, alternatively, sufficient to permit observation.
651	2.6.09.09	9. Emergency exits through the vital area boundaries are locked, alarmed, and equipped with a crash bar to allow for emergency egress.	Test, inspection, or a combination of tests and inspections of the emergency exits through the vital area boundaries will be performed.	The emergency exits through the vital area boundaries are locked, alarmed, and equipped with a crash bar to allow for emergency egress.
		10. Not used		
		11. Not used		
		12. Not used		
652	2.6.09.13a	13.a) The central and secondary alarm stations have conventional (landline) telephone service with the main control room and local law enforcement authorities.	Tests, inspections, or a combination of tests and inspections of the central and secondary alarm stations' conventional telephone services will be performed.	The central and secondary alarm stations are equipped with conventional (landline) telephone service with the main control room and local law enforcement authorities.
653	2.6.09.13b	13.b) The central and secondary alarm stations are capable of continuous communication with security personnel.	Tests, inspections, or a combination of tests and inspections of the central and secondary alarm stations' continuous communication capabilities will be performed.	The central and secondary alarm stations are equipped with the capability to continuously communicate with security officers, watchmen, armed response individuals, or any security personnel that have responsibilities during a contingency event.
654	2.6.09.13c	13.c) Non-portable communication equipment in the central and secondary alarm stations remains operable from an independent power source in the event of loss of normal power.	Tests, inspections, or a combination of tests and inspections of the non-portable communications equipment will be performed.	Non-portable communication devices (including conventional telephone systems) in the central and secondary alarm stations are wired to an independent power supply that enables the system to remain operable in the event of loss of normal power.
		14. Not used.		

Table 2.6.9-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
655	2.6.09.15a	15.a) Security alarm devices, including transmission lines to annunciators, are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power). Alarm annunciation shall indicate the type of alarm (e.g., intrusion alarms and emergency exit alarm) and location.	A test will be performed to verify that security alarms, including transmission lines to annunciators, are tamper indicating and self-checking (e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when on standby power) and that alarm annunciation indicates the type of alarm (e.g., intrusion alarms and emergency exit alarms) and location.	A report exists and concludes that security alarm devices, including transmission lines to annunciators, are 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) and that alarm annunciation indicates the type of alarm (e.g., intrusion alarms and emergency exit alarms) and location.
656	2.6.09.15b	15.b) Intrusion detection and assessment systems concurrently provide visual displays and audible annunciation of alarms in the central and secondary alarm stations.	Tests will be performed on intrusion detection and assessment equipment.	The intrusion detection system concurrently provides visual displays and audible annunciations of alarms in both the central and secondary alarm stations.
657	2.6.09.16	16. Equipment exists to record onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.	Test, analysis, or a combination of test and analysis will be performed to ensure that equipment is capable of recording each onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.	A report exists and concludes that equipment is capable of recording each onsite security alarm annunciation, including the location of the alarm, false alarm, alarm check, and tamper indication; and the type of alarm, location, alarm circuit, date, and time.

C.2.6.9 Physical Security

Table C.2.6.9-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
658	C.2.6.09.01	1. The external walls, doors, ceiling, and floors in the location within which the last access control function for access to the protected area is performed are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.	Type test, analysis, or a combination of type test and analysis will be performed for the external walls, doors, ceilings, and floors in the location within which the last access control function for access to the protected area is performed.	The external walls, doors, ceilings, and floors in the location within which the last access control function for access to the protected area is performed are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.

Table C.2.6.9-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
659	C.2.6.09.02	2. Physical barriers for the protected area perimeter are not part of vital area barriers.	An inspection of the protected area perimeter barrier will be performed.	Physical barriers at the perimeter of the protected area are separated from any other barrier designated as a vital area barrier.
660	C.2.6.09.03a	3.a) Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area that allows 20 feet of observation on either side of the barrier. Where permanent buildings do not allow a 20 foot observation distance on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier.	Inspections will be performed of the isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area.	Isolation zones exist in outdoor areas adjacent to the physical barrier at the perimeter of the protected area and allow 20 feet of observation and assessment of the activities of people on either side of the barrier. Where permanent buildings do not allow a 20-foot observation and assessment distance on the inside of the protected area, the building walls are immediately adjacent to, or an integral part of, the protected area barrier and the 20-foot observation and assessment distance does not apply.
661	C.2.6.09.03b	3.b) The isolation zones are monitored with intrusion detection equipment that provides the capability to detect and assess unauthorized persons.	Inspections will be performed of the intrusion detection equipment within the isolation zones.	The isolation zones are equipped with intrusion detection equipment that provides the capability to detect and assess unauthorized persons.
662	C.2.6.09.04a	4. The intrusion detection and assessment equipment at the protected area perimeter: a) detects penetration or attempted penetration of the protected area barrier and concurrently alarms in both the Central Alarm Station and Secondary Alarm Station;	Tests, inspections or a combination of tests and inspections of the intrusion detection and assessment equipment at the protected area perimeter and its uninterruptible power supply will be performed.	The intrusion detection and assessment equipment at the protected area perimeter: a) detects penetration or attempted penetration of the protected area barrier and concurrently alarms in the central alarm station and secondary alarm station;

Table C.2.6.9-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
663	C.2.6.09.04b	4. The intrusion detection and assessment equipment at the protected area perimeter: b) remains operable from an uninterruptible power supply in the event of the loss of normal power.	Tests, inspections or a combination of tests and inspections of the intrusion detection and assessment equipment at the protected area perimeter and its uninterruptible power supply will be performed.	The intrusion detection and assessment equipment at the protected area perimeter: b) remains operable from an uninterruptible power supply in the event of the loss of normal power.
664	C.2.6.09.05a	5. Access control points are established to: a) control personnel and vehicle access into the protected area.	Tests, inspections, or combination of tests and inspections of installed systems and equipment at the access control points to the protected area will be performed.	The access control points for the protected area: a) are configured to control personnel and vehicle access.
665	C.2.6.09.05b	5. Access control points are established to: b) detect firearms, explosives, and incendiary devices at the protected area personnel access points.	Tests, inspections, or combination of tests and inspections of installed systems and equipment at the access control points to the protected area will be performed.	The access control points for the protected area: b) include detection equipment that is capable of detecting firearms, incendiary devices, and explosives at the protected area personnel access points.
666	C.2.6.09.06	6. An access control system with numbered picture badges is installed for use by individuals who are authorized access to protected areas and vital areas without escort.	A test of the access control system with numbered picture badges will be performed.	The access authorization system with numbered picture badges can identify and authorize protected area and vital area access only to those personnel with unescorted access authorization.
667	C.2.6.09.07	7. Access to vital equipment physical barriers requires passage through the protected area perimeter barrier.	Inspection will be performed to confirm that access to vital equipment physical barriers requires passage through the protected area perimeter barrier.	Vital equipment is located within a protected area such that access to vital equipment physical barriers requires passage through the protected area perimeter barrier.
668	C.2.6.09.08a	8.a) Penetrations through the protected area barrier are secured and monitored.	Inspections will be performed of penetrations through the protected area barrier.	Penetrations and openings through the protected area barrier are secured and monitored.

Table C.2.6.9-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
669	C.2.6.09.08b	8.b) Unattended openings (such as underground pathways) that intersect the protected area boundary or vital area boundary will be protected by a physical barrier and monitored by intrusion detection equipment or provided surveillance at a frequency sufficient to detect exploitation.	Inspections will be performed of unattended openings that intersect the protected area boundary or vital area boundary.	Unattended openings (such as underground pathways) that intersect the protected area boundary or vital area boundary are protected by a physical barrier and monitored by intrusion detection equipment or provided surveillance at a frequency sufficient to detect exploitation.
670	C.2.6.09.09	9. Emergency exits through the protected area perimeter are alarmed and secured with locking devices to allow for emergency egress.	Tests, inspections, or a combination of tests and inspections of emergency exits through the protected area perimeter will be performed.	Emergency exits through the protected area perimeter are alarmed and secured by locking devices that allow prompt egress during an emergency.

2.6.10 Main Generation System

No entry. Covered in Section 2.6.1, Main ac Power System.

2.6.11 Excitation and Voltage Regulation System

No entry for this system.

C.2.6.12 Transmission Switchyard and Offsite Power System

Table C.2.6.12-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
671	C.2.6.12.01	1. A minimum of one offsite circuit supplies electric power from the transmission network to the interface with the onsite alternating current (ac) power system.	Inspections of the as-built offsite circuit will be performed.	At least one offsite circuit is provided from the transmission switchyard interface to the interface with the onsite ac power system.
672	C.2.6.12.02	2. Each offsite power circuit interfacing with the onsite ac power system is adequately rated to supply assumed loads during normal, abnormal and accident conditions.	Analyses of the offsite power system will be performed to evaluate the as-built ratings of each offsite circuit interfacing with the onsite ac power system against the load assumptions.	A report exists and concludes that each as-built offsite circuit is rated to supply the load assumptions during normal, abnormal and accident conditions.

Table C.2.6.12-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
673	C.2.6.12.03	3. During steady state operation, each offsite power source is capable of supplying required voltage to the interface with the onsite ac power system that will support operation of assumed loads during normal, abnormal and accident conditions.	Analyses of the as-built offsite circuit will be performed to evaluate the capability of each offsite circuit to supply the voltage requirements at the interface with the onsite ac power system.	A report exists and concludes that during steady state operation each as-built offsite circuit is capable of supplying the voltage at the interface with the onsite ac power system that will support operation of assumed loads during normal, abnormal and accident conditions.
674	C.2.6.12.04	4. During steady state operation, each offsite circuit is capable of supplying required frequency to the interface with the onsite ac power system that will support operation of assumed loads during normal, abnormal and accident conditions.	Analyses of the as-built offsite circuit will be performed to evaluate the capability of each offsite circuit to supply the frequency requirements at the interface with the onsite ac power system.	A report exists and concludes that during steady state operation each as-built offsite circuit is capable of supplying the frequency at the interface with onsite ac power system that will support operation of assumed loads during normal, abnormal and accident conditions.
675	C.2.6.12.05	5. The fault current contribution of each offsite circuit is compatible with the interrupting capability of the onsite short circuit interrupting devices.	Analyses of the as-built offsite circuit will be performed to evaluate the fault current contribution of each offsite circuit at the interface with the onsite ac power system.	A report exists and concludes the short circuit contribution of each as-built offsite circuit at the interface with the onsite ac power system is compatible with the interrupting capability of the onsite fault current interrupting devices.
676	C.2.6.12.06	6. The reactor coolant pumps continue to receive power from either the main generator or the grid for a minimum of 3 seconds following a turbine trip.	Analyses of the as-built offsite power system will be performed to confirm that power will be available to the reactor coolant pumps for a minimum of 3 seconds following a turbine trip when the buses powering the reactor coolant pumps are aligned to either the unit auxiliary transformers (UATs) or the reserve auxiliary transformers (RATs).	A report exists and concludes that voltage at the high-side of the generator stepup transformer (GSU), and the RATs, does not drop more than 0.15 per unit (pu) from the pre-trip steady-state voltage for a minimum of 3 seconds following a turbine trip when the buses powering the reactor coolant pumps are aligned to either the UATs or the RATs.

2.7 HVAC Systems

2.7.1 Nuclear Island Nonradioactive Ventilation System

Design Description

The nuclear island nonradioactive ventilation system (VBS) serves the main control room (MCR), control support area (CSA), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room (RSR), reactor coolant pump trip switchgear rooms, adjacent corridors, and passive containment cooling system (PCS) valve room during normal plant operation. The VBS consists of the following independent subsystems: the main control room/control support area HVAC subsystem, the class 1E electrical room HVAC subsystem, and the passive containment cooling system valve room heating and ventilation subsystem. The VBS provides heating, ventilation, and cooling to the areas served when ac power is available. The system provides breathable air to the control room and maintains the main control room and control support area areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations. The VBS monitors the main control room supply air for radioactive particulate and iodine concentrations and provides filtration of main control room/control support area air during conditions of abnormal (high) airborne radioactivity. In addition, the VBS isolates the HVAC penetrations in the main control room boundary on "high-high" particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes. The Sanitary Drainage System (SDS) also isolates a penetration in the main control room boundary on “high-high” particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes. Additional penetrations from the SDS and Potable Water System (PWS) into the main control room boundary are maintained leak tight using a loop seal in the piping, and the Waste Water System (WWS) is isolated using a normally closed safety related manual isolation valve. These features support operation of the main control room emergency habitability system (VES), and have been included in Tables 2.7.1-1 and 2.7.1-2.

The VBS is as shown in Figure 2.7.1-1 and the component locations of the VBS are as shown in Table 2.7.1-5.

1. The functional arrangement of the VBS is as described in the Design Description of this subsection 2.7.1.
2. a) The components identified in Table 2.7.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.7.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.
3. a) Pressure boundary welds in components identified in Table 2.7.1-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.7.1-2 as ASME Code Section III meet ASME Code Section III requirements.
4. a) The components identified in Table 2.7.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.7.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. The seismic Category I equipment identified in Table 2.7.1-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E components identified in Table 2.7.1-1 are powered from their respective Class 1E division.

   b) Separation is provided between VBS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

7. The VBS and SDS provide the safety-related function to isolate the pipes that penetrate the MCR pressure boundary.

8. The VBS provides the following nonsafety-related functions:

   a) The VBS provides cooling to the MCR, CSA, RSR, and Class 1E electrical rooms.

   b) The VBS provides ventilation cooling to the Class 1E battery rooms.

   c) The VBS maintains MCR and CSA habitability when radioactivity is detected.

   d) The VBS provides ventilation cooling via the ancillary equipment in Table 2.7.1-3 to the MCR and the division B&C Class 1E I&C rooms.

9. Safety-related displays identified in Table 2.7.1-1 can be retrieved in the MCR.

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.7.1-1 to perform their active functions.

    b) The valves identified in Table 2.7.1-1 as having protection and safety monitoring system (PMS) control perform their active safety function after receiving a signal from the PMS.

11. After loss of motive power, the valves identified in Table 2.7.1-1 assume the indicated loss of motive power position.

12. Controls exist in the MCR to cause the components identified in Table 2.7.1-3 to perform the listed function.

13. Displays of the parameters identified in Table 2.7.1-3 can be retrieved in the MCR.

14. The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.

Table 2.7.1-1									
Equipment Name	**Tag No.**	**ASME Code Section III**	**Seismic Cat. I**	**Remotely Operated Valve**	**Class 1E/ Qual. for Harsh Envir.**	**Safety-Related Display**	**Control PMS/DAS[1]**	**Active Function**	**Loss of Motive Power Position**
MCR Supply Air Isolation Valve	VBS-PL-V186	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Supply Air Isolation Valve	VBS-PL-V187	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Return Air Isolation Valve	VBS-PL-V188	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Return Air Isolation Valve	VBS-PL-V189	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Exhaust Air Isolation Valve	VBS-PL-V190	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR Exhaust Air Isolation Valve	VBS-PL-V191	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
PWS MCR Isolation Valve	PWS-PL-V418	Yes	Yes	No	-/-	No	No	Transfer Closed	-
PWS MCR Isolation Valve	PWS-PL-V420	Yes	Yes	No	-/-	No	No	Transfer Closed	-
PWS MCR Vacuum Relief	PWS-PL-V498	Yes	Yes	No	-/-	No	No	Transfer Open	-
MCR SDS (Vent) Isolation Valve	SDS-PL-V001	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR SDS (Vent) Isolation Valve	SDS-PL-V002	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes/No	Transfer Closed	As Is
MCR WWS Isolation Valve	WWS-PL-V506	Yes	Yes	No	–	No	No	–	–

1. DAS = diverse actuation system

Table 2.7.1-2				
Line Name	**Line Number**	**ASME Code Section III**	**Leak Before Break**	**Functional Capability Required**
Main Control Room Supply	VBS-L311	Yes	No	No
Main Control Room Exhaust	VBS-L312	Yes	No	No
Main Control Room Toilet Exhaust	VBS-L313	Yes	No	No
Main Control Room Sanitary Vent Line	SDS-PL-L016	Yes	No	No
Main Control Room Sanitary Drain Line	SDS-PL-L179	Yes	No	No
Main Control Room Sanitary Drain Line	SDS-PL-L182	Yes	No	No
Main Control Room Water Line	PWS-PL-L319	Yes	No	No
Main Control Room Water Line	PWS-PL-L320	Yes	No	No
Main Control Room Waste Water Line	WWS-PL-L808	Yes	No	No
Main Control Room Water Line	WWS-PL-L851	Yes	No	No

Table 2.7.1-3			
Equipment	**Tag No.**	**Display**	**Control Function**
Supplemental Air Filtration Unit Fan A	VBS-MA-03A	Yes (Run Status)	Start
Supplemental Air Filtration Unit Fan B	VBS-MA-03B	Yes (Run Status)	Start
MCR/CSA Supply Air Handling Units (AHU) A Fans	VBS-MA-01A VBS-MA-02A	Yes (Run Status)	Start
MCR/CSA Supply AHU B Fans	VBS-MA-01B VBS-MA-02B	Yes (Run Status)	Start
Division "A" and "C" Class 1E Electrical Room AHU A Fans	VBS-MA-05A VBS-MA-06A	Yes (Run Status)	Start
Division "A" and "C" Class 1E Electrical Room AHU C Fans	VBS-MA-05C VBS-MA-06C	Yes (Run Status)	Start

Table 2.7.1-3			
Equipment	**Tag No.**	**Display**	**Control Function**
Division "B" and "D" Class 1E Electrical Room AHU D Fans	VBS-MA-05D VBS-MA-06D	Yes (Run Status)	Start
Division "A" and "C" Class 1E Battery Room Exhaust Fans	VBS-MA-07A VBS-MA-07C	Yes (Run Status)	Start
Division "B" and "D" Class 1E Battery Room Exhaust Fans	VBS-MA-07B VBS-MA-07D	Yes (Run Status)	Start
MCR Ancillary Fans	VBS-MA-10A VBS-MA-10B	No	Run
Division B Room Ancillary Fan	VBS-MA-11	No	Run
Division C Room Ancillary Fan	VBS-MA-12	No	Run

Table 2.7.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
677	2.7.01.01	1. The functional arrangement of the VBS is as described in the Design Description of this subsection 2.7.1	Inspection of the as-built system will be performed.	The as-built VBS conforms with the functional arrangement described in the Design Description of this subsection 2.7.1.
678	2.7.01.02a	2.a) The components identified in Table 2.7.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.7.1-1 as ASME Code Section III.
679	2.7.01.02b	2.b) The piping identified in Table 2.7.1-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME code Section III design reports exist for the as-built piping identified in Table 2.7.1-2 as ASME Code Section III.
680	2.7.01.03a	3.a) Pressure boundary welds in components identified in Table 2.7.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for nondestructive examination of pressure boundary welds.

Table 2.7.1-4
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
681	2.7.01.03b	3.b) Pressure boundary welds in piping identified in Table 2.7.1-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for nondestructive examination of pressure boundary welds.
682	2.7.01.04a	4.a) The components identified in Table 2.7.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A pressure test will be performed on the components required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the components identified in Table 2.7.1-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
683	2.7.01.04b	4.b) The piping identified in Table 2.7.1-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A pressure test will be performed on the piping required by the ASME Code Section III to be pressure tested.	A report exists and concludes that the results of the pressure test of the piping identified in Table 2.7.1-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
684	2.7.01.05.i	5. The seismic Category I equipment identified in Table 2.7.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.7.1-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.7.1-1 is located on the Nuclear Island.
685	2.7.01.05.ii	5. The seismic Category I equipment identified in Table 2.7.1-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
686	2.7.01.05.iii	5. The seismic Category I equipment identified in Table 2.7.1-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
687	2.7.01.06a	6.a) The Class 1E components identified in Table 2.7.1-1 are powered from their respective Class 1E division.	Testing will be performed on the VBS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.7.1-1 when the assigned Class 1E division is provided the test signal.

Table 2.7.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
688	2.7.01.06b	6.b) Separation is provided between VBS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d.	See ITAAC Table 3.3-6, item 7.d.
689	2.7.01.07	7. The VBS and SDS provide the safety-related function to isolate the pipe that penetrates the MCR pressure boundary.	See item 10.b in this table.	See item 10.b in this table.
690	2.7.01.08a	8.a) The VBS provides cooling to the MCR, CSA, RSR, and Class 1E electrical rooms.	See item 12 in this table.	See item 12 in this table.
691	2.7.01.08b	8.b) The VBS provides ventilation cooling to the Class 1E battery rooms.	See item 12 in this table.	See item 12 in this table.
692	2.7.01.08c	8.c) The VBS maintains MCR and CSA habitability when radioactivity is detected.	See item 12 in this table.	See item 12 in this table.
693	2.7.01.08d	8.d) The VBS provides ventilation cooling via the ancillary equipment in Table 2.7.1-3 to the MCR and the division B&C Class 1E I&C rooms.	Testing will be performed on the components in Table 2.7.1-3.	The fans start and run.
694	2.7.01.09	9. Safety-related displays identified in Table 2.7.1-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.7.1-1 can be retrieved in the MCR.
695	2.7.01.10a	10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.7.1-1 to perform their active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.7.1-1 using the controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves identified in Table 2.7.1-1 to perform their active functions.
696	2.7.01.10b	10.b) The valves identified in Table 2.7.1-1 as having PMS control perform their active safety function after receiving a signal from the PMS.	Testing will be performed using real or simulated signals into the PMS.	The valves identified in Table 2.7.1-1 as having PMS control perform their active safety function after receiving a signal from PMS.
697	2.7.01.11	11. After loss of motive power, the remotely operated valves identified in Table 2.7.1-1 assume the indicated loss of motive power position.	Testing of the remotely operated valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valves identified in Table 2.7.1-1 assumes the indicated loss of motive power position.

Table 2.7.1-4 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
698	2.7.01.12	12. Controls exist in the MCR to cause the components identified in Table 2.7.1-3 to perform the listed function.	Testing will be performed on the components in Table 2.7.1-3 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.1-3 to perform the listed functions.
699	2.7.01.13	13. Displays of the parameters identified in Table 2.7.1-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.1-3 can be retrieved in the MCR.
700	2.7.01.14	14. The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.	The as-built VBS will be operated, and background noise levels in the MCR and RSR will be measured.	The background noise level in the MCR and RSR does not exceed 65 dB(A) when the VBS is operating.

Table 2.7.1-5		
Component Name	**Tag No.**	**Component Location**
Supplemental Air Filtration Unit A	VBS-MS-01A	Auxiliary Building
Supplemental Air Filtration Unit B	VBS-MS-01B	Auxiliary Building
MCR/CSA Supply Air Handling Unit A	VBS-MS-02A	Auxiliary Building
MCR/CSA Supply Air Handling Unit B	VBS-MS-02B	Annex Building
Division "A" and "C" Class 1E Electrical Room AHU A	VBS-MS-03A	Auxiliary Building
Division "A" and "C" Class 1E Electrical Room AHU C	VBS-MS-03C	Auxiliary Building
Division "B" and "D" Class 1E Electrical Room AHU B	VBS-MS-03B	Auxiliary Building
Division "B" and "D" Class 1E Electrical Room AHU D	VBS-MS-03D	Auxiliary Building
MCR Toilet Exhaust Fan	VBS-MA-04	Auxiliary Building
Division "A&C" Class 1E Battery Room Exhaust Fan	VBS-MA-07A	Auxiliary Building
Division "A&C" Class 1E Battery Room Exhaust Fan	VBS-MA-07C	Auxiliary Building
Division "B&D" Class 1E Battery Room Exhaust Fan	VBS-MA-07B	Auxiliary Building

Table 2.7.1-5		
Component Name	**Tag No.**	**Component Location**
Division "B&D" Class 1E Battery Room Exhaust Fan	VBS-MA-07D	Auxiliary Building
PCS Valve Room Vent Fan	VBS-MA-08	Containment Shield Building
CSA Toilet Exhaust Fan	VBS-MA-09	Annex Building
MCR Ancillary Fan A	VBS-MA-10A	Auxiliary Building
MCR Ancillary Fan B	VBS-MA-10B	Auxiliary Building
Division B Ancillary Fan	VBS-MA-11	Auxiliary Building
Division C Ancillary Fan	VBS-MA-12	Auxiliary Building

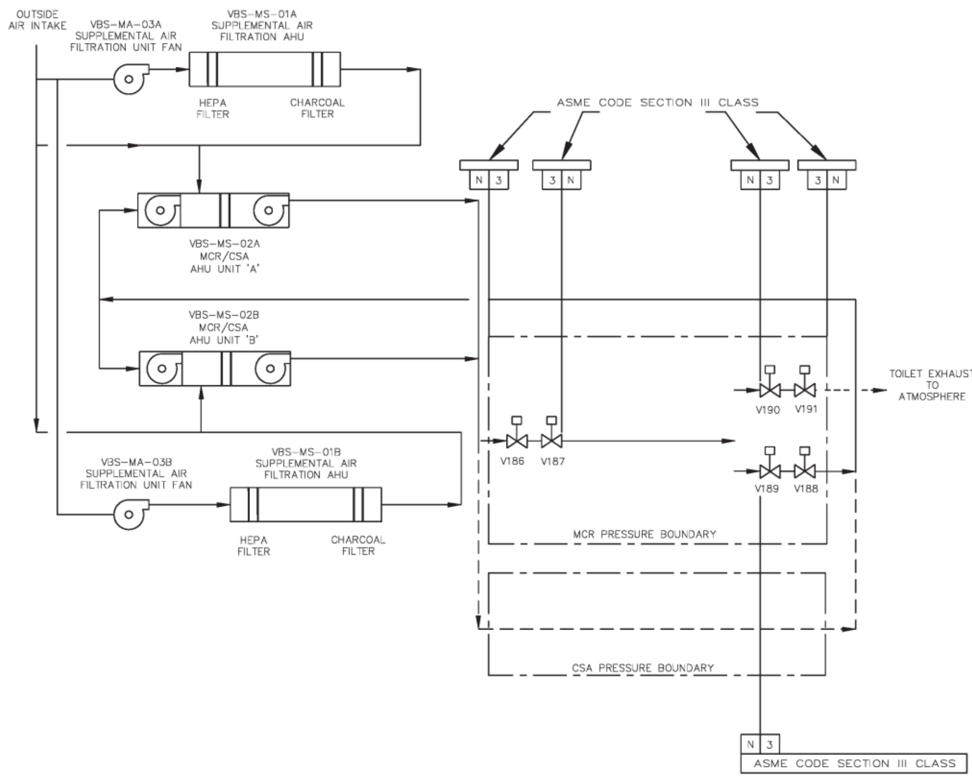


Figure 2.7.1-1 (Sheet 1 of 2)
Nuclear Island Nonradioactive Ventilation System

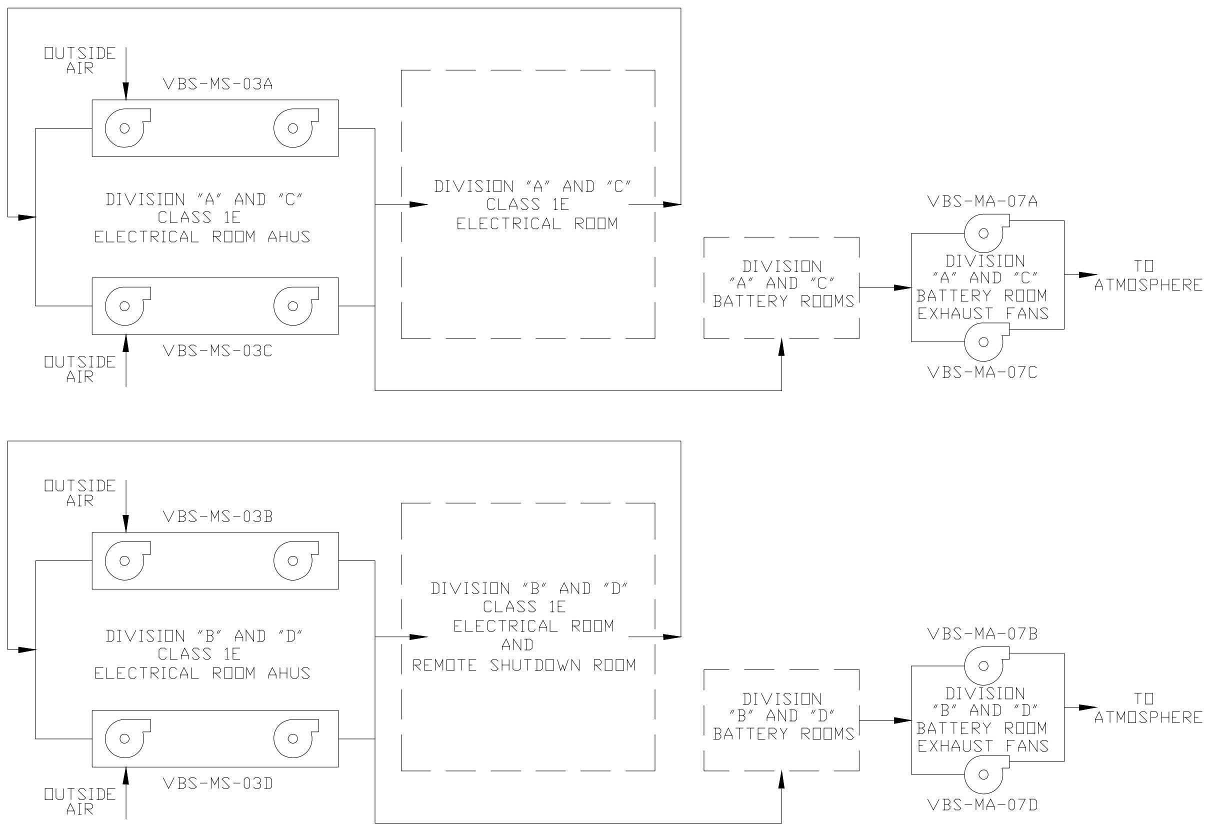


Figure 2.7.1-1 (Sheet 2 of 2)
Nuclear Island Nonradioactive Ventilation System

2.7.2 Central Chilled Water System

Design Description

The plant heating, ventilation, and air conditioning (HVAC) systems require chilled water as a cooling medium to satisfy the ambient air temperature requirements for the plant. The central chilled water system (VWS) supplies chilled water to the HVAC systems and is functional during reactor full-power and shutdown operation. The VWS also provides chilled water to selected process systems.

The VWS is as shown in Figure 2.7.2-1 and the component locations of the VWS are as shown Table 2.7.2-3.

1. The functional arrangement of the VWS is as described in the Design Description of this Section 2.7.2.
2. The VWS provides the safety-related function of preserving containment integrity by isolation of the VWS lines penetrating the containment.
3. The VWS provides the following nonsafety-related functions:
   a) The VWS provides chilled water to the supply air handling units serving the MCR, the Class 1E electrical rooms, and the unit coolers serving the RNS and CVS pump rooms.
   b) The VWS air-cooled chillers transfer heat from the VWS to the surrounding atmosphere.
4. Controls exist in the MCR to cause the components identified in Table 2.7.2-1 to perform the listed function.
5. Displays of the parameters identified in Table 2.7.2-1 can be retrieved in the MCR.

Table 2.7.2-1

Equipment Name	Tag No.	Display	Control Function
Air-cooled Chiller	VWS-MS-02	Yes (Run Status)	Start
Air-cooled Chiller	VWS-MS-03	Yes (Run Status)	Start
Air-cooled Chiller Pump	VWS-MP-02	Yes (Run Status)	Start
Air-cooled Chiller Pump	VWS-MP-03	Yes (Run Status)	Start
CVS Pump Room Unit Cooler Fan A	VAS-MA-07A	Yes (Run Status)	Start

Table 2.7.2-1			
Equipment Name	**Tag No.**	**Display**	**Control Function**
CVS Pump Room Unit Cooler Fan B	VAS-MA-07B	Yes (Run Status)	Start
RNS Pump Room Unit Cooler Fan A	VAS-MA-08A	Yes (Run Status)	Start
RNS Pump Room Unit Cooler Fan B	VAS-MA-08B	Yes (Run Status)	Start
Air-cooled Chiller Water Valve	VWS-PL-V210	Yes (Position Status)	Open
Air-cooled Chiller Water Valve	VWS-PL-V253	Yes (Position Status)	Open

Table 2.7.2-2

Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
701	2.7.02.01	1. The functional arrangement of the VWS is as described in the Design Description of this Section 2.7.2.	Inspection of the as-built system will be performed.	The as-built VWS conforms with the functional arrangement as described in the Design Description of this Section 2.7.2.
702	2.7.02.02	2. The applicable portions of the VWS provide the safety-related function of preserving containment integrity by isolation of the VWS lines penetrating the containment.	See ITAAC Table 2.2.1-3, items 1 and 7.	See ITAAC Table 2.2.1-3, items 1 and 7.
703	2.7.02.03a	3.a) The VWS provides chilled water to the supply air handling units serving the MCR, the Class 1E electrical rooms, and the unit coolers serving the RNS and CVS pump rooms.	Testing will be performed by measuring the flow rates to the chilled water cooling coils.	The water flow to each cooling coil equals or exceeds the following: Coil — Flow (gpm) VBS MY C01A/B — 138 VBS MY C02A/C — 108 VBS MY C02B/D — 84 VAS MY C07A/B — 24 VAS MY C12A/B — 15 VAS MY C06A/B — 15
704	2.7.02.03b	3.b) The VWS air-cooled chillers transfer heat from the VWS to the surrounding atmosphere.	Inspection will be performed for the existence of a report that determines the heat transfer capability of each air-cooled chiller.	A report exists and concludes that the heat transfer rate of each air-cooled chiller is greater than or equal to 230 tons.

Table 2.7.2-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
705	2.7.02.04	4. Controls exist in the MCR to cause the components identified in Table 2.7.2-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.2-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.2-1 to perform the listed functions.
706	2.7.02.05	5. Displays of the parameters identified in Table 2.7.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of parameters in the MCR.	The displays identified in Table 2.7.2-1 can be retrieved in the MCR.

Table 2.7.2-3		
Component Name	**Tag No.**	**Component Location**
Water Chiller Pump A	VWS-MP-01A	Turbine Building
Water Chiller Pump B	VWS-MP-01B	Turbine Building
Air Cooled Chiller Pump 2	VWS-MP-02	Auxiliary Building
Air Cooled Chiller Pump 3	VWS-MP-03	Annex Building
Water Chiller A	VWS-MS-01A	Turbine Building
Water Chiller B	VWS-MS-01B	Turbine Building
Air Cooled Chiller 2	VWS-MS-02	Auxiliary Building
Air Cooled Chiller 3	VWS-MS-03	Auxiliary Building

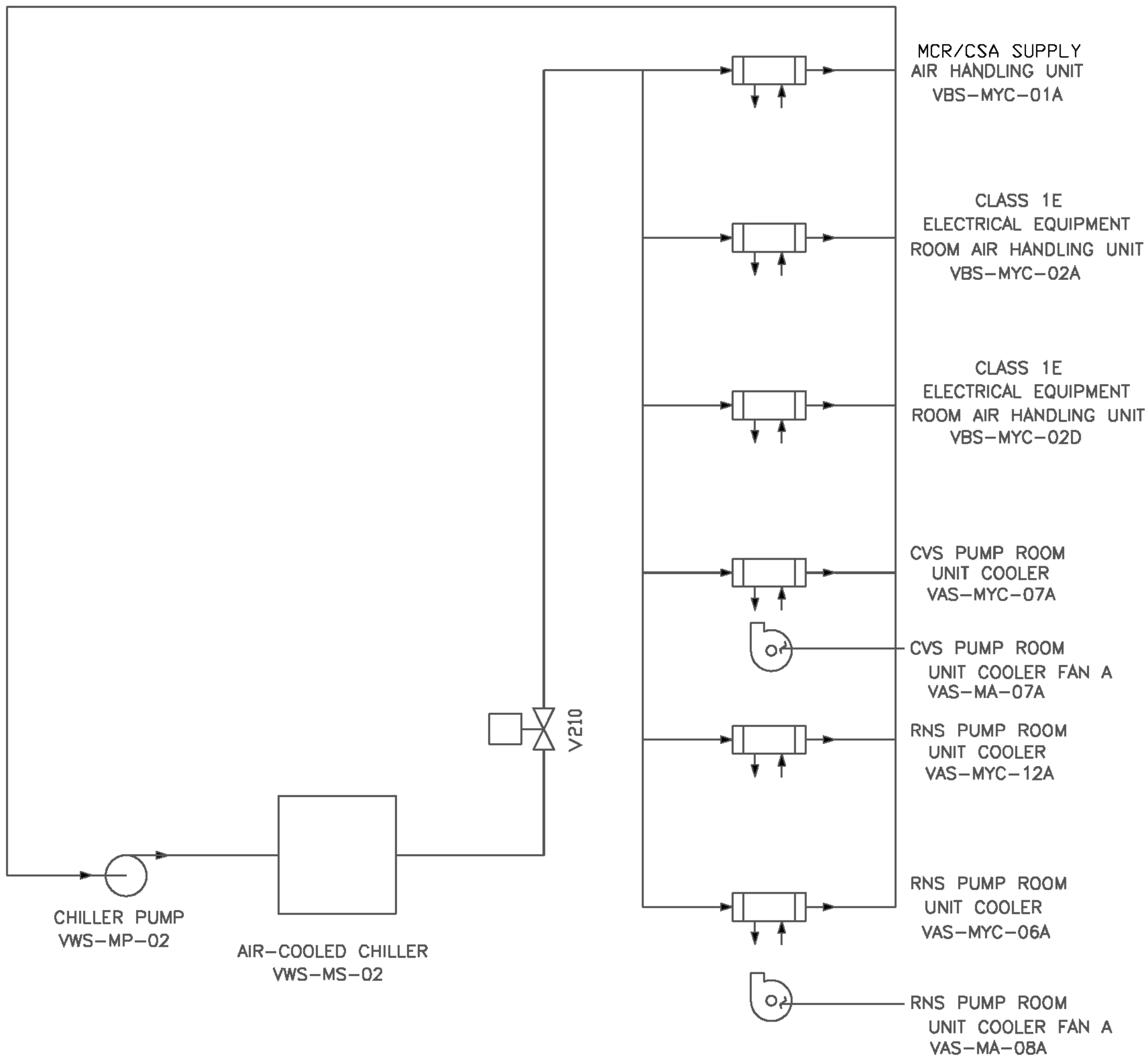


Figure 2.7.2-1 (Sheet 1 of 2)
Central Chilled Water System

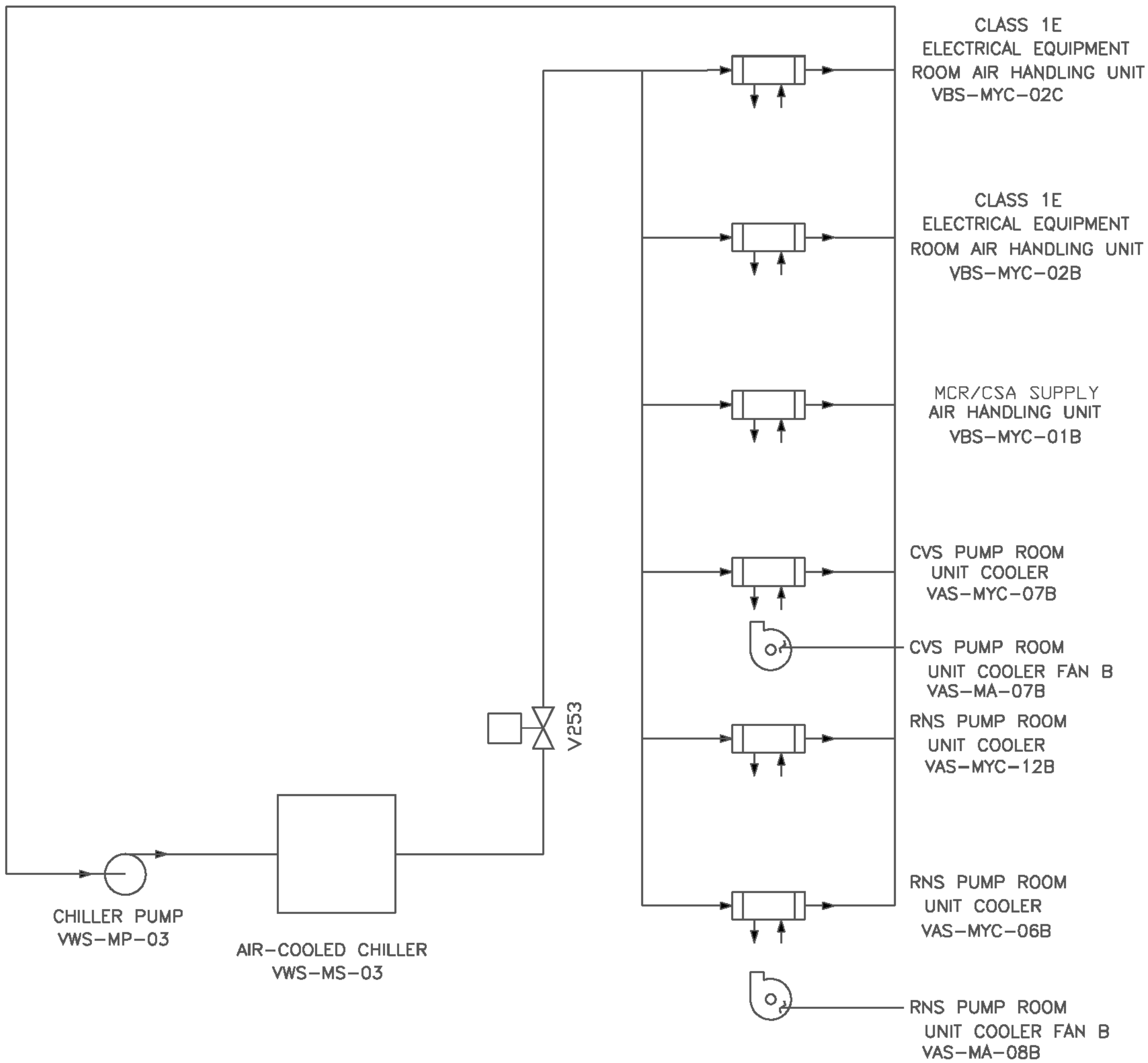


Figure 2.7.2-1 (Sheet 2 of 2)
Central Chilled Water System

2.7.3 Annex/Auxiliary Building Nonradioactive Ventilation System

Design Description

The annex/auxiliary buildings nonradioactive HVAC system (VXS) serves the nonradioactive personnel and equipment areas, electrical equipment rooms, clean corridors, the ancillary diesel generator room and demineralized water deoxygenating room in the annex building, and the main steam isolation valve compartments, reactor trip switchgear rooms, and piping and electrical penetration areas in the auxiliary building. The VXS consists of the following independent subsystems: the general area HVAC subsystem, the switchgear room HVAC subsystem, the equipment room HVAC subsystem, the MSIV compartment HVAC subsystem, the mechanical equipment areas HVAC subsystem and the valve/piping penetration room HVAC subsystem.

The VXS is as shown in Figure 2.7.3-1 and the component locations of the VXS are as shown in Table 2.7.3-3.

1. The functional arrangement of the VXS is as described in the Design Description of this Section 2.7.3.
2. The VXS provides the following nonsafety-related functions:
   a) The VXS provides cooling to the electrical switchgear, the battery charger, and the annex building nonradioactive air handling equipment rooms.
   b) The VXS provides ventilation cooling to the electrical switchgear, the battery charger, and the annex building nonradioactive air handling equipment rooms when the ZOS operates during a loss of offsite power coincident with loss of chilled water.
3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.7.3-1 to perform the listed function.
4. Displays of the parameters identified in Table 2.7.3-1 can be retrieved in the MCR.

Table 2.7.3-1

Equipment Name	Tag No.	Display	Control Function
Switchgear Room Air Handling Units (AHU) A Fans	VXS-MA-05A VXS-MA-06A	Yes (Run Status)	Start
Switchgear Room AHU B Fans	VXS-MA-05B VXS-MA-06B	Yes (Run Status)	Start
Equipment Room AHU A Fans	VXS-MA-01A VXS-MA-02A	Yes (Run Status)	Start
Equipment Room AHU B Fans	VXS-MA-01B VXS-MA-02B	Yes (Run Status)	Start

Table 2.7.3-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
707	2.7.03.01	1. The functional arrangement of the VXS is as described in the Design Description of this Section 2.7.3.	Inspection of the as-built system will be performed.	The as-built VXS conforms with the functional arrangement described in the Design Description of this Section 2.7.3.
708	2.7.03.02a	2.a) The VXS provides cooling to the electrical switchgear, the battery charger, and the annex building nonradioactive air handling equipment rooms when the ZOS operates and chilled water is available.	See item 3 in this table.	See item 3 in this table.
709	2.7.03.02b	2.b) The VXS provides ventilation cooling to the electrical switchgear, the battery charger, and the annex building nonradioactive air handling equipment rooms when the ZOS operates during a loss of offsite power coincident with loss of chilled water.	See item 3 in this table.	See item 3 in this table.
710	2.7.03.03	3. Controls exist in the MCR to cause the components identified in Table 2.7.3-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.3-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.3-1 to perform the listed functions.
711	2.7.03.04	4. Displays of the parameters identified in Table 2.7.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.3-1 can be retrieved in the MCR.

Table 2.7.3-3		
Component Name	**Tag No.**	**Component Location**
Annex Building General Area AHU A	VXS-MS-01A	Annex Building
Annex Building General Area AHU B	VXS-MS-01B	Annex Building
Annex Building Equipment Room AHU A	VXS-MS-02A	Annex Building
Annex Building Equipment Room AHU B	VXS-MS-02B	Annex Building
MSIV Compartment A AHU-A	VXS-MS-04A	Auxiliary Building
MSIV Compartment B AHU-B	VXS-MS-04B	Auxiliary Building
MSIV Compartment B AHU-C	VXS-MS-04C	Auxiliary Building
MSIV Compartment A AHU-D	VXS-MS-04D	Auxiliary Building
Switchgear Room AHU A	VXS-MS-05A	Annex Building

Table 2.7.3-3		
Component Name	**Tag No.**	**Component Location**
Switchgear Room AHU B	VXS-MS-05B	Annex Building
Mechanical Equipment Area AHU Unit A	VXS-MS-07A	Annex Building
Mechanical Equipment Area AHU Unit B	VXS-MS-07B	Annex Building
Valve/Piping Penetration Room AHU A	VXS-MS-08A	Auxiliary Building
Valve/Piping Penetration Room AHU B	VXS-MS-08B	Auxiliary Building
Battery Room #1 Exhaust Fan	VXS-MA-09A	Annex Building
Battery Room #2 Exhaust Fan	VXS-MA-09B	Annex Building
Toilet Exhaust Fan	VXS-MA-13	Annex Building
Annex Building Nonradioactive Air Handling Equipment Room Unit Heater A	VXS-MY-W01A	Annex Building
Annex Building Nonradioactive Air Handling Equipment Room Unit Heater B	VXS-MY-W01B	Annex Building
Annex Building Nonradioactive Air Handling Equipment Room Unit Heater C	VXS-MY-W01C	Annex Building

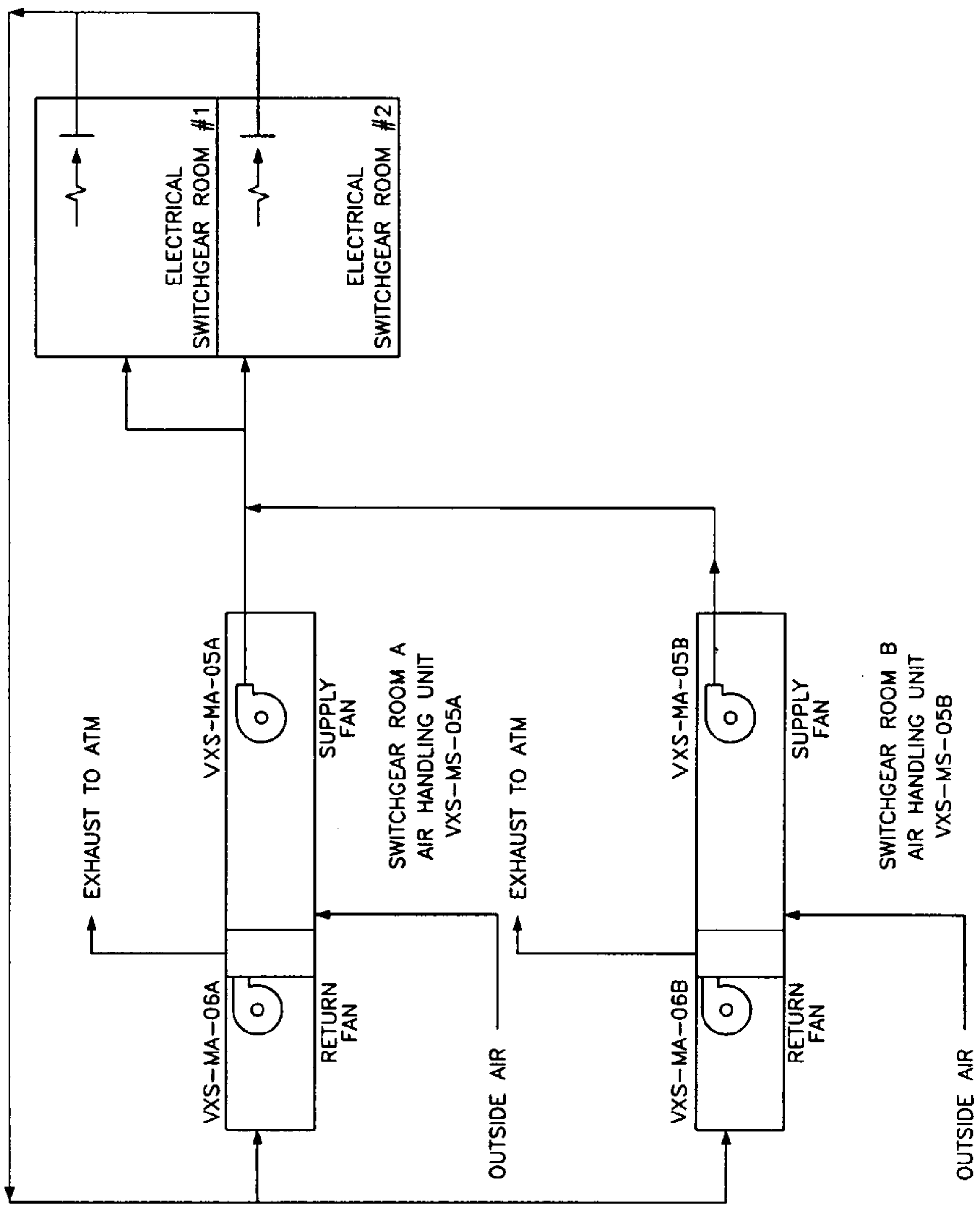


Figure 2.7.3-1 (Sheet 1 of 2)
Annex/Auxiliary Building Nonradioactive Ventilation System

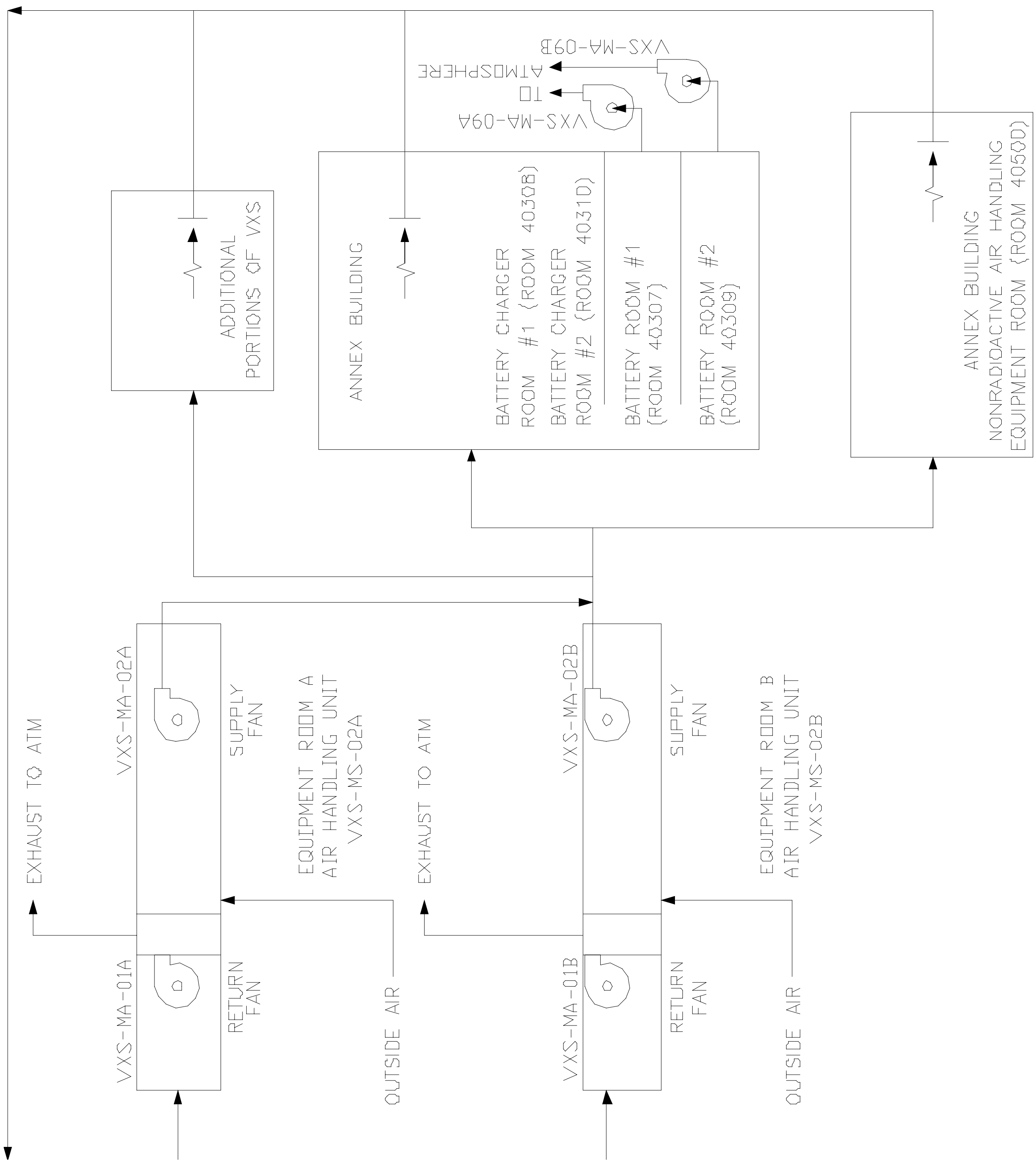


Figure 2.7.3-1 (Sheet 2 of 2)
Annex/Auxiliary Building Nonradioactive Ventilation System

2.7.4 Diesel Generator Building Ventilation System

Design Description

The diesel generator building ventilation system (VZS) provides ventilation cooling of the diesel generator building for the onsite standby power system. The VZS also provides heating and ventilation within the diesel oil transfer module enclosure. The VZS consists of the following subsystems: the normal diesel building heating and ventilation subsystem, the standby diesel building exhaust ventilation subsystem, the fuel oil day tank vault exhaust subsystem and the diesel oil transfer module enclosures ventilation and heating subsystem.

The VZS is as shown in Figure 2.7.4-1 and the component locations of the VZS are as shown in Table 2.7.4-3.

1. The functional arrangement of the VZS is as described in the Design Description of this Section 2.7.4.

2. The VZS provides the following nonsafety-related functions:

   a) The VZS provides ventilation cooling to the diesel generator rooms when the diesel generators are operating.

   b) The VZS provides ventilation cooling to the electrical equipment service modules when the diesel generators are operating.

   c) The VZS provides normal heating and ventilation to the diesel oil transfer module enclosure.

3. Controls exist in the main control room (MCR) to cause the components identified in Table 2.7.4-1 to perform the listed functions.

4. Displays of the parameters identified in Table 2.7.4-1 can be retrieved in the MCR.

Table 2.7.4-1

Equipment Name	Tag No.	Display	Control Function
Diesel Generator Room A Standby Exhaust Fans	VZS-MY-V01A VZS-MY-V02A	Yes (Run Status)	Start
Diesel Generator Room B Standby Exhaust Fans	VZS-MY-V01B VZS-MY-V02B	Yes (Run Status)	Start
Service Module A Air Handling Units (AHU) Supply Fan	VZS-MA-01A	Yes (Run Status)	Start
Service Module B AHU Supply Fan	VZS-MA-01B	Yes (Run Status)	Start

Table 2.7.4-1			
Equipment Name	**Tag No.**	**Display**	**Control Function**
Diesel Oil Transfer Module Enclosure A Exhaust Fan	VZS-MY-V03A	Yes (Run Status)	Start
Diesel Oil Transfer Module Enclosure A Electric Unit Heater	VZS-MY-U03A	Yes (Run Status)	Energize
Diesel Oil Transfer Module Enclosure B Exhaust Fan	VZS-MY-V03B	Yes (Run Status)	Start
Diesel Oil Transfer Module Enclosure B Electric Unit Heater	VZS-MY-U03B	Yes (Run Status)	Energize

Table 2.7.4-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
712	2.7.04.01	1. The functional arrangement of the VZS is as described in the Design Description of this Section 2.7.4.	Inspection of the as-built system will be performed.	The as-built VZS conforms with the functional arrangement described in the Design Description of this Section 2.7.4.
713	2.7.04.02a	2.a) The VZS provides ventilation cooling to the diesel generator rooms when the diesel generators are operating.	See item 3 in this table.	See item 3 in this table.
714	2.7.04.02b	2.b) The VZS provides ventilation cooling to the electrical equipment service modules when the diesel generators are operating.	See item 3 in this table.	See item 3 in this table.
715	2.7.04.02c	2.c) The VZS provides normal heating and ventilation to the diesel oil transfer module enclosure.	See item 3 in this table.	See item 3 in this table.
716	2.7.04.03	3. Controls exist in the MCR to cause the components identified in Table 2.7.4-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.4-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.4-1 to perform the listed functions.
717	2.7.04.04	4. Displays of the parameters identified in Table 2.7.4-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.4-1 can be retrieved in the MCR.

Table 2.7.4-3		
Component Name	**Tag No.**	**Component Location**
Service Module AHU A	VZS-MS-01A	Diesel-Generator Building
Service Module AHU B	VZS-MS-01B	Diesel-Generator Building
Diesel Oil Transfer Module Enclosure A Unit Heater	VZS-MY-U03A	Yard
Diesel Oil Transfer Module Enclosure B Unit Heater	VZS-MY-U03B	Yard
D/G Building Standby Exhaust Fan 1A	VZS-MY-V01A	Diesel-Generator Building
D/G Building Standby Exhaust Fan 1B	VZS-MY-V01B	Diesel-Generator Building
D/G Building Standby Exhaust Fan 2A	VZS-MY-V02A	Diesel-Generator Building
D/G Building Standby Exhaust Fan 2B	VZS-MY-V02B	Diesel-Generator Building
Diesel Oil Transfer Module Enclosure A Exhaust Fan	VZS-MY-V03A	Yard
Diesel Oil Transfer Module Enclosure B Exhaust Fan	VZS-MY-V03B	Yard
Fuel Oil Day Tank Vault Exhaust Fan	VZS-MA-02A	Diesel-Generator Building
Fuel Oil Day Tank Vault Exhaust Fan	VZS-MA-02B	Diesel-Generator Building

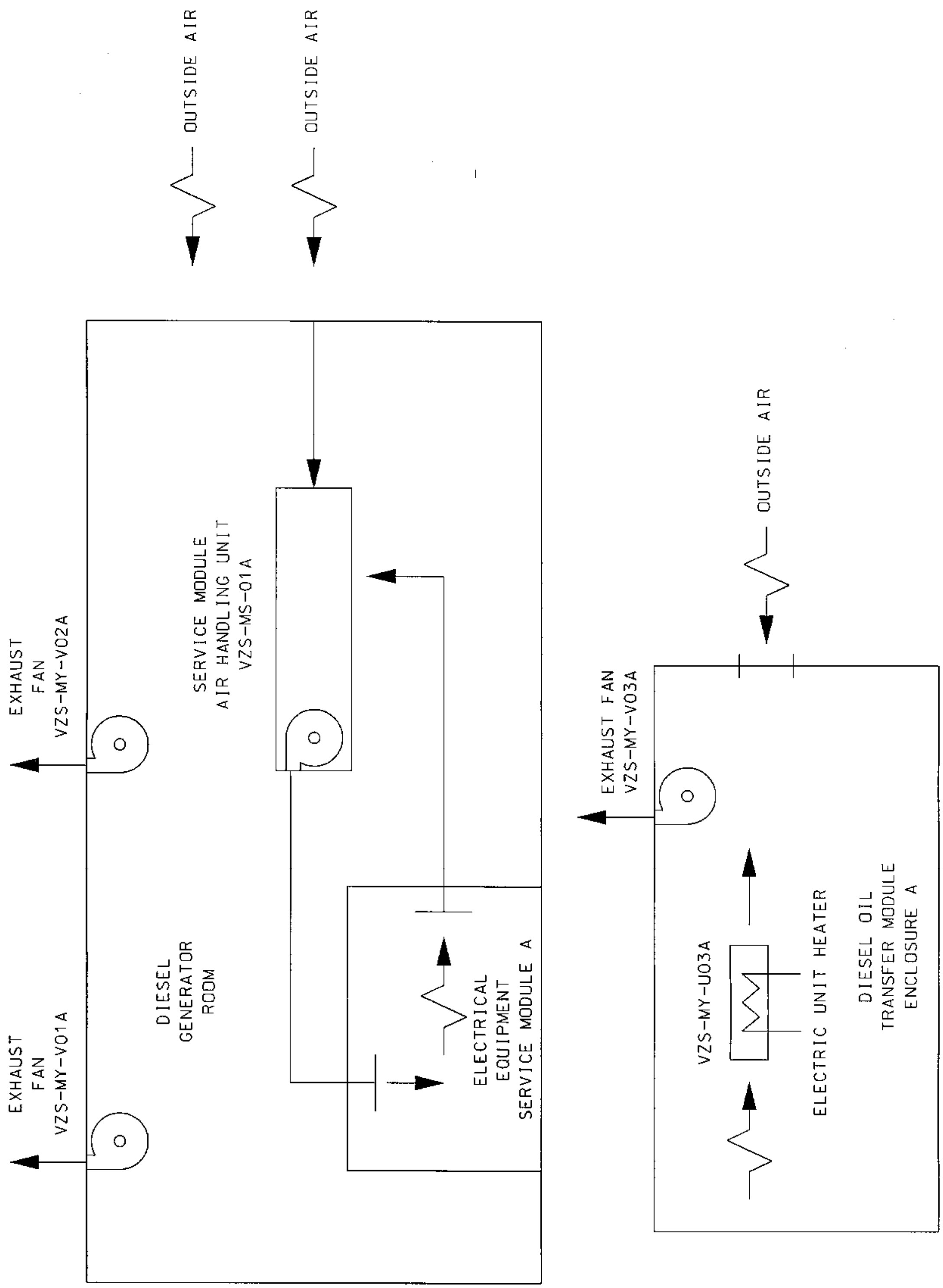


Figure 2.7.4-1 (Sheet 1 of 2)
Diesel Generator Building Ventilation System

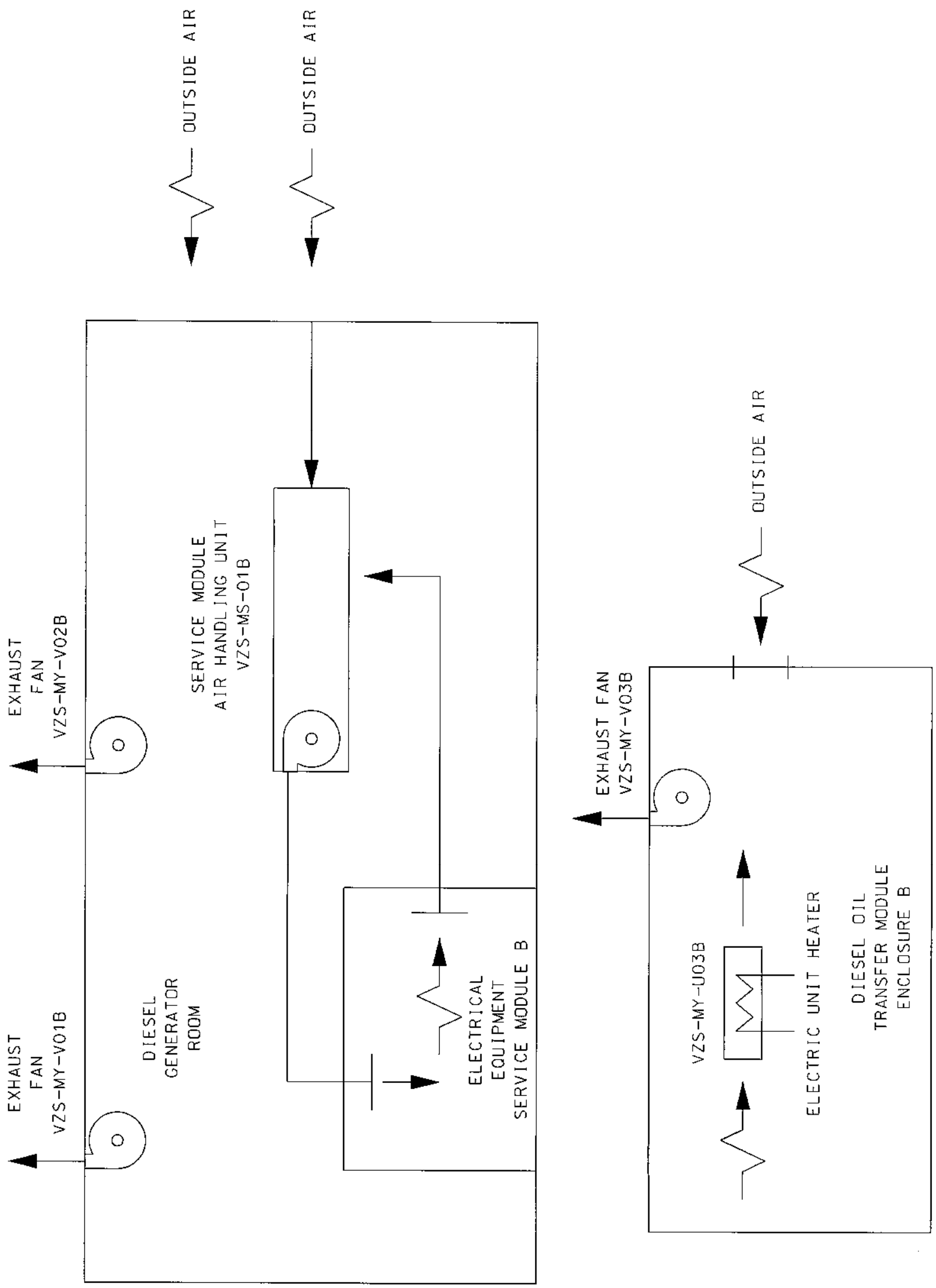


Figure 2.7.4-1 (Sheet 2 of 2)
Diesel Generator Building Ventilation System

2.7.5 Radiologically Controlled Area Ventilation System

Design Description

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas, which are provided with a separate ventilation system (VHS). The VAS consists of two subsystems: the auxiliary/annex building ventilation subsystem and the fuel handling area ventilation subsystem. The subsystems provide ventilation to maintain occupied areas, and access and equipment areas within their design temperature range. They provide outside air for plant personnel and prevent the unmonitored release of airborne radioactivity to the atmosphere or adjacent plant areas. The VAS automatically isolates selected building areas by closing the supply and exhaust duct isolation dampers and starts the containment air filtration system (VFS) when high airborne radioactivity in the exhaust air duct or high ambient pressure differential is detected.

The component locations of the VAS are as shown in Table 2.7.5-3.

1. The functional arrangement of the VAS is as described in the Design Description of this Section 2.7.5.
2. The VAS maintains each building area at a slightly negative pressure relative to the atmosphere or adjacent clean plant areas.
3. Displays of the parameters identified in Table 2.7.5-1 can be retrieved in the main control room (MCR).

Table 2.7.5-1

Equipment	Tag No.	Display	Control Function
Annex Building Pressure Differential Indicator	VAS-032	Yes	-
Auxiliary Building Pressure Differential Indicator	VAS-033	Yes	-
Fuel Handling Area Pressure Differential Indicator	VAS-030	Yes	-

Note: Dash (-) indicates not applicable.

Table 2.7.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
718	2.7.05.01	1. The functional arrangement of the VAS is as described in the Design Description of this Section 2.7.5.	Inspection of the as-built system will be performed.	The as-built VAS conforms with the functional arrangement described in the Design Description of this Section 2.7.5.

Table 2.7.5-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
719	2.7.05.02.i	2. The VAS maintains each building area at a slightly negative pressure relative to the atmosphere or adjacent clean plant areas.	i) Testing will be performed to confirm that the VAS maintains each building at a slightly negative pressure when operating all VAS supply AHUs and all VAS exhaust fans.	i) The time average pressure differential in the served areas of the annex, fuel handling and radiologically controlled auxiliary buildings as measured by each of the instruments identified in Table 2.7.5-1 is negative.
720	2.7.05.02.ii	2. The VAS maintains each building area at a slightly negative pressure relative to the atmosphere or adjacent clean plant areas.	ii) Testing will be performed to confirm the ventilation flow rate through the auxiliary building fuel handling area when operating all VAS supply AHUs and all VAS exhaust fans.	ii) A report exists and concludes that the calculated exhaust flow rate based on the measured flow rates is greater than or equal to 15,300 cfm.
721	2.7.05.02.iii	2. The VAS maintains each building area at a slightly negative pressure relative to the atmosphere or adjacent clean plant areas.	iii) Testing will be performed to confirm the auxiliary building radiologically controlled area ventilation flow rate when operating all VAS supply AHUs and all VAS exhaust fans.	iii) A report exists and concludes that the calculated exhaust flow rate based on the measured flow rates is greater than or equal to 22,500 cfm.
722	2.7.05.03	3. Displays of the parameters identified in Table 2.7.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.5-1 can be retrieved in the MCR.

Table 2.7.5-3		
Component Name	**Tag No.**	**Component Location**
Auxiliary/Annex Building Supply AHU A	VAS-MS-01A	Annex Building
Auxiliary/Annex Building Supply AHU B	VAS-MS-01B	Annex Building
Fuel Handling Area Supply AHU A	VAS-MS-02A	Annex Building
Fuel Handling Area Supply AHU B	VAS-MS-02B	Annex Building
CVS Pump Room Unit Cooler A	VAS-MS-05A	Auxiliary Building
CVS Pump Room Unit Cooler B	VAS-MS-05B	Auxiliary Building
RNS Pump Room Unit Cooler A	VAS-MS-06A	Auxiliary Building
RNS Pump Room Unit Cooler B	VAS-MS-06B	Auxiliary Building
Auxiliary/Annex Building Exhaust Fan A	VAS-MA-02A	Auxiliary Building
Auxiliary/Annex Building Exhaust Fan B	VAS-MA-02B	Auxiliary Building

Table 2.7.5-3		
Component Name	**Tag No.**	**Component Location**
Fuel Handling Area Exhaust Fan A	VAS-MA-06A	Auxiliary Building
Fuel Handling Area Exhaust Fan B	VAS-MA-06B	Auxiliary Building

2.7.6 Containment Air Filtration System

Design Description

The containment air filtration system (VFS) provides intermittent flow of outdoor air to purge and filter the containment atmosphere of airborne radioactivity during normal plant operation, and continuous flow during hot or cold plant shutdown conditions to reduce airborne radioactivity levels for personnel access. The VFS can also provide filtered exhaust for the radiologically controlled area ventilation system (VAS) during abnormal conditions.

The VFS is as shown in Figure 2.7.6-1 and the component locations of the VFS are as shown in Table 2.7.6-3.

1. The functional arrangement of the VFS is as described in the Design Description of this Section 2.7.6.
2. The VFS provides the safety-related functions of preserving containment integrity by isolation of the VFS lines penetrating containment and providing vacuum relief for the containment vessel.
3. The VFS provides the intermittent flow of outdoor air to purge the containment atmosphere during normal plant operation, and continuous flow during hot or cold plant shutdown conditions.
4. Controls exist in the main control room (MCR) to cause the components identified in Table 2.7.6-1 to perform the listed function.
5. Displays of the parameters in Table 2.7.6-1 can be retrieved in the MCR.

Table 2.7.6-1			
Equipment	**Tag No.**	**Display**	**Control Function**
Containment Air Handling Units (AHU) Supply Fan A	VFS-MA-01A	Yes (Run Status)	Start
Containment AHU Supply Fan B	VFS-MA-01B	Yes (Run Status)	Start
Containment AHU Supply Fan A Flow Sensor	VFS-012A	Yes	-
Containment AHU Supply Fan B Flow Sensor	VFS-012B	Yes	-

Table 2.7.6-1			
Equipment	**Tag No.**	**Display**	**Control Function**
Containment Exhaust Fan A	VFS-MA-02A	Yes (Run Status)	Start
Containment Exhaust Fan B	VFS-MA-02B	Yes (Run Status)	Start
Containment Exhaust Fan A Flow Sensor	VFS-011A	Yes	-
Containment Exhaust Fan B Flow Sensor	VFS-011B	Yes	-

Table 2.7.6-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
723	2.7.06.01	1. The functional arrangement of the VFS is as described in the Design Description of this Section 2.7.6.	Inspection of the as-built system will be performed.	The as-built VFS conforms with the functional arrangement described in the Design Description of this Section 2.7.6.
724	2.7.06.02.i	2. The VFS provides the safety-related functions of preserving containment integrity by isolation of the VFS lines penetrating containment and providing vacuum relief for the containment vessel.	i) See ITAAC Table 2.2.1-3, items 1 and 7.	i) See ITAAC Table 2.2.1-3, items 1 and 7.
725	2.7.06.02.ii	2. The VFS provides the safety-related functions of preserving containment integrity by isolation of the VFS lines penetrating containment and providing vacuum relief for the containment vessel.	ii) Testing will be performed to demonstrate that remotely operated containment vacuum relief isolation valves open within the required response time.	ii) The containment vacuum relief isolation valves (VFS-PL-V800A and VFS-PL-V800B) open within 30 seconds.
726	2.7.06.03.i	3. The VFS provides the intermittent flow of outdoor air to purge the containment atmosphere during normal plant operation, and continuous flow during hot or cold plant shutdown conditions.	i) Testing will be performed to confirm that containment supply AHU fan A when operated with containment exhaust fan A provides a flow of outdoor air.	i) The flow rate measured at each fan is greater than or equal to 3,600 scfm.
727	2.7.06.03.ii	3. The VFS provides the intermittent flow of outdoor air to purge the containment atmosphere during normal plant operation, and continuous flow during hot or cold plant shutdown conditions.	ii) Testing will be performed to confirm that containment supply AHU fan B when operated with containment exhaust fan B provides a flow of outdoor air.	ii) The flow rate measured at each fan is greater than or equal to 3,600 scfm.

Table 2.7.6-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
728	2.7.06.03.iii	3. The VFS provides the intermittent flow of outdoor air to purge the containment atmosphere during normal plant operation, and continuous flow during hot or cold plant shutdown conditions.	iii) Inspection will be conducted of the containment purge discharge line (VFS-L204) penetrating the containment.	iii) The nominal line size is ≥ 36 in.
729	2.7.06.04	4. Controls exist in the MCR to cause the components identified in Table 2.7.6-1 to perform the listed function.	Testing will be performed on the components in Table 2.7.6-1 using controls in the MCR.	Controls in the MCR operate to cause the components listed in Table 2.7.6-1 to perform the listed functions.
730	2.7.06.05	5. Displays of the parameters identified in Table 2.7.6-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.6-1 can be retrieved in the MCR.

Table 2.7.6-3		
Component Name	**Tag No.**	**Component Location**
Containment Air Filtration Supply AHU A	VFS-MS-01A	Annex Building
Containment Air Filtration Supply AHU B	VFS-MS-01B	Annex Building
Containment Air Filtration Exhaust Unit A	VFS-MS-02A	Annex Building
Containment Air Filtration Exhaust Unit B	VFS-MS-02B	Annex Building

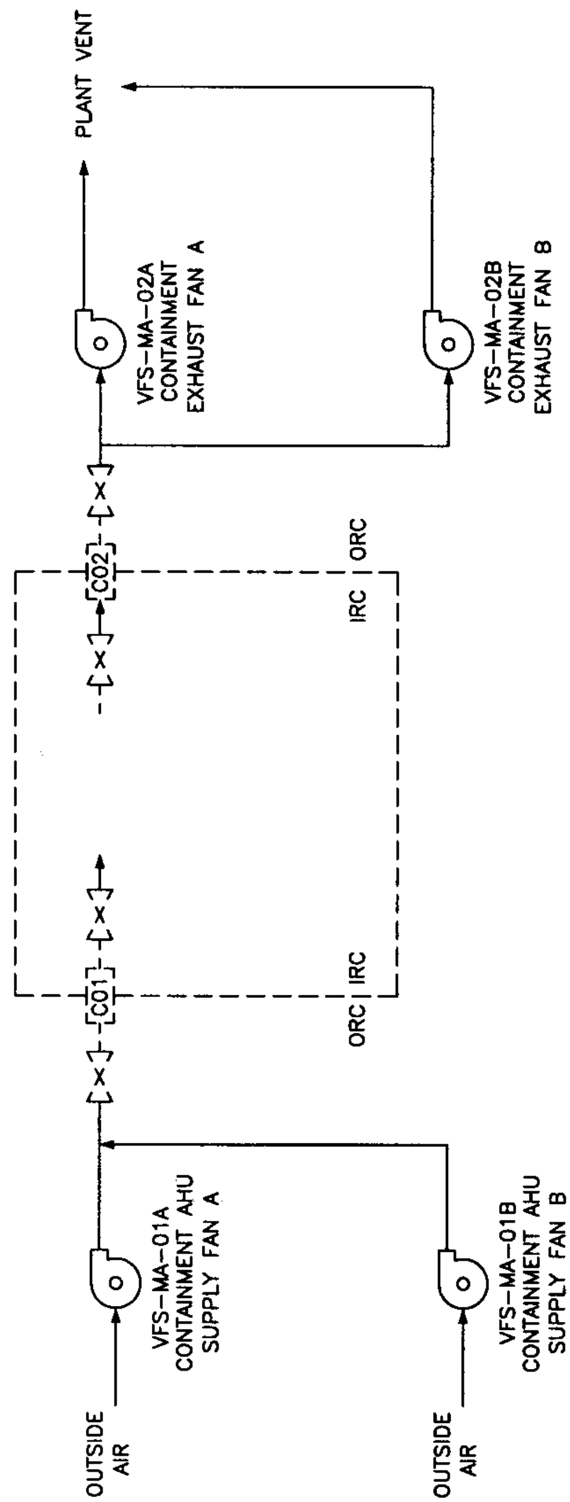


Figure 2.7.6-1
Containment Air Filtration System

2.7.7 Containment Recirculation Cooling System

Design Description

The containment recirculation cooling system (VCS) controls the containment air temperature and humidity during normal operation, refueling and shutdown.

The locations of the VCS are as shown in Table 2.7.7-3.

1. The functional arrangement of the VCS is as described in the Design Description of this Section 2.7.7.
2. Displays of the parameters identified in Table 2.7.7-1 can be retrieved in the main control room (MCR).

Table 2.7.7-1		
Equipment Name	**Tag No.**	**Display**
Containment Temperature Channel	VCS-061	Yes
Containment Fan Cooler Fan	VCS-MA-01A VCS-MA-01C VCS-MA-01B VCS-MA-01D	Yes (Run Status) Yes (Run Status) Yes (Run Status) Yes (Run Status)

Note: Dash (-) indicates not applicable.

Table 2.7.7-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
731	2.7.07.01	1. The functional arrangement of the VCS is as described in the Design Description of this Section 2.7.7.	Inspection of the as-built system will be performed.	The as-built VCS conforms with the functional arrangement described in the Design Description of this Section 2.7.7.
732	2.7.07.02	2. Displays of the parameters identified in Table 2.7.7-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the parameters in the MCR.	The displays identified in Table 2.7.7-1 are retrieved in the MCR.

Table 2.7.7-3		
Component Name	**Tag No.**	**Component Location**
Reactor Containment Recirculation Fan Coil Unit Assembly A	VCS-MS-01A	Containment
Reactor Containment Recirculation Fan Coil Unit Assembly B	VCS-MS-01B	Containment

2.7.8 Radwaste Building HVAC System

No ITAAC for this system.

2.7.9 Turbine Island Building Ventilation System

No entry for this system.

2.7.10 Health Physics and Hot Machine Shop HVAC System

No ITAAC for this system.

2.7.11 Hot Water Heating System

No entry for this system.

3.0 Non-System Based Design Descriptions and ITAAC

3.1 Emergency Response Facilities

Design Description

The technical support center (TSC) is a facility from which management and technical support is provided to main control room (MCR) personnel during emergency conditions. The operations support center (OSC) provides an assembly area where operations support personnel report in an emergency. The control support area (CSA) is an area nearby the main control room from which support can be provided to the main control room.

1. The TSC has floor space of at least 75 ft^2 per person for a minimum of 25 persons.
2. The TSC has voice communication equipment for communication with the MCR, emergency operations facility, OSC, and the U.S. Nuclear Regulatory Commission (NRC).
3. The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, Data Display and Processing System (DDS), with a "Yes" in the "Display" column, can be retrieved in the TSC.
4. The OSC has voice communication equipment for communication with the MCR and TSC.
5. The TSC and OSC are in different locations.
6. The CSA provides a habitable workspace environment.

Table 3.1-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
733	3.1.00.01	1. The TSC has floor space of at least 75 ft^2 per person for a minimum of 25 persons.	An inspection will be performed of the TSC floor space.	The TSC has at least 1875 ft^2 of floor space.
734	3.1.00.02	2. The TSC has voice communication equipment for communication with the MCR, emergency operations facility, OSC, and the NRC.	An inspection and test will be performed of the TSC voice communication equipment.	Communications equipment is installed, and voice transmission and reception are accomplished.
735	3.1.00.03	3. The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, DDS, with a "Yes" in the "Display" column, can be retrieved in the TSC.	An inspection will be performed for retrievability of the plant parameters in the TSC.	The plant parameters listed in Table 2.5.4-1, minimum inventory table, in subsection 2.5.4, DDS, with a "Yes" in the "Display" column, can be retrieved in the TSC.
736	3.1.00.04	4. The OSC has voice communication equipment for communication with the MCR and TSC.	Inspection will be performed of the OSC voice communication equipment.	Communications equipment is installed, and voice transmission and reception are accomplished.
737	3.1.00.05	5. The TSC and OSC are in different locations.	An inspection will be performed of the location of the TSC and OSC.	The TSC and OSC are in different locations.
738	3.1.00.06	6. The CSA provides a habitable workspace environment.	See ITAAC Table 2.7.1-4, items 1, 8.a), 8.c), 12, and 13, Nuclear Island Nonradioactive Ventilation System.	See ITAAC Table 2.7.1-4, items 1, 8.a), 8.c), 12, and 13, Nuclear Island Nonradioactive Ventilation System.

3.2 Human Factors Engineering

Design Description

The AP1000 human-system interface (HSI) will be developed and implemented based upon a human factors engineering (HFE) program. Figure 3.2-1 illustrates the HFE program elements. The HSI scope includes the design of the operation and control centers system (OCS) and each of the HSI resources. For the purposes of the HFE program, the OCS includes the main control room (MCR), the remote shutdown workstation (RSW), the local control stations, and the associated workstations for each of these centers. The HSI resources include the wall panel information system, alarm system, plant information system (nonsafety-related displays), qualified safety-related displays, and soft and dedicated controls. Minimum inventories of controls, displays, and visual alerts are specified as part of the HSI for the MCR and the RSW.

The MCR provides a facility and resources for the safe control and operation of the plant. The MCR includes a minimum inventory of displays, visual alerts and fixed-position controls. Refer to item 8.a and Table 2.5.2-5 of subsection 2.5.2 for this minimum inventory.

The remote shutdown room (RSR) provides a facility and resources to establish and maintain safe shutdown conditions for the plant from a location outside of the MCR. The RSW includes a minimum inventory of displays, controls, and visual alerts. Refer to item 2 and Table 2.5.4-1 of subsection 2.5.4 for this minimum inventory. As stated in item 8.b of subsection 2.5.2, the protection and safety monitoring system (PMS) provides for the transfer of control capability from the MCR to the RSW.

The mission of local control stations is to provide the resources, outside of the MCR, for operations personnel to perform monitoring and control activities.

Implementation of the HFE program includes activity 1 below. The MCR includes design features specified by items 2 through 4 below. The RSW includes the design features specified by items 5 through 8 below. Local control stations include the design feature of item 9.

1. The HFE program verification and validation implementation plans are developed in accordance with the programmatic level description of the AP1000 human factors verification and validation plan. The implementation plans establish the methods for conducting evaluations of the integrated HSI design. The development of the HFE verification and validation plans are complete. The following documents were developed:
   a) HSI task support verification – APP-OCS-GEH-220, "AP1000 Human Factors Engineering Task Support Verification Plan," Westinghouse Electric Company LLC
   b) HFE design verification – APP-OCS-GEH-120, "AP1000 Human Factors Engineering Design Verification Plan," Westinghouse Electric Company LLC
   c) Integrated system validation – APP-OCS-GEH-320, "AP1000 Human Factors Engineering Integrated System Validation Plan," Westinghouse Electric Company LLC
   d) Issue resolution verification – APP-OCS-GEH-420, "AP1000 Human Factors Engineering Discrepancy Resolution Process," Westinghouse Electric Company LLC
   e) Plant HFE/HSI (as designed at the time of plant startup) verification – APP-OCS-GEH-520, "AP1000 Plant Startup Human Factors Engineering Design Verification Plan," Westinghouse Electric Company LLC
2. The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.
3. The MCR provides a suitable workspace environment for use by MCR operators.
4. The HSI resources available to the MCR operators include the alarm system, plant information system (nonsafety-related displays), wall panel information system, nonsafety-related controls (soft and dedicated), and computerized procedure system.
5. The RSW includes reactor operator workstation(s) from which licensed operators perform remote shutdown operations.

6. The RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.
7. The HSI resources available at the RSW include the alarm system displays, the plant information system, and the controls.
8. The RSW and the available HSI permit execution of tasks by licensed operators to establish and maintain safe shutdown.
9. The capability to access displays and controls is provided (controls as assigned by the MCR operators) for local control and monitoring from selected locations throughout the plant.

Table 3.2.-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
739	3.2.00.01a	1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: a) HSI Task support verification	a) An evaluation of the implementation of the HSI task support verification will be performed.	a) A report exists and concludes that: Task support verification was conducted in conformance with the implementation plan and includes verification that the information and controls provided by the HSI match the display and control requirements generated by the function-based task analyses and the operational sequence analyses.
740	3.2.00.01b	1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: b) HFE design verification	b) An evaluation of the implementation of the HFE design verification will be performed.	b) A report exists and concludes that: HFE design verification was conducted in conformance with the implementation plan and includes verification that the HSI design is consistent with the AP1000 specific design guidelines developed for each HSI resource.

Table 3.2.-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
741	3.2.00.01c.i	1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: c) Integrated system validation	c) (i) An evaluation of the implementation of the integrated system validation will be performed.	c) (i) A report exists and concludes that: The test scenarios listed in the implementation plan for integrated system validation were executed in conformance with the plan and noted human deficiencies were addressed.
742	3.2.00.01c.ii	1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: c) Integrated system validation	c) (ii) Tests and analyses of the following plant evolutions and transients, using a facility that physically represents the MCR configuration and dynamically represents the MCR HSI and the operating characteristics and responses of the AP1000 design, will be performed: – Normal plant heatup and startup to 100% power – Normal plant shutdown and cooldown to cold shutdown – Transients: reactor trip and turbine trip – Accidents: - Small-break LOCA - Large-break LOCA - Steam line break - Feedwater line break - Steam generator tube rupture	c) (ii) A report exists and concludes that: The test and analysis results demonstrate that the MCR operators can perform the following: – Heat up and start up the plant to 100% power – Shut down and cool down the plant to cold shutdown – Bring the plant to safe shutdown following the specified transients – Bring the plant to a safe, stable state following the specified accidents
743	3.2.00.01d	1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: d) Issue resolution verification	d) An evaluation of the implementation of the HFE design issue resolution verification will be performed.	d) A report exists and concludes that: HFE design issue resolution verification was conducted in conformance with the implementation plan and includes verification that human factors issues documented in the design issues tracking system have been addressed in the final design.

Table 3.2.-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
744	3.2.00.01e	1. The HFE verification and validation program is performed in accordance with the HFE verification and validation implementation plan and includes the following activities: e) Plant HFE/HSI (as designed at the time of plant startup) verification	e) An evaluation of the implementation of the plant HFE/HSI (as designed at the time of plant startup) verification will be performed.	e) A report exists and concludes that: The plant HFE/HSI, as designed at the time of plant startup, is consistent with the HFE/HSI verified in 1.a) through 1.d).
745	3.2.00.02	2. The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.	An inspection of the MCR workstations and control panels will be performed.	The MCR includes reactor operator workstations, supervisor workstation(s), safety-related displays, and safety-related controls.
746	3.2.00.03.i	3. The MCR provides a suitable workspace environment for use by the MCR operators.	i) See subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System.	i) See subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System.
747	3.2.00.03.ii	3. The MCR provides a suitable workspace environment for use by the MCR operators.	ii) See subsection 2.2.5, MCR Emergency Habitability System.	ii) See subsection 2.2.5, MCR Emergency Habitability System.
748	3.2.00.03.iii	3. The MCR provides a suitable workspace environment for use by the MCR operators.	iii) See subsection 2.6.3, Class 1E dc and UPS System.	iii) See subsection 2.6.3, Class 1E dc and UPS system.
749	3.2.00.03.iv	3. The MCR provides a suitable workspace environment for use by the MCR operators.	iv) See subsection 2.6.5, Lighting System.	iv) See subsection 2.6.5, Lighting System.
750	3.2.00.03.v	3. The MCR provides a suitable workspace environment for use by the MCR operators.	v) See subsection 2.3.19, Communication System.	v) See subsection 2.3.19, Communication System.
751	3.2.00.04	4. The HSI resources available to the MCR operators include the alarm system, plant information system (nonsafety-related displays), wall panel information system, nonsafety-related controls (soft and dedicated), and computerized procedure system.	An inspection of the HSI resources available in the MCR for the MCR operators will be performed.	The HSI (at the time of plant startup) includes an alarm system, plant information system (nonsafety-related displays), wall panel information system, nonsafety-related controls (soft and dedicated), and computerized procedure system.
752	3.2.00.05	5. The RSW includes reactor operator workstation(s) from which licensed operators perform remote shutdown operations.	An inspection of the RSW will be performed.	The RSW includes reactor operator workstation(s).

Table 3.2.-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
753	3.2.00.06.i	6. The RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.	i) See subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System.	i) See subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System.
754	3.2.00.06.ii	6. The RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.	ii) See subsection 2.6.5, Lighting System.	ii) See subsection 2.6.5, Lighting System.
755	3.2.00.06.iii	6. The RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.	iii) See subsection 2.3.19, Communication System.	iii) See subsection 2.3.19, Communication System.
756	3.2.00.07	7. The HSI resources available at the RSW include the alarm system displays, the plant information system, and the controls.	An inspection of the HSI resources available at the RSW will be performed.	The as-built HSI at the RSW includes the alarm system displays, the plant information system, and the controls.
757	3.2.00.08	8. The RSW and the available HSI permit execution of tasks by licensed operators to establish and maintain safe shutdown.	Test and analysis, using a workstation that physically represents the RSW and dynamically represents the RSW HSI and the operating characteristics and responses of the AP1000, will be performed.	A report exists and concludes that the test and analysis results demonstrate that licensed operators can achieve and maintain safe shutdown conditions from the RSW.
758	3.2.00.09	9. The capability to access displays and controls is provided (controls as assigned by the MCR operators) for local control and monitoring from selected locations throughout the plant.	An inspection of the local control and monitoring capability is provided.	The capability for local control and monitoring from selected locations throughout the plant exists.

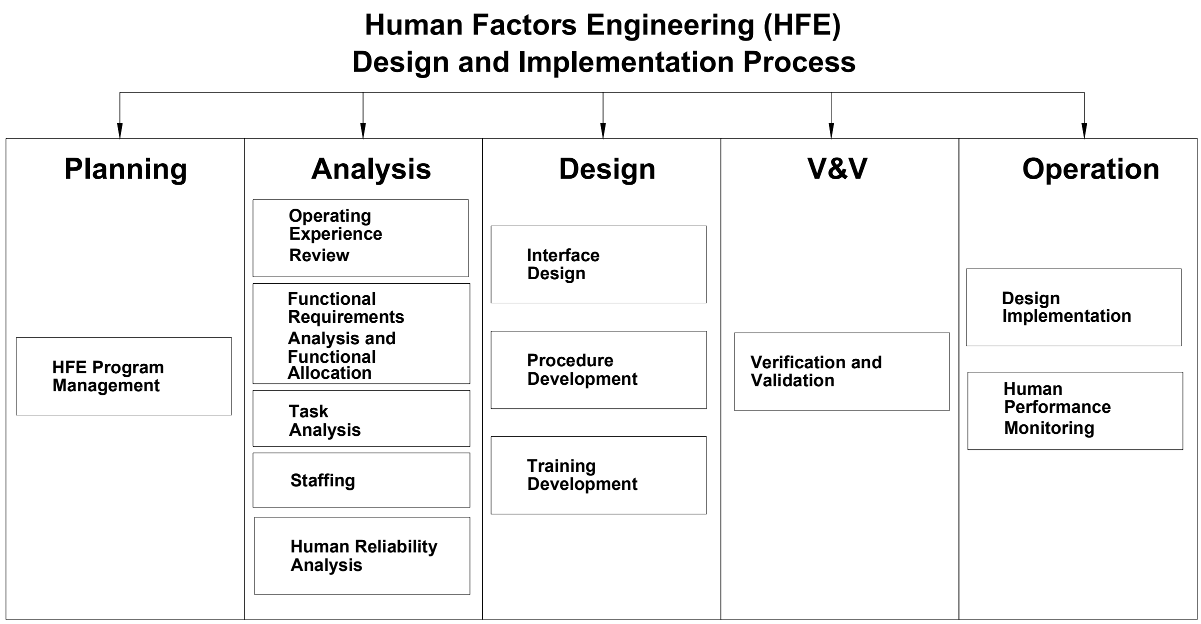


Figure 3.2-1
Human Factors Engineering (HFE)
Design and Implementation Process

3.3 Buildings

Design Description

The nuclear island structures include the containment (the steel containment vessel and the containment internal structure) and the shield and auxiliary buildings. The containment, shield and auxiliary buildings are structurally integrated on a common basemat which is embedded below the finished plant grade level. The containment vessel is a cylindrical welded steel vessel with elliptical upper and lower heads, supported by embedding a lower segment between the containment internal structures concrete and the basemat concrete. The containment internal structure is reinforced concrete with structural modules used for some walls and floors. The shield building cylinder is a composite steel and concrete (SC) structure except for the portion surrounded by the auxiliary building, which is reinforced concrete (RC). The shield building, in conjunction with the internal structures of the containment building, provides shielding for the reactor coolant system and the other radioactive systems and components housed in the containment. The shield building roof is a reinforced concrete structure containing an integral, steel lined passive containment cooling water storage tank. The auxiliary building is reinforced concrete and houses the safety-related mechanical and electrical equipment located outside the containment and shield buildings.

The portion of the annex building adjacent to the nuclear island is a structural steel and reinforced concrete seismic Category II structure and houses the control support area, non-1E electrical equipment, and hot machine shop.

The radwaste building is a steel framed structure and houses the low level waste processing and storage.

The turbine building is a non-safety related structure that houses the main turbine generator and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the turbine building. The turbine building is located on a separate foundation. The turbine building structure is adjacent to the auxiliary building to the south and the annex building to the south and east. The turbine building consists of two separate superstructures, the first bay and the main area, both supported on a common reinforced concrete basemat. The first bay, next to the auxiliary building, consists of a combination of reinforced concrete walls and steel framing with reinforced concrete and steel grated floors. It is classified as a seismic Category II structure due to its immediate proximity to the auxiliary building. The main area of the turbine building, immediately to the north of the first bay, is a steel framed building with reinforced concrete and steel grated floors. It is classified as a non-seismic structure. The non-seismic portion of the turbine building is designed with a mix of concentrically and eccentrically braced framing.

The diesel generator building is a non-safety related structure that houses the two standby diesel engine powered generators and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the diesel generator building. The diesel generator building is located on a separate foundation at a distance from the nuclear island structures.

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required. The component locations of the PGS are located in the yard areas.

1. The physical arrangement of the nuclear island structures, the annex building, and the turbine building is as described in the Design Description of this Section 3.3, and as shown on Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building and the diesel generator building is as described in the Design Description of this Section 3.3.

2. a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads, as specified in the Design Description, without loss of structural integrity and the safety-related functions. The design bases loads are those loads associated with:

      - Normal plant operation (including dead loads, live loads, lateral earth pressure loads, and equipment loads, including hydrodynamic loads, temperature and equipment vibration);
      - External events (including rain, snow, flood, tornado, tornado generated missiles and earthquake); and
      - Internal events (including flood, pipe rupture, equipment failure, and equipment failure generated missiles).

   b) Site grade level is located relative to floor elevation 100′-0″ per Table 3.3-5. Floor elevation 100′-0″ is defined as the elevation of the floor at design plant grade.

   c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.[1]

   d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.

   e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.

   f) The key dimensions of the nuclear island structures are as defined on Table 3.3-5.

   g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.

   h) The containment free volume below elevation 108′ provides containment floodup during a postulated loss-of-coolant accident.

3. Walls and floors of the nuclear island structures as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

4. a) Walls and floors of the annex building as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

   b) The walls on the outside of the waste accumulation room in the radwaste building provide shielding from accumulated waste.

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

c) Deleted.

5. a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.

   b) The boundaries between mechanical equipment rooms and the electrical and instrumentation and control (I&C) equipment rooms of the auxiliary building as identified in Table 3.3-2 are designed to prevent flooding of rooms that contain safety-related equipment up to the maximum flood level for each room defined in Table 3.3-2.

   c) The boundaries between the following rooms, which contain safety-related equipment – passive core cooling system (PXS) valve/accumulator room A (11206), PXS valve/accumulator room B (11207), and chemical and volume system (CVS) room (11209) – are designed to prevent flooding between these rooms.

6. a) The radiologically controlled area of the auxiliary building between floor elevations 66′-6″ and 82′-6″ contains adequate volume to contain the liquid volume of faulted liquid radwaste system (WLS) storage tanks. The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66′-6″ and 82′-6″ exceeds the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).

   b) The radwaste building waste accumulation room has a volume greater than or equal to 1417 cubic feet.

7. a) Class 1E electrical cables, fiber optic cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.

   b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.

   c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.

   d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

   e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

8. Systems, structures, and components identified as essential targets are protected from the dynamic and environmental effects of postulated pipe ruptures.

9. The reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.

10. The shield building roof and the passive containment cooling system (PCS) storage tank support and retain the PCS water. The passive containment cooling system tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided over the tank boundary liner welds.

11. Deleted.
12. The extended turbine generator axis intersects the shield building.
13. Separation is provided between the structural elements of the turbine, annex, and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in a safe shutdown earthquake without impact between structural elements of the buildings.
14. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.
15. Deleted.
16. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.
17. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.
18. Deleted.

Table 3.3-1 Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)				
Wall or Section Description	**Column Lines(7)**	**Floor Elevation or Elevation Range(7)(8)**	**Concrete Thickness(2)(3)(4)(5)(9)**	**Applicable Radiation Shielding Wall (Yes/No)**
Containment Building Internal Structure				
Shield Wall between Reactor Vessel Cavity and RCDT Room	E-W wall parallel with column line 7 (Inside face is 3'-0" north of column line 7. Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 71'-6" to 83'-0"	3'-0"(10)	Yes
West Reactor Vessel Cavity Wall	N-S wall parallel with column line N (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 83'-0" to 98'-0"	7'-6"(11)	Yes
North Reactor Vessel Cavity Wall	E-W wall parallel with column line 7 (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 83'-0" to 98'-0"	9'-0"(11)	Yes
East Reactor Vessel Cavity Wall	N-S wall parallel with column line N (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 83'-0" to 98'-0"	7'-6"(11)	Yes
West Refueling Cavity Wall	N-S wall parallel with column line N	From 98'-0" to 135'-3"	4'-0"	Yes
North Refueling Cavity Wall	E-W wall parallel with column line 7	From 98'-0" to 135'-3"	4'-0"	Yes
East Refueling Cavity Wall	N-S wall parallel with column line N	From 98'-0" to 135'-3"	4'-0"	Yes
South Refueling Cavity Wall	E-W wall parallel with column line 7	From 98'-0" to 135'-3"	4'-0"	Yes
South wall of west steam generator compartment	Not Applicable	From 103'-0" to 153'-0"	2'-6"	Yes
West wall of west steam generator compartment	N-S wall parallel with column line N	From 103'-0" to 153'-0"	2'-6"	Yes
North wall of west steam generator compartment	Not Applicable	From 103'-0" to 153'-0"	2'-6"	Yes

1. The column lines and floor elevations are identified and included on Figures 3.3-1 through 3.3-13.
2. These wall (and floor) thicknesses have a construction tolerance of ± 1 inch, except as noted and for exterior walls below grade where the tolerance is +12 inches, - 1 inch. These tolerances are not applicable to the nuclear island basemat.
3. For walls that are part of structural modules, the concrete thickness also includes the steel face plates. Where faceplates with a nominal thickness of 0.5 inches are used in the construction of the wall modules, the wall thicknesses in this column apply. Where faceplates thicker than the nominal 0.5 inches are used in the construction of the structural wall modules, the wall thicknesses in the area of the thicker faceplates are greater than indicated in this column by the amount of faceplate thickness increase over the nominal 0.5 inches. Overlay plates are not considered part of the faceplates, and thus are not considered in the wall thicknesses identified in this column.
4. For floors with steel surface plates, the concrete thickness also includes the plate thickness.
5. Where a wall (or a floor) has openings, the concrete thickness does not apply at the opening.
6. The elevation ranges for the shield building items are rounded to the nearest inch.

7. The Wall or Section Description, Column Line information, and Floor Elevation or Elevation Ranges are provided as reference points to define the general location. The concrete thickness of an item intersecting other walls, roofs or floors at a designated location (e.g., column line) is not intended to be measured to the stated column line, but only to the point where the intersection occurs.
8. Where applicable, the upper wall portions extend to their associated roofs, which may vary in elevation, e.g., sloped roofs.
9. From one wall/floor section to another, the concrete thickness transitions from one thickness to another, consistent with the configurations in Figures 3.3-1 through 3.3-14.
10. This wall thickness has a tolerance of ±1-1/4 inch.
11. These wall thicknesses have a tolerance of ±1-5/8 inch.
12. These wall thicknesses have a tolerance of -1 inch and +4 inch for a length of 24 inches at the interface of these reinforced concrete walls to structural module connections.

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)	Applicable Radiation Shielding Wall (Yes/No)
South wall of pressurizer compartment	Not Applicable	From 103'-0" to 153'-6"	2'-6"	Yes
West wall of pressurizer compartment	N-S wall parallel with column line N	From 107'-2" to 160'-0"	2'-6"	Yes
North wall of pressurizer compartment	E-W wall parallel with column line 7	From 107'-2" to 160'-0"	2'-6"	Yes
East wall of pressurizer compartment	N-S wall parallel with column line N	From 118'-6" to 160'-0"	2'-6"	Yes
North-east wall of in-containment refueling water storage tank	Parallel to column line N	From 103'-0" to 135'-3"	2'-6"	No
West wall of in-containment refueling water storage tank	Not applicable	From 103'-0" to 135'-3"	5/8" steel plate with stiffeners	No
South wall of east steam generator compartment	Not Applicable	From 87'-6" to 153'-0"	2'-6"	Yes
East wall of east steam generator compartment	N-S wall parallel with column line N	From 94'-0" to 153'-0"	2'-6"	Yes
North wall of east steam generator compartment	Not Applicable	From 87'-6" to 153'-0" with a 158'-0" portion	2'-6"	Yes
Shield Building(6)				
Shield Building Cylinder	Not Applicable	From 100'-0" to 248-6"	3'-0" (including 3/4 inch thick min. steel plate liner on each face on portion not protected by auxiliary building)	Yes
Air Inlet	Not Applicable	From 248'-6" to 251'-6"	3’-0" (including 3/4 inch thick min. steel plate liner on each face)	Yes
		From 251'-6" to 254'-6"	3'-0" to 4'-6" (including 1 inch thick steel plate liner on each face)	Yes
		From 254'-6" to 266'-4"	4'-6" (including 1 inch thick min. steel plate liner on each face)	Yes
Tension Ring	Not Applicable	From 266'-4" to 271'-6" (at top of plate)	4'-6" (including 1-1/2 inch thick steel plate liner on	Yes

Table 3.3-1 (cont.) Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[(1)]				
Wall or Section Description	**Column Lines[(7)]**	**Floor Elevation or Elevation Range[(7)(8)]**	**Concrete Thickness[(2)(3)(4)(5)(9)]**	**Applicable Radiation Shielding Wall (Yes/No)**
			each face)	
Conical Roof	Not Applicable	From 271'-6" to 293'-9"	3'-0" (including 1/2 inch thick min. steel plate liner on bottom face), outside of PCS tank exterior wall	Yes
PCS Tank External Cylindrical Wall	Not Applicable	From 293'-9" to 328'-9"	2'-0"	Yes
PCS Tank Internal Cylindrical Wall	Not Applicable	From 309'-4" to 329'-0"	1'-6"	Yes
PCS Tank Roof	Not Applicable	328'-9" (Lowest) 329'-0" (Highest)	1'-3"	No
Nuclear Island Basemat	Below shield building	From 60'-6" to containment vessel or 82'-6"	6'-0" to 22'-0" (varies)	No
Auxiliary Building Walls/Floors Radiologically Controlled				
Column Line 1 wall	From I to N	From 66'-6" to 100'-0"	3'-0"	No
Column Line 1 wall	From I to 5'-6" east of L-2	From 100'-0" to 180'-0"	2'-3"	Yes
Column Line 1 wall	From 5'-6" east of L-2 to N	From 100'-0" to 125'-0"	3'-0"	Yes
Column Line 1 wall	From 5'-6" east of L-2 to N	From 125'-0" to 180'-0"	2'-3"	Yes
Column Line 2 wall	From I to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From K-2 to L-2	From 66'-6" to 135'-3"	5'-0"	Yes
Column Line 2 wall	From L-2 to N	From 82'-6" to 135'-3"	2'-6"	Yes
Column Line 2 wall	From I to J-1	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line 3 wall	From J-1 to J-2	From 66'-6" to 82'-6"	2'-6"	Yes
Column Line 3 wall	From J-1 to J-2	From 100'-0" to 135'-3"	2'-6"	Yes
Column Line 3 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 3 wall	From K-2 to L-2	From 66'-6" to 92'-8 1/2"	2'-6"	Yes
Column Line 4 wall	From I to J-1	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 66'-6" to 92'-6"	2'-6"	Yes
Column Line 4 wall	From J-1 to J-2	From 107'-2" to 135'-3"	2'-6"	Yes
Column Line 4 wall	From J-2 to K-2	From 66'-6" to 135'-3"	2'-6"	Yes

Table 3.3-1 (cont.) Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]				
Wall or Section Description	**Column Lines[7]**	**Floor Elevation or Elevation Range[7][8]**	**Concrete Thickness[2][3][4][5][9]**	**Applicable Radiation Shielding Wall (Yes/No)**
Column Line 4 wall	From I to intersection with shield building wall	From 135'-3" to 180'-0"	2'-0"	Yes
Column Line 5 wall	From I to shield building; with opening east of J-1 (below 107'-2" floor).	From 66'-6" to 160'-6"	2'-0"	Yes
Wall, 17'-3" north of Column Line 7	From I to 8' east of J-1	From 66'-6" to 82'-6"	2'-0"	Yes
Wall, 10'-6" south of Column Line 7.3	From I to 5'-6"east of J-1	From 66'-6" to 82'-6"	2'-0"	Yes
Wall, 10'-6" south of Column Line 7.3	From I to just east of J	From 82'-6" to 100'-0"	2'-0"	Yes
Column Line I wall	From 1 to 4	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 3 to 4	From 100'-0" to 107'-2"	3'-0"	Yes
Column Line I wall	From 4 to 16'-0" south of 5	From 66'-6" to 107'-2"	3'-0"	No
Column Line I wall	From 16'-0" south of 5 to 5	From 66'-6" to 105'-0"	3'-0"	No
Column Line I wall	From 5 to 7.3	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 1 to 3	From 100'-0" to roof	2'-0"	Yes
Column Line I wall	From 3 to 4	From 107'-2" to roof	2'-0"	Yes
Column Line I wall	From 4 to 16'-0" south of 5	From 107'-2" to roof	2'-0"	No
Column Line I wall	From 16'-0" south of 5 to 5	From 105'-0" to roof	2'-0"	No
Column Line J-1 wall	From 1 to 2	From 82'-6" to 100'-0"	2'-0"	Yes
Column Line J-1 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-1 wall	From 2 to 4	From 135'-3" to 153'-0"	2'-0"	Yes
Column Line J-1 wall	From 4 to shield building	From 66'-6" to 107'-2"	2'-0"[12]	Yes
Column Line J-2 wall	From 2 to 4	From 66'-6" to 135'-3"	2'-6"	Yes
Column Line J-2 wall	From 4 to intersection with shield building wall	From 66'-6" to 135'-3"	2'-0"[12]	Yes
Column Line K-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-9"	Yes
Column Line L-2 wall	From 2 to 4	From 66'-6" to 135'-3"	4'-0"	Yes
Column Line N wall	From 1 to 2	From 66'-6" to 100'-0"	3'-0"	No
Column Line N wall	From 1 to 12'-9" north of 1	From 100'-0" to 125'-0"	3'-9"	Yes
Column Line N wall	From 1 to 12'-9" north of 1	From 125'-0" to 135'-3"	2'-0"	Yes

Table 3.3-1 (cont.)
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)

Wall or Section Description	Column Lines(7)	Floor Elevation or Elevation Range(7)(8)	Concrete Thickness(2)(3)(4)(5)(9)	Applicable Radiation Shielding Wall (Yes/No)
Column Line N wall	From 12'-9" north of 1 to 2	From 100'-0" to 118'-2 1/2"	3'-0"	Yes
Column Line N wall	From 12'-9" north of 1 to 2	From 118'-2 1/2" to 135'-3"	2'-0"	Yes
Column Line N wall	From 2 to 4 (or to shield building)	From 66'-6" to 98'-1"	5'-6"	No
Column Line N wall	From 2 to 4 (or to shield building)	From 98'-1" to 135'-3"	5'-6"	Yes
Column Line N wall	From 1 to 4 (or to shield building)	From 135'-3" to 180'-0"	2'-0"	Yes
Labyrinth Wall between Col. Line 3 and 4 and J-1 to 7'-3" from J-2	Not Applicable	From 82'-6" to 92'-6"	2'-6"	Yes
N-S Shield Wall (low wall)	5'-7" west of column line K-2 extending 16'-0" from column line 1 north	From 100'-0" to 110'-0"	2'-6"	Yes
N-S Shield Wall	2'-9" east of column line L-2 extending 12'-9" from column line 1 north	From 100'-0" to 125'-0"	2'-9"	Yes
E-W Shield Wall	Between 1 and 2 extending 16'-3" from column line N east	From 100'-0" to 125'-0"	2'-9"	Yes
Auxiliary Area Basemat	From 1-7.3 and I-N, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Floor	From 1 to 2 and I to N	82'-6"	2'-0"	Yes
Floor	From 2 to 4 and J-1 to J-2	82'-6"	2'-0"	Yes
Floor	From 4 north to the shield building ending 17'-4" south of column line 5 and J-1 to J-2	82'-6"	0'-9"	Yes
Pipe Chase Floor	From 2 north to the shield building ending 17'-4" south of column line 5 and J-1 to J-2	92'-6"	2'-0"	Yes
Floor	From 2 to 3 and J-2 to K-2	90'-3"	3'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	92'-6"	2'-0"	Yes
Floor	From 4 to 7.3 and I to J-1	82'-6"	2'-0"	Yes
Floor	From 1 to 2 and I to N	100'-0"	3'-0"	Yes
Floor	From 2 to 4 and K-2 to L-2	92'-8 1/2"	3'-2 1/2"	Yes
Floor	From I to J-2 and 4 to shield building and vertical wall 17'-0" south of column line 5	107'-2"	2'-0"	Yes

Table 3.3-1 (cont.) Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building(1)				
Wall or Section Description	**Column Lines(7)**	**Floor Elevation or Elevation Range(7)(8)**	**Concrete Thickness(2)(3)(4)(5)(9)**	**Applicable Radiation Shielding Wall (Yes/No)**
Floor	From I to shield building wall and from intersecting vertical wall before column line 5 to column line 5	105'-0"	0'-9"	Yes
Floor	From column line 1 to 10'-0" north of column line 1 and from 2'-9" east of column line L-2 to N	125'-0"	3'-0"	Yes
Floor	From 12'-9" north of column lines 1 to 2 and from 2'-9" east of column lines L-2 to N	118'-2 1/2"	2'-0"	Yes
Floor	From 3 to 4 and J-2 to K-2	117'-6"	2'-0"	Yes
Floor	From 2 to 4 and I to J-1	153'-0"	0'-9"	Yes
Roof	From 1 to 4 and I to N	180'-0" to 180'-9"	1'-3"	Yes
Floor	From 4 to 16'-0" south of column line 5 and from I to intersection with shield building wall	135'-3"	0'-9"	Yes
Floor	From 16'-0" south of column line 5 to column line 5 and from I to intersection with shield building wall	133'-0"	0'-9"	Yes
Auxiliary Building Walls/Floors Non-Radiologically Controlled				
Column Line 11 wall	From I to Q	From 66'-6" to 100'-0"	3'-0"	No
Column Line 11 wall	From I to Q	From 100'-0" to 117'-6"	2'-0"	Yes
Column Line 11 wall	From I to L	From 117'-6" to roof	2'-0"	Yes
Column Line 11 wall	From L to M	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From M to P	From 117'-6" to 135'-3"	2'-0"	Yes
Column Line 11 wall	From P to Q	From 117'-6" to 135'-3"	4'-0"	Yes
Column Line 11 wall	From L to Q	From 135'-3" to roof	2'-0"	Yes
Column Line 7.3 wall	From I to shield building	From 66'-6" to 100'-0"	3'-0"	Yes
Column Line 7.3 wall	From I to shield building	From 100'-0" to roof	2'-0"	No
Column Line I wall	From 7.3 to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line I wall	From 5 to 11	From 100'-0" to roof	2'-0"	No
Column Line J wall	From 7.3 to 11	From 66'-6" to 117'-6"	2'-0"	No
Column Line K wall	From 7.3 to 11	From 66'-6" to 135'-3"	2'-0"	Yes

Table 3.3-1 (cont.) Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]				
Wall or Section Description	**Column Lines[7]**	**Floor Elevation or Elevation Range[7][8]**	**Concrete Thickness[2][3][4][5][9]**	**Applicable Radiation Shielding Wall (Yes/No)**
Column Line L wall	From shield building wall to 11	From 66'-6" to roof	2'-0"	Yes
Column Line M wall	From shield building wall to 11	From 66'-6" to roof	2'-0"	Yes
Column Line P wall	From shield building wall to 11	From 66'-6" to roof	2'-0"	Yes
Column Line Q wall	From shield building wall to 11	From 66'-6" to 100'-0"	3'-0"	No
Column Line Q wall	From shield building wall to 11	From 100'-0" to roof	2'-0"	Yes
Column Line 9.2 wall	From I to J and K to L	From 117'-6" to 135'-3"	2'-0"	Yes
Labyrinth Wall between Column Line 7.3 and 9.2 and J to K	J to K	From 117'-6" to 135'-3"	2'-0"	Yes
Auxiliary Area Basemat	From 7.3-11 and I-Q, excluding shield building	From 60'-6" to 66'-6"	6'-0"	No
Floor	From 5 to 10'-6" south of 7.3 and I to shield building wall	100'-0"	2'-0"	Yes
Floor	From 10'-6" south of 7.3 to 7.3 and I to shield building wall	100'-0"	3'-0"	Yes
Floor	From K to L and shield building wall to column line 10	100'-0"	0'-9"	Yes
Main Control Room Floor	From 9.2 to 11 and I to L	117'-6"	2'-0"	Yes
Floor	Bounded by shield bldg, 7.3, J, 9.2 and L	117'-6"	2'-0"	Yes
Floor	From shield building to 11 and L to Q	117'-6"	2'-0"	Yes
Floor	From 5 to 7.3 and from I to intersection with shield building wall	135'-3"	0'-9"	Yes
Annex Building				
Column line 2 wall	From E to H	From 107'-2" to 135'-3"	19 3/4"	Yes
Column line 4 wall	From E to H	From 107'-2" to 162'-6" & 166'-0"	2'-0"	Yes
N-S Shield Wall between E and F	From 2 to 4	From 107'-2" to 135'-3"	1'-0"	Yes
Column line 4.1 wall	From E to H	From 107'-2" to 135'-3"	2'-0"	Yes
N-S Labyrinth Wall between column line 7.8 and 9 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes
E-W Labyrinth Wall between column line 7.1 and 7.8 and G to H	Not Applicable	From 100'-0" to 112'-0"	2'-0"	Yes

Table 3.3-1 (cont.) Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]				
Wall or Section Description	**Column Lines[7]**	**Floor Elevation or Elevation Range[7][8]**	**Concrete Thickness[2][3][4][5][9]**	**Applicable Radiation Shielding Wall (Yes/No)**
Column Line 9 wall	From E to connecting wall between G and H	From 107'-2" to 117'-6"	2'-0"	Yes
Column Line E wall	From 9 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line 13 wall	From E to I.1	From 100'-0" to 135'-3"	2'-0"	Yes
Column Line I.1 wall	From 11.09 to 13	From 100'-0" to 135'-3"	2'-0"	Yes
Corridor Wall between G and H	From 9 to near 13	From 100'-0" to 117'-6"	1'-6"	Yes
Corridor Wall between G and H	From 9 to 13	From 117'-6" to 135'-3"	1'-6"	Yes
Column Line 9 wall	From E to H	From 117'-6" to 158'-0"	2'-0"	Yes
Floor	From 2 to 4 and E to H	135'-3"	0'-8"	Yes
Floor	From 4 to 4.1 and E to H	135'-3"	1'-0"	Yes
Floor	From 9 to 13 and E to I.1	117'-6"	0'-8"	Yes
Floor	From 9 to 13 and E to I.1	135'-3"	0'-8"	Yes
Containment Filtration Rms A and B (North Wall)	Between column line E to H	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rms A and B (East wall)	Between column line E to F	From 135'-3" to 158'-0"	1'-0"	Yes
Containment Filtration Rm A (West wall)	Between column line G to H	From 135'-3" to 150'-3"	1'-0"	Yes
Containment Filtration Rm A (Floor)	Between column line E to H	135'-3"	1'-0"	Yes
Containment Filtration Rm B (Floor)	Between column line E to H	150'-3"	0'-8"	Yes
Containment Filtration Rm B (West wall)	Between column line G to H	From 150'-3" to 158'-0"	1'-0"	Yes
Turbine Building				
Wall adjacent to Column Line I.2	From Col. Line 11.05 to 11.2	From 100'-0" to 169'-0"	3'-0"	No
Wall along Column Line 11.2	From near I.2 to near Col. Line R	From 100'-0" to 169'-0"	2'-0"	No
Wall adjacent to Column Line R	From Col. Line 11.2 to Col. Line 11.02	From 100'-0" to 169'-0"	3'-0"	No
Wall along Column Line 11.02	From near Col. Line R to near Col. Line Q	From 100'-0" to 169'-0"	2'-0"	No
Wall along Column Line 11.05	From Col. Line K.4 to near Col. Line I.2	From 100'-0" to 169'-0"	2'-0"	No

Table 3.3-2
Nuclear Island Building Room Boundaries
Required to Have Flood Barrier Floors and Walls

Boundary/ Maximum Flood Level (inches)	Between Room Number to Room Number	
	Room with Postulated Flooding Source	Adjacent Room
Floor/36	12306	12211
Floor/3	12303	12203/12207
Floor/3	12313	12203/12207
Floor/1	12300	12201/12202/12207 12203/12204/12205
Floor/3	12312	12212
Wall/36	12306	12305
Floor/1	12401	12301/12302/12303 12312/12313
Wall/1	12401	12411/12412
Floor/36	12404	12304
Floor/4	12405	12305
Floor/36	12406	12306
Wall/36	12404	12401
Wall/1	12421	12452
Floor/3	12501	12401/12411/12412
Floor/3	12555	12421/12423/12422
Wall/36	12156/12158	12111/12112

Table 3.3-3 Class 1E Divisions in Nuclear Island Fire Areas				
	Class 1E Divisions			
Fire Area Number	**A**	**C**	**B**	**D**
Auxiliary Building Radiologically Controlled				
1200 AF 01	Yes	Yes	–	–
1204 AF 01	Yes	–	–	–
Auxiliary Building Non-Radiologically Controlled				
1200 AF 03	–	–	Yes	Yes
1201 AF 02	–	–	Yes	–
1201 AF 03	–	–	–	Yes
1201 AF 04	–	–	Yes	Yes
1201 AF 05	–	–	Yes	Yes
1201 AF 06	–	–	Yes	Yes
1202 AF 03	–	Yes	–	–
1202 AF 04	Yes	–	–	–
1220 AF 01	–	–	Yes	Yes
1220 AF 02	–	–	–	Yes
1230 AF 01	Yes	Yes	–	–
1230 AF 02	–	–	Yes	Yes
1240 AF 01	Yes	Yes	–	–
1242 AF 02	Yes		–	

Note: Dash (–) indicates not applicable.

Table 3.3-4 is not used.

Table 3.3-5 Key Dimensions of Nuclear Island Building Features			
Key Dimension	**Reference Dimension (Figure 3.3-14)**	**Nominal Dimension**	**Tolerance**
Distance between Outside Surface of walls at Column Line I & N when Measured at Column Line 1	X1	91 ft-0 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line 1 to Column Line 7 when Measured at Column Line I	X2	138 ft-0 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line 11 to Column Line 7 when Measured at Column Line I	X3	118 ft-0 in	+3 ft -1 ft
Distance between Outside Surface of walls at Column Line I & Q when Measured at Column Line 11	X4	117 ft-6 in	+3 ft -1 ft
Distance from Outside Surface of wall at Column Line Q to Column Line N when Measured at Column Line 11	X5	29 ft-0 in	+3 ft -1 ft
Distance between Outside Surface of shield building wall to shield building centerline when Measured on West Edge of Shield Building	X6	72 ft-6 in	+3 ft -1 ft
Distance between shield building centerline to Reactor Vessel centerline when Measured along Column Line N in North-South Direction	X7	7 ft-6 in	± 3 in
Distance from Bottom of Containment Sump to Top Surface of Embedded Containment Shell	–	2 ft-8 in	± 3 in
Distance from top of Basemat to Design Plant Grade	–	33 ft-6 in	± 1 ft
Distance of Design Plant Grade (Floor elevation 100'-0") relative to Site Grade	–	0 ft	± 3 ft-6 in
Distance from Design Plant Grade to Top Surface of Shield Building Roof	–	229 ft-0 in	± 1 ft

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
759	3.3.00.01	1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.	An inspection of the nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building will be performed.	The as-built nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building conform with the physical arrangement as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14.
760	3.3.00.02a.i.a	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i.a) A report exists which reconciles deviations during construction and concludes that the as-built containment internal structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
761	3.3.00.02a.i.b	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i.b) A report exists which reconciles deviations during construction and concludes that the as-built shield building structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
762	3.3.00.02a.i.c	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i.c) A report exists which reconciles deviations during construction and concludes that the as-built structures in the non-radiologically controlled area of the auxiliary building, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
763	3.3.00.02a.i.d	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i.d) A report exists which reconciles deviations during construction and concludes that the as-built structures in the radiologically controlled area of the auxiliary building, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
764	3.3.00.02a.ii.a	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii) An inspection of the as-built concrete thickness will be performed.	ii.a) A report exists that concludes that the containment internal structures as-built concrete thicknesses conform to the building sections defined in Table 3.3-1.
765	3.3.00.02a.ii.b	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii) An inspection of the as-built concrete thickness will be performed.	ii.b) A report exists that concludes that the as-built concrete thicknesses of the shield building sections conform to the building sections defined in Table 3.3-1.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
766	3.3.00.02a.ii.c	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii) An inspection of the as-built concrete thickness will be performed.	ii.c) A report exists that concludes that as-built concrete thicknesses of the non-radiologically controlled area of the auxiliary building sections conform to the building sections defined in Table 3.3-1.
767	3.3.00.02a.ii.d	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii) An inspection of the as-built concrete thickness will be performed.	ii.d) A report exists that concludes that the as-built concrete thicknesses of the radiologically controlled area of the auxiliary building sections conform to the building sections defined in Table 3.3-1.
768	3.3.00.02a.ii.e	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii) An inspection of the as-built concrete thickness will be performed.	ii.e) A report exists that concludes that the as-built concrete thicknesses of the annex building sections conform with the building sections defined in Table 3.3-1.
769	3.3.00.02a.ii.f	2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	ii) An inspection of the as-built concrete thickness will be performed.	ii.f) A report exists that concludes that the as-built concrete thicknesses of the turbine building sections conform to the building sections defined in Table 3.3-1.
770	3.3.00.02b	2.b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5.	Inspection of the as-built site grade will be conducted.	Site grade is consistent with design plant grade within the dimension defined on Table 3.3-5.
771	3.3.00.02c	2.c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.[1]	See ITAAC Table 2.2.1-3, Items 2.a, 2.b, 3.a, and 3.b.	See ITAAC Table 2.2.1-3, Items 2.a, 2.b, 3.a, and 3.b.

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
772	3.3.00.02d	2.d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.	See ITAAC Table 2.2.1-3, Items 4.a and 4.b.	See ITAAC Table 2.2.1-3, Items 4.a and 4.b.
773	3.3.00.02e	2.e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.	See ITAAC Table 2.2.1-3, Items 4.a, 4.b, and 7.	See ITAAC Table 2.2.1-3, Items 4.a, 4.b, and 7.
774	3.3.00.02f	2.f) The key dimensions of nuclear island structures are defined on Table 3.3-5.	An inspection will be performed of the as-built configuration of the nuclear island structures.	A report exists and concludes that the key dimensions of the as-built nuclear island structures are consistent with the dimensions defined on Table 3.3-5.
775	3.3.00.02g	2.g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.	The maximum containment vessel inside height from the operating deck is measured and the inner radius below the spring line is measured at two orthogonal radial directions at one elevation.	The containment vessel maximum inside height from the operating deck is 146'-7" (with tolerance of +12", -6"), and the inside diameter is 130 feet nominal (with tolerance of +12", -6").
776	3.3.00.02h	2.h) The free volume in the containment allows for floodup to support long-term core cooling for postulated loss-of-coolant accidents.	An inspection will be performed of the as-built containment structures and equipment. The portions of the containment included in this inspection are the volumes that flood with a loss-of-coolant accident in passive core cooling system valve/equipment room B (11207). The in-containment refueling water storage tank volume is excluded from this inspection.	A report exists and concludes that the floodup volume of this portion of the containment is less than 73,500 ft^3 to an elevation of 108'.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
777	3.3.00.03a	3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	a) A report exists and concludes that the shield walls and floors of the containment internal structures as defined in Table 3.3-1, except for designed openings or penetrations, are consistent with the concrete wall thicknesses provided in Table 3.3-1.
778	3.3.00.03b	3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	b) A report exists and concludes that the shield walls of the shield building structures as defined in Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.
779	3.3.00.03c	3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	c) A report exists and concludes that the shield walls and floors of the non-radiologically controlled area of the auxiliary building as defined in Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.
780	3.3.00.03d	3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	d) A report exists and concludes that the shield walls and floors of the radiologically controlled area of the auxiliary building as defined in Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
781	3.3.00.04a	4.a) Walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built annex building wall and floor thicknesses will be performed.	A report exists and concludes that the shield walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the minimum concrete wall thicknesses provided in Table 3.3-1.
782	3.3.00.04b	4.b) Walls of the waste accumulation room in the radwaste building except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built radwaste building wall thicknesses will be performed.	A report exists and concludes that the shield walls of the waste accumulation room in the radwaste building except for designed openings or penetrations are consistent with the minimum concrete wall thickness of 1'-4", and a minimum concrete wall thickness of 1'-8" near the radwaste bunkers.
783	3.3.00.04c	4.c) Deleted		
784	3.3.00.05a	5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built water barrier will be performed during construction.	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.
785	3.3.00.05b	5.b) The boundaries between rooms identified in Table 3.3-2 of the auxiliary building are designed to prevent flooding of rooms that contain safety-related equipment.	An inspection of the auxiliary building rooms will be performed.	A report exists that confirms floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
786	3.3.00.05c	5.c) The boundaries between the following rooms, which contain safety-related equipment – PXS valve/accumulator room A (11206), PXS valve/accumulator room B (11207), and CVS room (11209) – are designed to prevent flooding between these rooms.	An inspection of the boundaries between the following rooms which contain safety-related equipment – PXS Valve/ Accumulator Room A (11206), PXS Valve/Accumulator Room B (11207), and CVS Room (11209) – will be performed.	A report exists that confirms that flooding of the PXS Valve/ Accumulator Room A (11206), and the PXS Valve/Accumulator Room B (11207) is prevented to a maximum flood level as follows: PXS A 110'-2", PXS B 110'-1"; and of the CVS room (11209) to a maximum flood level of 110'-0".
787	3.3.00.06a	6.a) The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).	An inspection will be performed of the as-built radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" to define volume.	A report exists and concludes that the as-built available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).
788	3.3.00.06b	6.b) The radwaste building waste accumulation room has a volume greater than or equal to 1417 cubic feet.	An inspection of the radwaste building waste accumulation room (50351) is performed.	The volume of the radwaste building waste accumulation room (50351) is greater than or equal to 1417 cubic feet.
789	3.3.00.07aa	7.a) Class 1E electrical cables, communication cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and raceways will be conducted.	a) Class 1E electrical cables, and communication cables inside containment associated with only one division, and raceways are identified by the appropriate color code.
790	3.3.00.07ab	7.a) Class 1E electrical cables, communication cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and raceways will be conducted.	b) Class 1E electrical cables, and communication cables in the non-radiologically controlled area of the auxiliary building associated with only one division, and raceways are identified by the appropriate color code.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
791	3.3.00.07ac	7.a) Class 1E electrical cables, communication cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and raceways will be conducted.	c) Class 1E electrical cables, and communication cables in the radiologically controlled area of the auxiliary building associated with only one division, and raceways are identified by the appropriate color code.
792	3.3.00.07ba	7.b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and raceways will be conducted.	a) Class 1E electrical cables and communication cables inside containment associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
793	3.3.00.07bb	7.b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and raceways will be conducted.	b) Class 1E electrical cables and communication cables in the non-radiologically controlled area of the auxiliary building associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
794	3.3.00.07bc	7.b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and raceways will be conducted.	c) Class 1E electrical cables and communication cables in the radiologically controlled area of the auxiliary building associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
795	3.3.00.07c.i.a	7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	i) Inspections of the as-built Class 1E division electrical cables, communication cables associated with only one division, and raceways located in the fire areas identified in Table 3.3-3 will be conducted.	i.a) Results of the inspection will confirm that the separation between Class 1E divisions in the non-radiologically controlled area of the auxiliary building is consistent with Table 3.3-3.
796	3.3.00.07c.i.b	7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	i) Inspections of the as-built Class 1E division electrical cables, communication cables associated with only one division, and raceways located in the fire areas identified in Table 3.3-3 will be conducted.	i.b) Results of the inspection will confirm that the separation between Class 1E divisions in the radiologically controlled area of the auxiliary building is consistent with Table 3.3-3.
797	3.3.00.07c.ii.a	7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	ii) Inspections of the as-built fire barriers between the fire areas identified in Table 3.3-3 will be conducted.	ii.a) Results of the inspection will confirm that fire barriers exist between fire areas identified in Table 3.3-3 inside the non-radiologically controlled area of the auxiliary building.
798	3.3.00.07c.ii.b	7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	ii) Inspections of the as-built fire barriers between the fire areas identified in Table 3.3-3 will be conducted.	ii.b) Results of the inspection will confirm that fire barriers exist between fire areas identified in Table 3.3-3 inside the radiologically controlled area of the auxiliary building.
799	3.3.00.07d.i	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: i) Within the main control room and remote shutdown room, the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: i) Within the main control room and remote shutdown room, the vertical separation is 3 inches or more and the horizontal separation is 1 inch or more.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
800	3.3.00.07d.ii.a	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: ii) Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. 4) For configurations involving an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: ii.a) Within other plant areas inside containment (limited hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. 4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.

Table 3.3-6

Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
801	3.3.00.07d.ii.b	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: ii) Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. 4) For configurations involving an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: ii.b) Within other plant areas inside the non-radiologically controlled area of the auxiliary building (limited hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables < 2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. 4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
802	3.3.00.07d.ii.c	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: ii) Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. 4) For configurations involving an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: ii.c) Within other plant areas inside the radiologically controlled area of the auxiliary building (limited hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables < 2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. 4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
803	3.3.00.07d.iii.a	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iii) Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iii.a) Where minimum separation distances are not met inside containment, the circuits are run in enclosed raceways or barriers are provided.
804	3.3.00.07d.iii.b	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iii) Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iii.b) Where minimum separation distances are not met inside the non-radiologically controlled area of the auxiliary building, the circuits are run in enclosed raceways or barriers are provided.
805	3.3.00.07d.iii.c	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iii) Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iii.c) Where minimum separation distances are not met inside the radiologically controlled area of the auxiliary building, the circuits are run in enclosed raceways or barriers are provided.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
806	3.3.00.07d.iv.a	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iv) Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iv.a) For areas inside containment, a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
807	3.3.00.07d.iv.b	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iv) Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iv.b) For areas inside the non-radiologically controlled area of the auxiliary building, a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
808	3.3.00.07d.iv.c	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iv) Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: iv.c) For areas inside the radiologically controlled area of the auxiliary building, a report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed.
809	3.3.00.07d.v.a	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: v) Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: v.a) For areas inside containment, non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
810	3.3.00.07d.v.b	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: v) Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: v.b) For areas inside the non-radiologically controlled area of the auxiliary building, non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.
811	3.3.00.07d.v.c	7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: v) Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: v.c) For areas inside the radiologically controlled area of the auxiliary building, non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.
812	3.3.00.07e	7.e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.	Inspections of the as-built Class 1E communication cables will be conducted.	Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
813	3.3.00.08	8. Systems, structures, and components identified as essential targets are protected from the dynamic and environmental effects of postulated pipe ruptures.	Following as-built reconciliation, an inspection will be performed of the as-built high and moderate energy pipe rupture mitigation features for systems, structures, and components identified as essential targets.	An as-built Pipe Rupture Hazard Analysis Report exists and concludes that systems, structures, and components identified as essential targets can withstand the effects of postulated pipe rupture without loss of required safety function.
814	3.3.00.09	9. The reactor cavity sump has a minimum concrete thickness as shown in Table 3.3-5 between the bottom of the sump and the steel containment.	An inspection of the as-built containment building internal structures will be performed.	A report exists and concludes that the reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.
815	3.3.00.10.i	10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	i) A test will be performed to measure the leakage from the PCS storage tank based on measuring the water flow out of the leak chase collection system.	i) A report exists and concludes that total water flow from the leak chase collection system does not exceed 10 gal/hr.
816	3.3.00.10.ii	10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	ii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The vertical elevation of the shield building roof will be measured at a location at the outer radius of the roof (tension ring) and at a location on the same azimuth at the outer radius of the PCS storage tank before and after filling the PCS storage tank.	ii) A report exists and concludes that inspection and measurement of the PCS storage tank and the tension ring structure, before and after filling of the tank, shows structural behavior under normal loads to be acceptable.

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
817	3.3.00.10.iii	10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	iii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The boundaries of the PCS storage tank and the shield building roof above the tension ring will be inspected visually for excessive concrete cracking.	iii) A report exists and concludes that there is no visible water leakage from the PCS storage tank through the concrete and that there is no visible excessive cracking in the boundaries of the PCS storage tank and the shield building roof above the tension ring.
		11. Deleted		
818	3.3.00.12	12. The extended turbine generator axis intersects the shield building.	An inspection of the as-built turbine generator will be performed.	The extended axis of the turbine generator intersects the shield building.
819	3.3.00.13	13. Separation is provided between the structural elements of the turbine, annex and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in the safe shutdown earthquake without impact between structural elements of the buildings.	An inspection of the separation of the nuclear island from the annex, radwaste and turbine building structures will be performed. The inspection will verify the specified horizontal clearance between structural elements of the adjacent buildings, consisting of the reinforced concrete walls and slabs, structural steel columns and floor beams.	The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the annex and radwaste buildings and the nuclear island is 4 inches. The minimum horizontal clearance above floor elevation 100′-0″ between the structural elements of the turbine building and the nuclear island is 4 inches.
820	3.3.00.14	14. The external walls, doors, ceiling, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.	Type test, analysis, or a combination of type test and analysis will be performed for the external walls, doors, ceilings, and floors in the main control room, the central alarm station, and the secondary alarm station.	A report exists and concludes that the external walls, doors, ceilings, and floors in the main control room, the central alarm station, and the secondary alarm station are bullet-resistant to at least Underwriters Laboratory Ballistic Standard 752, level 4.
		15. Deleted		

Table 3.3-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
821	3.3.00.16	16. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.	An inspection will be performed to ensure that the location of the secondary security power supply equipment for alarm annunciator equipment and non-portable communications equipment is within a vital area.	Secondary security power supply equipment for alarm annunciator equipment and non-portable communication equipment is located within a vital area.
822	3.3.00.17	17. Vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion into a vital area.	An inspection of the as-built vital areas, and central and secondary alarm stations are performed.	Vital areas are locked and alarmed with active intrusion detection systems and intrusion is detected and annunciated in both the central and secondary alarm stations.
		18. Deleted		

Table 3.3-7 Nuclear Island Critical Structural Sections
<u>Containment Internal Structures</u>
South west wall of the refueling cavity
South wall of the west steam generator compartment
North east wall of the in-containment refueling water storage tank
In-containment refueling water storage tank steel wall
Column supporting the operating floor
<u>Auxiliary and Shield Building</u>
South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0"
Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6"
West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0"
North wall of MSIV east compartment (column line 11 between lines L and M), elevation 117'-6" to elevation 153'-0"
Roof slab at elevation 180'-0" adjacent to shield building cylinder
Floor slab on metal decking at elevation 135'-3"
2'-0" slab in auxiliary building (tagging room ceiling) at elevation 135'-3"
Finned floor in the main control room at elevation 135'-3"
Shield building roof, exterior wall of the PCS water storage tank
Shield building roof, interior wall of the PCS water storage tank
Shield building roof, tension ring and air inlets
Divider wall between the spent fuel pool and the fuel transfer canal
Shield building SC cylinder
Shield building SC to RC connection
<u>Nuclear Island Basemat Below Auxiliary Building</u>
Bay between reference column lines 9.1 and 11, and K and L
Bay between reference column lines 1 and 2 and K-2 and N

Figures 3.3-1 through 3.3-14 contain Security-Related Information. (See Attachment 1 to VCSNS Unit 2 COL Appendix C)

3.4 Initial Test Program

No ITAAC for this system.

3.5 Radiation Monitoring

Design Description

Radiation monitoring is provided for those plant areas where there is a significant potential for airborne contamination, for those process and effluent streams where contamination is possible, and in accessible areas to provide indication of unusual radiological events as identified in Tables 3.5-1, 3.5-2, 3.5-3, 3.5-4, and 3.5-5. The radiation monitoring component locations are as shown in Table 3.5-7.

1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.
2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
3. Separation is provided between system Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.
4. Safety-related displays identified in Table 3.5-1 can be retrieved in the main control room (MCR).
5. The process radiation monitors listed in Table 3.5-2 are provided.
6. The effluent radiation monitors listed in Table 3.5-3 are provided.
7. The airborne radiation monitors listed in Table 3.5-4 are provided.
8. The area radiation monitors listed in Table 3.5-5 are provided.

Table 3.5-1

Equipment Name	Tag No.	Seismic Cat. I	Class 1E	Qual. for Harsh Envir.	Safety-Related Display
Containment High Range Monitor	PXS-RE160	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE161	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE162	Yes	Yes	Yes	Yes
Containment High Range Monitor	PXS-RE163	Yes	Yes	Yes	Yes
MCR Radiation Monitoring Package A(1)	VBS-JS01A	Yes	Yes	No	No
MCR Radiation Monitoring Package B(1)	VBS-JS01B	Yes	Yes	No	No
Containment Atmosphere Monitor (Gaseous)	PSS-RE026	Yes	No	No	No
Containment Atmosphere Monitor (particulate, for RCS pressure boundary leakage detection)	PSS-RE027	Yes	No	No	No

Notes: (1) Each MCR Radiation Monitoring Package includes particulate, iodine and gaseous radiation monitors.

Table 3.5-2 Process Radiation Monitors	
Equipment List	**Equipment No.**
Steam Generator Blowdown	BDS-RE010
Steam Generator Blowdown	BDS-RE011
Component Cooling Water	CCS-RE001
Main Steam Line(1)	SGS-RY026
Main Steam Line(1)	SGS-RY027
Service Water Blowdown	SWS-RE008
Primary Sampling System Liquid Sample	PSS-RE050
Primary Sampling System Gaseous Sample	PSS-RE052
Containment Air Filtration Exhaust	VFS-RE001
Gaseous Radwaste Discharge	WGS-RE017

Note:

1. Each main steam line monitor includes a noble gas detector and primary-to-secondary side leak detector.

Table 3.5-3 Effluent Radiation Monitors	
Equipment List	**Equipment No.**
Plant Vent (Normal Range Particulate)	VFS-RE101
Plant Vent (Normal Range Iodine)	VFS-RE102
Plant Vent (Normal Range Radiogas)	VFS-RE103
Plant Vent (Mid Range Radiogas)	VFS-RE104A
Plant Vent (High Range Radiogas)	VFS-RE104B
Turbine Island Vent(1)	TDS-RE001
Liquid Radwaste Discharge	WLS-RE229
Wastewater Discharge	WWS-RE021

Note:

1. The turbine island vent includes a low and a high range detector.

Table 3.5-4 Airborne Radiation Monitors	
Equipment List	**Equipment No.**
Fuel Handling Area Exhaust Radiation Monitor	VAS-RE001
Auxiliary Building Exhaust Radiation Monitor	VAS-RE002
Auxiliary Building Exhaust Radiation Monitor	VAS-RE003
Annex Building Exhaust Radiation Monitor	VAS-RE008
Health Physics and Hot Machine Shop Exhaust Radiation Monitorp	VHS-RE001
Radwaste Building Exhaust Radiation Monitor	VRS-RE023

Table 3.5-5 Area Radiation Monitors	
Primary Sampling Room	RMS-RE008
Containment Area – Personnel Hatch Operating Deck (135'-3" Elevation)	RMS-RE009
Main Control Room	RMS-RE010
Chemistry Laboratory	RMS-RE011
Fuel Handling Area 1	RMS-RE012
Rail Car Bay/Filter Storage Area (Auxiliary Building Loading Bay)	RMS-RE013
Liquid and Gaseous Radwaste Area	RMS-RY014
Control Support Area	RMS-RE016
Radwaste Building Mobile Systems Facility	RMS-RE017
Hot Machine Shop	RMS-RE018
Annex Staging and Storage Area	RMS-RE019
Fuel Handling Area 2	RMS-RE020
Containment Area – Personnel Hatch Maintenance Level (100'-0" Elevation)	RMS-RE021

Note:

1. Deleted.

Table 3.5-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
823	3.5.00.01.i	1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 3.5-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 3.5-1 is located on the Nuclear Island.
824	3.5.00.01.ii	1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
825	3.5.00.01.iii	1. The seismic Category I equipment identified in Table 3.5-1 can withstand seismic design basis loads without loss of safety function.	iii) Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
826	3.5.00.02.i	2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	i) Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	i) A report exists and concludes that Class 1E equipment identified in Table 3.5-1 as being located in a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
827	3.5.00.02.ii	2. The Class 1E equipment identified in Table 3.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	ii) Inspection will be performed of the as-built Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	ii) A report exists and concludes that the as-built Class 1E equipment and the associated wiring, cables, and terminations identified in Table 3.5-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.
828	3.5.00.03	3. Separation is provided between system Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See ITAAC Table 3.3-6, item 7.d).	See ITAAC Table 3.3-6, item 7.d).

Table 3.5-6 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
829	3.5.00.04	4. Safety-related displays identified in Table 3.5-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays in the MCR.	Safety-related displays identified in Table 3.5-1 can be retrieved in the MCR.
830	3.5.00.05	5. The process radiation monitors listed in Table 3.5-2 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-2 exists.
831	3.5.00.06	6. The effluent radiation monitors listed in Table 3.5-3 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-3 exists.
832	3.5.00.07	7. The airborne radiation monitors listed in Table 3.5-4 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-4 exists.
833	3.5.00.08	8. The area radiation monitors listed in Table 3.5-5 are provided.	Inspection for the existence of the monitors will be performed.	Each of the monitors listed in Table 3.5-5 exists.

Table 3.5-7		
Component Name	**Tag No.**	**Component Location**
Containment High Range Radiation Monitor	PXS-RE160	Containment
Containment High Range Radiation Monitor	PXS-RE161	Containment
Containment High Range Radiation Monitor	PXS-RE162	Containment
Containment High Range Radiation Monitor	PXS-RE163	Containment
MCR Radiation Monitoring Package A	VBS-RY01A	Auxiliary Building
MCR Radiation Monitoring Package B	VBS-RY01B	Auxiliary Building
Containment Atmosphere Radiation Monitor (Gaseous)	PSS-RE026	Auxiliary Building
Containment Atmosphere Radiation Monitor (particulate, for RCS pressure boundary leakage detection)	PSS-RE027	Auxiliary Building
Steam Generator Blowdown Radiation Monitor	BDS-RE010	Turbine Building
Steam Generator Blowdown Radiation Monitor	BDS-RE011	Turbine Building
Component Cooling Water Radiation Monitor	CCS-RE001	Turbine Building
Main Steam Line Radiation Monitor	SGS-RY026	Auxiliary Building
Main Steam Line Radiation Monitor	SGS-RY027	Auxiliary Building
Service Water Blowdown Radiation Monitor	SWS-RE008	Turbine Building

Table 3.5-7		
Component Name	**Tag No.**	**Component Location**
Primary Sampling System Liquid Sample Radiation Monitor	PSS-RE050	Auxiliary Building
Primary Sampling System Gaseous Sample Radiation Monitor	PSS-RE052	Auxiliary Building
Containment Air Filtration Exhaust Radiation Monitor	VFS-RE001	Annex Building
Gaseous Radwaste Discharge Radiation Monitor	WGS-RE017	Auxiliary Building
Plant Vent (Normal Range Particulate) Radiation Monitor	VFS-RE101	Auxiliary Building
Plant Vent (Normal Range Iodine) Radiation Monitor	VFS-RE102	Auxiliary Building
Plant Vent (Normal Range Radiogas) Radiation Monitor	VFS-RE103	Auxiliary Building
Plant Vent (Mid Range Radiogas) Radiation Monitor	VFS-RE104A	Auxiliary Building
Plant Vent (High Range Radiogas) Radiation Monitor	VFS-RE104B	Auxiliary Building
Turbine Island Vent Radiation Monitor	TDS-RE001	Turbine Building
Liquid Radwaste Discharge Monitor	WLS-RE229	Radwaste Building
Wastewater Discharge Radiation Monitor	WWS-RE021	Turbine Building
Fuel Handling Area Exhaust Radiation Monitor	VAS-RE001	Auxiliary Building
Auxiliary Building Exhaust Radiation Monitor	VAS-RE002	Auxiliary Building
Auxiliary Building Exhaust Radiation Monitor	VAS-RE003	Auxiliary Building
Annex Building Exhaust Radiation Monitor	VAS-RE008	Annex Building
Health Physics and Hot Machine Shop Exhaust Radiation Monitor	VHS-RE001	Annex Building
Radwaste Building Exhaust Radiation Monitor	VRS-RE023	Radwaste Building
Primary Sampling Room	RMS-RE008	Auxiliary Building
Containment Area – Personnel Hatch – Operating Deck	RMS-RE009	Auxiliary Building
Main Control Room	RMS-RE010	Auxiliary Building
Chemistry Laboratory	RMS-RE011	Auxiliary Building
Fuel Handling Area 1	RMS-RE012	Auxiliary Building
Rail Car Bay/Filter Storage Area (Auxiliary Building Loading Bay)	RMS-RE013	Auxiliary Building
Liquid and Gaseous Radwaste Area	RMS-RY014	Radwaste Building
Control Support Area	RMS-RE016	Annex Building
Radwaste Building Mobile Systems Facility	RMS-RE017	Radwaste Building

Table 3.5-7		
Component Name	**Tag No.**	**Component Location**
Hot Machine Shop	RMS-RE018	Annex Building
Annex Staging and Storage Area	RMS-RE019	Annex Building
Fuel Handling Area 2	RMS-RE020	Auxiliary Building
Containment Area – Personnel Hatch – Maintenance Level	RMS-RE021	Auxiliary Building

3.6 Reactor Coolant Pressure Boundary Leak Detection

Design Description

The reactor coolant pressure boundary leakage detection monitoring provides a means of detecting and quantifying the reactor coolant leakage. To detect unidentified leakage inside containment, the following diverse methods are provided to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Leakage detection monitoring is accomplished using instrumentation and other components of several systems.

1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.

Table 3.6-1
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
834	3.6.00.01.i	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See sections: i) See ITAAC Table 2.3.10-4, Item 7.a for the sump level measuring instruments WLS-034 and WLS-035.	See sections: i) See ITAAC Table 2.3.10-4, Item 7.a for the sump level measuring instruments WLS-034 and WLS-035.
835	3.6.00.01.ii	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See sections: ii) See ITAAC Table 3.5-6, Item 1 for the containment atmosphere radioactivity monitor PSS-RE027.	See ITAAC sections: ii) See ITAAC Table 3.5-6, Item 1 for the containment atmosphere radioactivity monitor PSS-RE027.

Table 3.6-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
836	3.6.00.01.iii	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See sections: iii) See ITAAC Table 2.1.2-4, Items 5.a), 7.a), and 10 for the pressurizer level measuring instruments RCS-195A, RCS-195B, RCS-195C, and RCS-195D.	See sections: iii) See ITAAC Table 2.1.2-4, Items 5.a), 7.a), and 10 for the pressurizer level measuring instruments RCS-195A, RCS-195B, RCS-195C, and RCS-195D.
837	3.6.00.01.iv	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See sections: iv) See ITAAC Table 2.1.2-4, Items 5.a) and 7.a) for the RCS hot and cold leg temperature instruments RCS-121A, RCS-121B, RCS-121C, RCS-121D, RCS-122A, RCS-122B, RCS-122C, RCS-122D, RCS-131A, RCS-131B, RCS-131C, RCS-131D, RCS-132A, RCS-132B, RCS-132C, and RCS-132D.	See sections: iv) See ITAAC Table 2.1.2-4, Items 5.a) and 7.a) for the RCS hot and cold leg temperature instruments RCS-121A, RCS-121B, RCS-121C, RCS-121D, RCS-122A, RCS-122B, RCS-122C, RCS-122D, RCS-131A, RCS-131B, RCS-131C, RCS-131D, RCS-132A, RCS-132B, RCS-132C, and RCS-132D.
838	3.6.00.01.v	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See sections: v) See ITAAC Table 2.1.2-4, Items 5.a), 7.a), and 10 for the RCS pressure instruments RCS-140A, RCS-140B, RCS-140C, and RCS-140D.	See sections: v) See ITAAC Table 2.1.2-4, Items 5.a), 7.a), and 10 for the RCS pressure instruments RCS-140A, RCS-140B, RCS-140C, and RCS-140D.
839	3.6.00.01.vi	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	See sections: vi) See ITAAC Table 2.3.2-4, Item 13 for the letdown and makeup flow instruments CVS-001 and CVS-025.	See sections: vi) See ITAAC Table 2.3.2-4, Item 13 for the letdown and makeup flow instruments CVS-001 and CVS-025.
840	3.6.00.01.vii	1. The diverse leak detection methods provide the nonsafety-related function of detecting small leaks when RCS leakage indicates possible reactor coolant pressure boundary degradation.	vii) See ITAAC Table 2.3.10-4, Item 10 for the reactor coolant drain tank level instrument WLS-002.	vii) See ITAAC Table 2.3.10-4, Item 10 for the reactor coolant drain tank level instrument WLS-002.

3.7 Design Reliability Assurance Program

The Design Reliability Assurance Program (D-RAP) is a program that will be performed during the detailed design and equipment specification phase prior to initial fuel load. The D-RAP evaluates and sets priorities for the structures, systems, and components (SSCs) in the design,

based on their degree of risk significance. The risk-significant components are listed in Table 3.7-1.

The objective of the D-RAP program is to provide reasonable assurance that risk-significant SSCs (Table 3.7-1) are designed such that: (1) assumptions from the risk analysis are utilized, (2) SSCs (Table 3.7-1) when challenged, function in accordance with the assumed reliability, (3) SSCs (Table 3.7-1) whose failure results in a reactor trip, function in accordance with the assumed reliability, and (4) maintenance actions to achieve the assumed reliability are identified.

1. The D-RAP ensures that the design of SSCs within the scope of the reliability assurance program (Table 3.7-1) is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability).

Table 3.7-1
Risk-Significant Components

Equipment Name	Tag No.
Component Cooling Water System (CCS)	
Component Cooling Water Pumps	CCS-MP-01A/B
Containment System (CNS)	
Containment Vessel	CNS-MV-01
Hydrogen Igniters	VLS-EH-1 through -66
Chemical and Volume Control System (CVS)	
Makeup Pumps	CVS-MP-01A/B
Makeup Pump Suction and Discharge Check Valves	CVS-PL-V113 CVS-PL-V160A/B
Letdown Discharge Isolation Valves	CVS-PL-V045 CVS-PL-V047
Diverse Actuation System (DAS)	
DAS Processor Cabinets and Control Panel (used to provide automatic and manual actuation)	DAS-JD-001 DAS-JD-002 DAS-JD-003 OCS-JC-20
Auxiliary Building UPS Distribution Panels (provide power to DAS)	EDS2-EA-12, EDS3-EA-14A
Control Cabinets for the Rod Drive MG Sets (generator field control relays)	PLS-JD-RDM001 PLS-JD-RDM002
Containment Isolation Valves Controlled by DAS	CVS-PL-V045, -V047 VFS-PL-V003, -V004, -V009, -V010 WLS-PL-V055, -V057
Main ac Power System (ECS)	

Table 3.7-1 Risk-Significant Components	
Equipment Name	**Tag No.**
Reactor Coolant Pump Switchgear	ECS-ES-31, -32, -41, -42, -51, -52, -61, -62
Ancillary Diesel Generators	ECS-MS-01, -02
6900 Vac Buses	ECS-ES-1, -2
Main and Startup Feedwater System (FWS)	
Startup Feedwater Pumps	FWS-MP-03A/B
General I&C	
IRWST Level Sensors	PXS-045, -046, -047, -048
RCS Hot Leg Level Sensors	RCS-160A/B
Pressurizer Pressure Sensors	RCS-191A/B/C/D
Pressurizer Level Sensors	RCS-195A/B/C/D
Steam Generator Narrow-Range Level Sensors	SGS-001, -002, -003, -004, -005, -006, -007, -008
Steam Generator Wide-Range Level Sensors	SGS-011, -012, -013, -014, -015, -016, -017, -018
Main Steam Line Pressure Sensors	SGS-030, -031, -032, -033, -034, -035, -036, -037
Main Feedwater Low-Range Flow Sensors	FWS-050A/C/E, -051A/C/E
Startup Feedwater Flow Sensors	SGS-055A/B/C/D/E, -056A/B/C/D/E
CMT Level Sensors	PXS-011A/B/C/D, -012A/B/C/D, -013A/B/C/D, -014A/B/C/D
Class 1E dc Power and Uninterruptible Power System (IDS)	
250 Vdc 24-Hour Batteries	IDSA-DB-1A/B, IDSB-DB-1A/B, IDSC-DB-1A/B, IDSD-DB-1A/B
250 Vdc 24-Hour Buses	IDSA-DS-1, IDSB-DS-1 IDSC-DS-1, IDSD-DS-1
250 Vdc 24-Hour Battery Chargers	IDSA-DC-1, IDSB-DC-1, IDSC-DC-1, IDSD-DC-1
250 Vdc and 120 Vac Distribution Panels	IDSA-DD-1, IDSA-EA-1/-2, IDSB-DD-1, IDSB-EA-1/-2/-3, IDSC-DD-1, IDSC-EA-1/-2/-3, IDSD-DD-1, IDSD-EA-1/-2
Fused Transfer Switch Boxes	IDSA-DF-1, IDSB-DF-1/-2, IDSC-DF-1/-2, IDSD-DF-1
250 Vdc Motor Control Centers	IDSA-DK-1, IDSB-DK-1, IDSC-DK-1, IDSD-DK-1

Table 3.7-1 Risk-Significant Components	
Equipment Name	**Tag No.**
250 Vdc 24-Hour Inverters	IDSA-DU-1, IDSB-DU-1, IDSC-DU-1, IDSD-DU-1
Passive Containment Cooling System (PCS)	
Recirculation Pumps	PCS-MP-01A/B
PCCWST Drain Isolation Valves	PCS-PL-V001A/B/C
Plant Control System (PLS)	
PLS Actuation Software (used to provide control functions)	Refer to Table 3.7-2
PLS Actuation Hardware (used to provide control functions)	Refer to Table 3.7-2
Protection and Monitoring System (PMS)	
PMS Actuation Software (used to provide automatic control functions)	Refer to Tables 2.5.2-2 and 2.5.2-3
PMS Actuation Hardware (used to provide automatic control functions)	Refer to Tables 2.5.2-2 and 2.5.2-3
MCR 1E Displays and System Level Controls	OCS-JC-10, -11
Reactor Trip Switchgear	PMS-JD-RTS A01/02, B01/02, C01/02, D01/02
Passive Core Cooling System (PXS)	
IRWST Hood Vent Covers	PXS-MY-Y21, -Y22, -Y23, -Y24, -Y25, -Y26, -Y27, -Y28, -Y29, -Y30, -Y31, -Y32, -Y33, -Y41, -Y47, -Y48
IRWST Steam Generator Wall Vent Covers	PXS-MY-Y61, -Y62, -Y63, -Y64
IRWST Overflow Weir Covers	PXS-MY-Y71, -Y72, -Y73, -Y74, -Y75, -Y76
IRWST Screens	PXS-MY-Y01A/B/C
Containment Recirculation Screens	PXS-MY-Y02A/B
CMT Discharge Isolation Valves	PXS-PL-V014A/B, -V015A/B
CMT Discharge Check Valves	PXS-PL-V016A/B, -V017A/B
Accumulator Discharge Check Valves	PXS-PL-V028A/B, -V029A/B
PRHR HX Control Valves	PXS-PL-V108A/B
Containment Recirculation Squib Valves	PXS-PL-V118A/B, -V120A/B
IRWST Injection Check Valves	PXS-PL-V122A/B, -V124A/B
IRWST Injection Squib Valves	PXS-PL-V123A/B, -V125A/B
IRWST Gutter Bypass Isolation Valves	PXS-PL-V130A/B
Reactor Coolant System (RCS)	
ADS Stage 1/2/3 Valves (MOVs)	RCS-PL-V001A/B, -V011A/B RCS-PL-V002A/B, -V012A/B RCS-PL-V003A/B, -V013A/B
ADS Stage 4 Valves (Squibs)	RCS-PL-V004A/B/C/D

Table 3.7-1 Risk-Significant Components	
Equipment Name	**Tag No.**
Pressurizer Safety Valves	RCS-PL-V005A/B
Reactor Vessel Insulation Water Inlet and Steam Vent Devices	RXS-MN-01
Reactor Cavity Doorway Damper	–
Fuel Assemblies	157 assemblies with tag numbers beginning with RXS-FA
Normal Residual Heat Removal System (RNS)	
Residual Heat Removal Pumps	RNS-MP-01A/B
RNS Motor-Operated Valves	RNS-PL-V011, -V022, -V023, -V055
RNS Stop Check Valves RNS Check Valves	RNS-PL-V015A/B RNS-PL-V017A/B
RNS Check Valves	RNS-PL-V007A/B, -V013, -V056
Spent Fuel Cooling System (SFS)	
Spent Fuel Cooling Pumps	SFS-MP-01A/B
Steam Generator System (SGS)	
Main Steam Safety Valves	SGS-PL-V030A/B, -V031A/B, -V032A/B, -V033A/B, -V034A/B, -V035A/B
Main Steam Line Isolation Valves	SGS-PL-V040A/B
Main Feedwater Isolation Valves	SGS-PL-V057A/B
Service Water System (SWS)	
Service Water Cooling Tower Fans	SWS-MA-01A/B
Service Water Pumps	SWS-MP-01A/B
Nuclear Island Nonradioactive Ventilation System (VBS)	
MCR Ancillary Fans	VBS-MA-10A/B
I&C Room B/C Ancillary Fans	VBS-MA-11, -12
Containment Air Filtration System (VFS)	
Containment Purge Isolation Valves	VFS-PL-V003 VFS-PL-V004 VFS-PL-V009 VFS-PL-V010
Chilled Water System (VWS)	
Air Cooled Chiller Pumps	VWS-MP-02, -03

Table 3.7-1 Risk-Significant Components	
Equipment Name	**Tag No.**
Air Cooled Chillers	VWS-MS-02, -03
Liquid Radwaste System (WLS)	
Sump Containment Isolation Valves	WLS-PL-V055 WLS-PL-V057
Onsite Standby Power System (ZOS)	
Engine Room Exhaust Fans	VZS-MY-V01A/B, -V02A/B
Onsite Diesel Generators	ZOS-MS-05A/B

Note: Dash (-) indicates not applicable.

Table 3.7-2 PLS D-RAP Control Functions
CVS Reactor Makeup
RNS Reactor Injection from cask loading pit
Startup Feedwater from CST
Spent Fuel Cooling
Component Cooling of RNS and SFS Heat Exchangers
Service Water Cooling of CCS Heat Exchangers
Onsite Diesel Generators
Hydrogen Igniters

Table 3.7-3 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
841	3.7.00.01	1. The D-RAP ensures that the design of SSCs within the scope of the reliability assurance program (Table 3.7-1) is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability).	An analysis will confirm that the design of RAP SSCs identified in Table 3.7-1 has been completed in accordance with applicable D-RAP activities.	An analysis report documents that safety-related SSCs identified in Table 3.7-1 have been designed in accordance with a 10 CFR 50 Appendix B quality program. An analysis report documents that non-safety-related SSCs identified in Table 3.7-1 have been designed in accordance with a program that satisfies quality assurance requirements for SSCs important to investment protection.

C.3.8 Emergency Planning ITAAC

C.3.8.1.1 Emergency Classification System

Table C.3.8-1 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
842	C.3.8.01.01.01	1.1 A standard emergency classification and emergency action level (EAL) scheme exists, and identifies facility system and effluent parameters constituting the bases for the classification scheme. [D.1**] [**D.1 corresponds to NUREG-0654/FEMA-REP-1 evaluation criteria.]	1.1 An inspection of the Control Room, Technical Support Center (TSC), and Emergency Operations Facility (EOF) will be performed to verify that they have displays for retrieving facility system and effluent parameters that are specified in the Emergency Classification and EAL scheme and the displays are functional.	1.1 The specified parameters as listed in UFSAR Table 7.5-1 are retrievable in the Control Room, TSC and EOF, and the ranges of the displays encompass the values specified in the Emergency Classification and EAL Technical Basis Document.

C.3.8.1.2 Notification Methods and Procedures

Table C.3.8-1 (continued)
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Program Commitment	Inspections, Tests, Analyses	Acceptance Criteria
843	C.3.8.01.02.01	2.1 The means exists to notify responsible State and local organizations within 15 minutes after the licensee declares an emergency. [E.1]	2.1. A test of the ESSX line will be performed to demonstrate the capabilities for providing initial notification to the offsite authorities after a simulated emergency classification.	2.1 Using the ESSX line the State of South Carolina and the counties of Fairfield, Lexington, Newberry and Richland received notification within 15 minutes after the declaration of an emergency from the Control Room and the EOF. A test of each facility ESSX line was successful using the standard South Carolina notification form.
844	C.3.8.01.02.02	2.2 The means exists to notify emergency response personnel. [E.2]	2.2 A test of the primary and back-up ERO notification systems will be performed.	2.2 Emergency response personnel received the notification message and mobilization communication was validated by personnel response to the notification system and by telephone during off-hours. Also demonstrated work hours electronic notification and plant page system during working hours.
845	C.3.8.01.02.03	2.3 The means exists to notify and provide instructions to the populace within the plume exposure EPZ. [E.6]	2.3 The full test of the ANS capabilities will be conducted.	2.3 The ANS was demonstrated to notify and provide instructions to the public and was demonstrated to meet the design objectives, as stated in the emergency plan.

C.3.8.1.3 Emergency Communications

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
846	C.3.8.01.03.01	3.1 The means exists for communications among the Control Room, TSC, EOF, principal State and local emergency operations centers (EOCs), and radiological field assessment teams. [F.1.d]	3.1 A test will be performed of the capabilities. The test for the contact with the principal EOCs and the radiological field assessment teams will be from the Control Room and the EOF. See also ITA 5.1.1.	3.1 Communications (both primary and secondary methods/systems) were established among the Control Room and the EOF with the South Carolina Emergency Management Division (SCEMD) warning point and EOC; Fairfield County Warning Point and EOC; Richland County Warning Point and EOC; Newberry County Warning Point and EOC; and Lexington County Warning Point and EOC. Communications were established between the Control Room and the EOF with the VCSNS radiological field monitoring teams. See also AC 5.1.4.
847	C.3.8.01.03.02	3.2 The means exists for communications from the Control Room, TSC, and EOF to the NRC headquarters and regional office EOCs (including establishment of the Emergency Response Data System (ERDS) [or its successor system] between the onsite computer system and the NRC Operations Center.) [F.1.f]	3.2 A test is performed of the capabilities to communicate using ENS from the Control Room, TSC and EOF to the NRC headquarters and regional office EOCs. HPN is tested to ensure communications between the TSC and EOF with the NRC Operations Center. ERDS is established [or its successor system] between the onsite computer systems and the NRC Operations Center.	3.2 Communication was established from the Control Room, TSC and EOF to the NRC headquarters and regional office EOCs utilizing the ENS. The TSC and EOF demonstrated communications with the NRC Operations Center using HPN. The access port for ERDS [or its successor system] successfully completed a transfer of data to the NRC Operations Center.

C.3.8.1.4 Public Education and Information

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
848	C.3.8.01.04.01	4.1 The licensee has provided space which may be used for a limited number of the news media. [G.3.b]	4.1 An inspection of the facility/area provided for the news media will be performed in the Joint Information Center (JIC). The space provides adequate equipment to support JIC operation, including communications with the site and with the Emergency Operation Centers in the state and counties as well as a limited number of news media.	4.1 The licensee has provided space which may be used for a limited number of the news media in the Joint Information Center. This space provides the needed equipment per approved administrative procedures

C.3.8.1.5 Emergency Facilities and Equipment

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
849	C.3.8.01.05.01.01	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.1 The TSC has at least 3000 ft^2 of floor space.
850	C.3.8.01.05.01.02	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.2 The TSC is located outside the Protected Area and advanced communication capabilities are available and utilized to ensure communications between the emergency response facilities. Procedures are in place to enhance passage through security checkpoints expeditiously.
851	C.3.8.01.05.01.03	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.3 The TSC ventilation system includes a high efficiency particulate air (HEPA) and charcoal filter and radiation monitors are installed.

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
852	C.3.8.01.05.01.04	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.4 TSC communications equipment is installed per specifications and is operable. Communications have been initiated and found to be acceptable in giving and receiving voice communications with the Control Room, the OSC and the EOF.
853	C.3.8.01.05.01.05	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.5 The TSC has the means to receive, store, process, and display plant and environmental information as listed in UFSAR Table 7.5-1, and to initiate emergency measures and conduct emergency assessment.
854	C.3.8.01.05.01.06	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.6 There is an OSC located inside the Unit. It is separate from the Control Room and within the Protected Area.
855	C.3.8.01.05.01.07	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.7 OSC communications equipment is installed, and voice transmission and reception have been demonstrated between the OSC, OSC Teams, the TSC, and Control Room.
856	C.3.8.01.05.01.08	5.1 The licensee has established a TSC and onsite OSC. [H.1, H.9]	5.1.1 An inspection of the TSC and OSC will be performed, including a test of the capabilities. These facilities will meet the criteria of NUREG-0696 with exceptions.	5.1.8 A reliable and backup electrical supply is available for the TSC.
857	C.3.8.01.05.02.01	5.2 The licensee has established an EOF. [H.2]	5.2 An inspection of the EOF will be performed, including a test of the capabilities. The EOF is located outside of the 10 mile Emergency Planning Zone.	5.2.1 The EOF working space size is consistent with NUREG-0696 (75 ft^2/ person), and is large enough for required systems, equipment, records and storage.

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
858	C.3.8.01.05.02.02	5.2 The licensee has established an EOF. [H.2]	5.2 An inspection of the EOF will be performed, including a test of the capabilities. The EOF is located outside of the 10 mile Emergency Planning Zone.	5.2.2 The EOF habitability is consistent with Table 2 of NUREG-0696. • Distance at or beyond 10 mi of the TSC • Built to meet the criteria of the County Building Code
859	C.3.8.01.05.02.03	5.2 The licensee has established an EOF. [H.2]	5.2 An inspection of the EOF will be performed, including a test of the capabilities. The EOF is located outside of the 10 mile Emergency Planning Zone.	5.2.3 EOF communications equipment is installed, and voice transmission and reception are accomplished with the Control Room, TSC, radiological monitoring teams, NRC, state and county agencies using typical data generated during facility activation.
860	C.3.8.01.05.02.04	5.2 The licensee has established an EOF. [H.2]	5.2 An inspection of the EOF will be performed, including a test of the capabilities. The EOF is located outside of the 10 mile Emergency Planning Zone.	5.2.4 Radiological data identified in the EP Unit Annex, meteorological data, and plant system data pertinent to determining offsite protective measures as listed in UFSAR Table 7.5-1 are available and displayed when activated in the EOF.

C3.8.1.6 Accident Assessment

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
861	C.3.8.01.06.01	6.1 The means exists to provide initial and continuing radiological assessment throughout the course of an accident. [I.2]	6.1 A test will be performed to demonstrate that the means exist to provide initial and continuing radiological assessment throughout the course of an accident through the plant computer or communications with the Control Room.	6.1 The means exist to provide initial and continuing radiological assessment through displays of instrumentation indicators in the Control Room, TSC and EOF during the course of drills and/or exercises.

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
862	C.3.8.01.06.02	6.2 The means exists to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors. [I.3]	6.2 A test will be performed to demonstrate that the means exist to determine the source term of releases of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.	6.2 Emergency Planning Implementing Procedures, through use in training and a drill, provided direction to accurately calculate the source terms and the magnitude of the release of postulated accident scenario releases.
863	C.3.8.01.06.03	6.3 The means exists to continuously assess the impact of the release of radioactive materials to the environment, accounting for the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions. [I.4]	6.3 A test will be performed to demonstrate that the impact of a radiological release to the environment is able to be assessed by utilizing the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions.	6.3 Response personnel demonstrated that the means exist to continuously assess the impact of the release of radioactive materials to the environment, accounting for the relationship between effluent monitor readings, and onsite and offsite exposures and contamination for various meteorological conditions under drill conditions.
864	C.3.8.01.06.04	6.4 The means exists to acquire and evaluate meteorological information. [I.5]	6.4 A test will be performed to acquire and evaluate meteorological data/ information.	6.4 Meteorological data was available at the EOF, TSC, Control Room, offsite NRC Operations Center, and the state of South Carolina. This data was in the format needed for the appropriate emergency planning implementing procedures.
865	C.3.8.01.06.05	6.5 The means exists to make rapid assessments of actual or potential magnitude and locations of radiological hazards through liquid or gaseous release pathways, including activation, notification means, field team composition, transportation, communication, monitoring equipment, and estimated deployment times. [I.8]	6.5 A test will be performed of the capabilities to make rapid assessments of actual or potential magnitude and locations of radiological hazards through liquid or gaseous release pathways, including activation, notification means, field team composition, transportation, communication, monitoring equipment, and estimated deployment times.	6.5 The field monitoring team(s) was activated and evaluated. They demonstrated an ability to make rapid assessment of actual or potential magnitude and locations of any radiological hazards through simulated liquid or gaseous release pathways. A qualified field team was notified, activated, briefed and dispatched from the EOF during a radiological release scenario. The team demonstrated the procedural guidance in team composition, use of monitoring equipment, communication

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				from the field, and locating specific sampling locations.
866	C.3.8.01.06.06	6.6 The capability exists to detect and measure radioiodine concentrations in air in the plume exposure EPZ, as low as 10^{-7} µCi/cc (microcuries per cubic centimeter) under field conditions. [I.9]	6.6 A test will be performed of the capabilities to detect and measure radioiodine concentrations in air in the plume exposure EPZ, as low as 10^{-7} µCi/cc (microcuries per cubic centimeter) under field conditions.	6.6 A field monitoring team was dispatched during a radiological release scenario and demonstrated the use of sampling and detection equipment for air concentrations in the plume exposure EPZ, as low as 10^{-7} µCi/cc.
867	C.3.8.01.06.07	6.7 The means exists to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the EPA protective action guides (PAGs). [I.10]	6.7 A test will be performed of the capabilities to estimate integrated dose from the projected and actual dose rates, and for comparing these estimates with the EPA protective action guides.	6.7 The means were demonstrated to estimate integrated dose from the dose assessment program and the field monitoring team reading during a radioactive release scenario. The results were compared with the EPA PAGs.

C3.8.1.7 Protective Response

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
868	C.3.8.01.07.01	7.1 The means exists to warn and advise onsite individuals of an emergency, including those in areas controlled by the operator, including: [J.1] 1. employees not having emergency assignments; 2. visitors; 3. contractor and construction personnel; and 4. other persons who may be in the public access areas, on or passing through the site, or within the owner controlled area.	7.1 A test will be performed of the capabilities to warn and advise onsite individuals of an emergency, including those in areas controlled by the operator.	7.1 The means exist and was successfully demonstrated to warn and advise onsite individuals including: 1. non-essential employees; 2. visitors; 3. contractor and construction personnel; and 4. other personnel within the owner controlled area.

C.3.8.1.8 Exercises and Drills

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
869	C3.8.01.08.01.01	8.1 Licensee conducts a full participation exercise to evaluate major portions of emergency response capabilities, which includes participation by each state and local agency within the plume exposure EPZ, and each state within the ingestion control EPZ. [N.1]	8.1 A full participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.	8.1.1 The exercise was completed within the specified time periods of Appendix E to 10 CFR Part 50, onsite exercise objectives were met, including: A. *Accident Assessment and Classification* 1. Demonstrate the ability to identify initiating conditions, determine emergency action levels (EAL) parameters, and correctly classify the emergency throughout the exercise. Standard Criteria: a. Determine the correct emergency classification level based on events which were in progress, considering past events and their impact on the current conditions within 15 minutes from the time the initiating condition(s) or EAL is exceeded during the exercise. B. *Notifications* 1. Demonstrate the ability to notify responsible state and local government agencies within 15 minutes and the NRC within 60 minutes after declaring an emergency. Standard Criteria: a. Accurately transmit information in accordance with Emergency Plan Implementing Procedures within 15 minutes of the emergency declaration. 2. Demonstrate the ability to alert, notify, and mobilize site emergency response personnel

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				during the exercise. Standard Criteria: a. Complete the designated actions in accordance with Emergency Plan Implementing Procedures and perform the announcement concerning the initial event classification of Alert or higher during the exercise. b. Mobilize site emergency responders in accordance with Emergency Plan Implementing Procedures at the initial event classification for an Alert or higher during the exercise. 3. Demonstrate the ability to warn or advise onsite individuals of emergency conditions. Standard Criteria: a. Initiate notification of onsite protective actions. 4. Demonstrate the capability of the Alert and Notification System (ANS) to operate properly when required. Standard Criteria: a. 90% of the sirens operate properly, as indicated by the feedback system. C. *Emergency Response* 1. Demonstrate the ability to direct and control emergency operations Standard Criteria a. Command and control is demonstrated by the Control Room (simulator) in the early phase of the emergency and by the Technical Support Center

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				(TSC) and Emergency Operations Facility (EOF) within 75 minutes of the emergency declaration. 2. Demonstrate the ability to transfer emergency direction from the Control Room (simulator) to the EOF. Standard Criteria: a. Turnover briefings are conducted in accordance with Emergency Plan Implementing Procedures. 3. Demonstrate the ability to prepare for around-the-clock staffing requirements. Standard Criteria: a. Complete 24-hour staffing assignments. 4. Demonstrate the ability to perform assembly and accountability for personnel in the Protected Area within 30 minutes of the declaration of a Site Area Emergency or higher classification. Standard Criteria: a. Protected Area personnel assembly and accountability completed within 30 minutes of the declaration of a Site Area Emergency or higher classification. D. *Emergency Response Facilities* 1. Demonstrate activation of the Operational Support Center (OSC), and full functional operation of the TSC and EOF within 75 minutes of a declaration of Alert or higher emergency classification.

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				Standard Criteria: a. The TSC, OSC, and EOF are activated within 75 minutes of the declaration of an Alert of higher emergency classification. 2. Demonstrate the adequacy of equipment, security, provisions, and habitability precautions for the TSC, OSC, and EOF, as appropriate. Standard Criteria: a. Demonstrate the adequacy of the emergency equipment in the emergency response facilities as specified in Emergency Plan Implementing Procedures, as appropriate. b. The security force implements and follows applicable security plan procedures as appropriate during the exercise. c. Demonstrate the capability of TSC and EOF equipment and data displays to clearly identify and reflect the affected unit. 3. Demonstrate the adequacy of communications for emergency support resources. Standard Criteria: a. Emergency response facility personnel are able to operate primary or backup communication systems in accordance with Emergency Plan Implementing Procedures as needed during the exercise. b. Primary or backup emergency response communication systems

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				listed in the Emergency Plan Implementing Procedures are available and operational for the duration of the exercise. E. *Radiological Assessment and Control* 1. Demonstrate the ability to obtain onsite radiological surveys and samples Standard Criteria: a. Health Physics personnel demonstrate the ability to obtain appropriate instruments and perform surveys as needed during the exercise. b. Airborne samples are taken, as appropriate, in accordance with Emergency Plan Implementing Procedures during the exercise. 2. Demonstrate the ability to continuously monitor and control radiation exposure to emergency workers. Standard Criteria: a. Emergency workers are issued self-reading dosimeters when radiation levels require, and exposures are controlled to 10 CFR Part 20 limits (unless the Emergency Coordinator authorizes emergency limits), as appropriate during the exercise. b. Exposure records are available during the exercise. 3. Demonstrate the ability to assemble and deploy field monitoring teams.

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				Standard Criteria: a. Field Monitoring Teams are briefed, obtain equipment, and are dispatched in accordance with Emergency Plan Implementing Procedures. 4. Demonstrate the ability to collect and disseminate field team data Standard Criteria: a. Field teams collect data for dose rate and airborne radioactivity levels, as applicable, in accordance with emergency plan implementing procedures b. Field team communicates data to the EOF in accordance with Emergency Plan Implementing Procedures during the exercise. 5. Demonstrate the ability to develop dose projections Standard Criteria: a. Timely and accurate dose projections are performed in accordance with Emergency Plan Implementing Procedures during the exercise. 6. Demonstrate the ability to develop appropriate Protective Action Recommendations (PARs) and notify appropriate authorities within 15 minutes, once data is available, after the declaration of a General Emergency or change in PARs during the exercise. Standard Criteria: a. Total Effective Dose Equivalent (TEDE) and Committed Dose Equivalent (CDE) dose

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
				projections from the dose assessment computer code are developed in accordance with Emergency Plan Implementing Procedures during the exercise. b. PARs are developed and transmitted within 15 minutes of data availability during the exercise.
870	C3.8.01.08.01.02	8.1 Licensee conducts a full participation exercise to evaluate major portions of emergency response capabilities, which includes participation by each state and local agency within the plume exposure EPZ, and each state within the ingestion control EPZ. [N.1]	8.1 A full participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.	8.1.2 Onsite emergency response personnel were mobilized in sufficient numbers to fill emergency response positions, and they successfully performed their assigned responsibilities.
871	C3.8.01.08.01.03	8.1 Licensee conducts a full participation exercise to evaluate major portions of emergency response capabilities, which includes participation by each state and local agency within the plume exposure EPZ, and each state within the ingestion control EPZ. [N.1]	8.1 A full participation exercise (test) will be conducted within the specified time periods of Appendix E to 10 CFR Part 50.	8.1.3 The exercise was completed within the specified time periods of Appendix E to 10 CFR Part 50, offsite exercise objectives were met, and there were no uncorrected offsite exercise deficiencies, or a license condition requires offsite deficiencies to be corrected prior to operation above 5% of rated power.

C.3.8.1.9 Implementing Procedures

Table C.3.8-1 (continued) Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Program Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
872	C3.8.01.09.01	9.1 The licensee has submitted detailed implementing procedures for its emergency plan no less than 180 days prior to fuel load	9.1 An inspection of the submittal letter will be performed.	9.1 The licensee submitted detailed implementing procedures for the onsite emergency plan no less than 180 days prior to fuel load.

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C.3.8.2 Piping Rupture Hazards Analysis

Table C.3.8-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
873	C.3.8.02.01	Systems, structures, and components (SSCs), that are required to be functional during and following a design basis event shall be protected against or qualified to withstand the dynamic and environmental effects associated with analyses of postulated failures in high and moderate energy piping.	Inspection of the as-designed pipe rupture hazard analysis report will be conducted. The report documents the analyses to determine where protection features are necessary to mitigate the consequence of a pipe break. Pipe break events involving high-energy fluid systems are analyzed for the effects of pipe whip, jet impingement, flooding, room pressurization, and temperature effects. Pipe break events involving moderate-energy fluid systems are analyzed for wetting from spray, flooding, and other environmental effects, as appropriate.	An as-designed pipe rupture hazard analysis report exists and concludes that the analysis performed for high and moderate energy piping confirms the protection of systems, structures, and components required to be functional during and following a design basis event.

C.3.8.3 Piping Design

Table C.3.8-3
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
874	C.3.8.03.01	The American Society of Mechanical Engineers (ASME) Code, Section III piping is designed in accordance with ASME Code, Section III requirements.	Inspection of the ASME Code Design Reports (NCA-3550) and required documents will be conducted for the set of lines chosen to demonstrate compliance.	The ASME Code Design Report(s) (NCA-3550) (certified, when required by ASME Code) exist and conclude that the design of the piping for lines chosen to demonstrate all aspects of the piping design complies with the requirements of the ASME Code section.

APPENDIX C, ATTACHMENT 1

VIRGIL C. SUMMER NUCLEAR STATION UNIT 2

INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

Security-Related Information – Withheld Under 10 CFR2.390